Millstone Power Station Unit 2
Safety Analysis Report

Chapter 15: License Renewal

CHAPTER 15—LICENSE RENEWAL

Table of Contents

CHAPTER 15—LICENSE RENEWAL

Table of Contents (Continued)

CHAPTER 15—LICENSE RENEWAL

Table of Contents (Continued)

CHAPTER 15—LICENSE RENEWAL

List of Tables

CHAPTER 15 – LICENSE RENEWAL

15.0 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix, which includes the following sections, comprises the FSAR supplement:

- Section 15.1 contains a listing of the aging management programs and the status of the program at the time the License Renewal Application was submitted. This section also contains a list of new aging management programs created after the issuance of the renewed operating license.
- Section 15.2 contains a description of the programs for managing the effects of aging.
- Section 15.3 contains the evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation.
- Section 15.4 contains a summarized description of the programs that support the TLAAs.
- Section 15.5 contains a summarized description of the plant-specific exemptions.
- Section 15.6 contains a matrix of the license renewal commitments.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the unit's previous 40 year operating license expiration date. Unless otherwise identified, references to the Operating License are considered a reference to the Renewed Operating License.

15.1 OVERVIEW OF AGING MANAGEMENT PROGRAMS

15.1.1 AGING MANAGEMENT PROGRAMS

The aging management programs for Millstone Unit 2 are described in the following sections. The programs are either consistent with generally accepted industry methods as discussed in NUREG-1801 (Reference 15.1-1), require enhancements to be consistent with generally accepted industry standards, or are site specific programs.

The following list reflects the status of these programs at the time this section was included in the FSAR and provides a historical perspective of their status at the completion of the NRC review of the License Renewal Application. The implementation status of the listed programs will change as new programs are developed and enhancements to existing programs are completed. Commitments for program additions and enhancements are identified in the appropriate sections.

1. Battery Rack Inspections (Section 15.2.1.1) (Existing - Requires Enhancement).
2. Boraflex Monitoring (Section 15.2.1.2) (Existing).
3. Boric Acid Corrosion (Section 15.2.1.3) (Existing).
4. Buried Pipe Inspection Program (Section 15.2.1.4) (Existing - Requires Enhancement).
5. Chemistry Control for Primary Systems Program (Section 15.2.1.5) (Existing).
6. Chemistry Control for Secondary Systems Program (Section 15.2.1.6) (Existing).
7. Closed-Cycle Cooling Water System (Section 15.2.1.7) (Existing - Requires Enhancement).
8. Electrical Cables and Connectors Not Subject to 10 CFR 50.49 (Section 15.2.1.8) (To Be Developed).
9. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements used in Instrumentation Circuits (Section 15.2.1.9) (Existing - Requires Enhancement).
10. Fire Protection Program (Section 15.2.1.10) (Existing - Requires Enhancement).
11. Flow-Accelerated Corrosion (Section 15.2.1.11) (Existing).
12. Fuel Oil Chemistry (Section 15.2.1.12) (Existing).
13. General Condition Monitoring (Section 15.2.1.13) (Existing - Requires Enhancement).

14. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section 15.2.1.14) (Existing - Requires Enhancement).

15. Infrequently Accessed Areas Inspection Program (Section 15.2.1.15) (To Be Developed).

16. Inservice Inspection Program: Containment Inspections (Section 15.2.1.16) (Existing).

17. Inservice Inspection Program: Reactor Vessel Internals (Section 15.2.1.17) (Existing - Requires Enhancement).

18. Inservice Inspection Program: Systems, Components and Supports (Section 15.2.1.18) (Existing - Enhancement complete).

19. Inspection Activities: Load Handling Cranes and Devices (Section 15.2.1.19) (Existing - Requires Enhancement).

20. Reactor Vessel Surveillance (Section 15.2.1.20) (Existing).

21. Service Water System (Open-Cycle Cooling) (Section 15.2.1.21) (Existing).

22. Steam Generator Structural Integrity (Section 15.2.1.22) (Existing).

23. Structures Monitoring Program (Section 15.2.1.23) (Existing - Requires Enhancement).

24. Tank Inspection Program (Section 15.2.1.24) (Existing - Requires Enhancement).

25. Work Control Process (Section 15.2.1.25) (Existing - Requires Enhancement).

26. Bolting Integrity Program (Section 15.2.1.26) (Existing).

15.1.2 TIME LIMITED AGING ANALYSES AGING MANAGEMENT PROGRAMS:

1. Electrical Equipment Qualification (Section 15.4.1) (Existing).

2. Metal Fatigue of Reactor Coolant Pressure Boundary (Section 15.4.2) (Existing).

15.1.3 AGING MANAGEMENT PROGRAMS CREATED AFTER ISSUANCE OF THE RENEWED OPERATING LICENSE

1. Alloy 600 Management Program (Section 15.2.1.27).

15.1.4 REFERENCES FOR SECTION 15.1

15.1-1 NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U. S. Nuclear Regulatory Commission, July 2001

15.2 PROGRAMS THAT MANAGE THE EFFECTS OF AGING ON STRUCTURES AND COMPONENTS WITHIN THE SCOPE OF LICENSE RENEWAL

This section provides summaries of the programs credited for managing the effects of aging on structures and components within the scope of license renewal.

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in NUREG-1800, Section A.2. The Quality Assurance program includes the elements of corrective action, confirmation process, and administrative controls and is applicable to the safety-related and non safety-related structures, and components that are within the scope of license renewal.

15.2.1 AGING MANAGEMENT PROGRAMS

15.2.1.1 Battery Rack Inspections

Program Description

Battery Rack Inspections is a plant-specific program that manages the aging effect of loss of material. The structural integrity of the support racks for the station batteries, within the scope of license renewal, is verified by visually inspecting for loss of material.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inclusion of In-Scope Battery Racks

  The existing inspection program has been modified to include those battery racks that require monitoring for license renewal, but were not originally included in the program. Inclusion of those battery racks completes the actions required for Commitment Item 1 in Table 15.6-1.

- Inspection Criteria

  Implementing procedures have been modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks for all in-scope batteries. Revision of those procedures completes the actions required for Commitment Item 2 in Table 15.6-1.

15.2.1.2 Boraflex Monitoring

Program Description

Boraflex is no longer credited as a neutron absorber. Thus, Boraflex monitoring is no longer performed.

15.2.1.3 Boric Acid Corrosion

Program Description

Boric Acid Corrosion corresponds to NUREG-1801, Section XI.M10 “Boric Acid Corrosion.” The program manages the aging effect of loss of material and ensures that systems, structures, and components susceptible to boric acid corrosion are properly monitored. The program uses visual inspections to detect the boric acid leakage source, path, and any targets of the leakage. It ensures that boric acid corrosion is consistently identified, documented, evaluated, trended, and effectively repaired. The Boric Acid Corrosion program provides both detection and analysis of leakage of borated water inside containment. The General Condition Monitoring program is the primary method for detecting borated water leakage outside containment. The analysis of the leakage is performed through the Boric Acid Corrosion program. Any necessary corrective actions are implemented through the Corrective Action Program.

Boric Acid Corrosion program implements the requirements of:

- NRC Bulletin 2001-01 (Reference 15.2-15)
- NRC Bulletin 2002-01 (Reference 15.2-16)
- NRC Bulletin 2002-02 (Reference 15.2-17)
- NRC Bulletin 2003-02 (Reference 15.2-18)
- NRC Order EA-03-009 (Reference 15.2-19)
- NRC Bulletin 2004-01 (Reference 15.2-20)

The acceptance criterion is the absence of any boric acid leakage or precipitation. If boric acid leakage or precipitation is found by any personnel, it is required to be reported using the

Corrective Action Program. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.4 Buried Pipe Inspection Program

Program Description

The Buried Pipe Inspection Program is an existing program that corresponds to NUREG-1801, Sections XI.M28, "Buried Piping and Tanks Surveillance" and XI.M34, "Buried Piping and Tanks Inspection." The program manages the aging effect of loss of material through the use of preventive measures and inspections. The inspections will be performed when the piping and components are excavated for maintenance or for any other reason.

There are no buried tanks within the scope of license renewal.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. In addition to visual inspections, the field inspections for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Baseline Inspection

  A baseline inspection of the in-scope buried piping located in a damp soil environment has been performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching has been performed by visual, and mechanical or other appropriate methods. These inspections complete the actions required for Commitment Item 3 in Table 15.6-1.

- Buried Piping Inspections

  The maintenance and work control procedures have been revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures include the inspection for the loss of material due to selective leaching, which will be performed by visual, and mechanical or other appropriate methods. These changes complete the actions required for Commitment Item 4 in Table 15.6-1.

15.2.1.5 Chemistry Control for Primary Systems Program

Program Description

Chemistry Control for Primary Systems Program corresponds to NUREG-1801, Section XI.M2, “Water Chemistry.” The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, fluorides, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-105714 (Reference 15.2-1) for primary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.6 Chemistry Control for Secondary Systems Program

Program Description

Chemistry Control for Secondary Systems Program corresponds to NUREG-1801, Section XI.M2, “Water Chemistry.” The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, sodium, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-102134 (Reference 15.2-2) for secondary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.7 Closed-Cycle Cooling Water System

Program Description

Closed-Cycle Cooling Water System corresponds to NUREG-1801, Section XI. M21, “Closed-Cycle Cooling Water System.” The program manages the aging effect of loss of material through the maintenance of process fluid chemistry and performance monitoring of closed-cycle cooling water systems to ensure parameters remain within acceptable limits. The program is based directly on guidance contained in EPRI Report TR-107396 (Reference 15.2-3).

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality

Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following commitment has been implemented prior to the period of extended operation:

- Heat Exchanger Baseline Inspection

  A baseline visual inspection has been performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one:

  - Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger,
  - Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and
  - Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.

The performance of these inspections complete the actions required for Commitment Item 29 in Table 15.6-1.

15.2.1.8 Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements" as modified by NRC Interim Staff Guidance-05 (Reference 15.2-4). This program manages the aging effects of cracking and embrittlement to ensure that electrical cables, connectors, and fuse holders within the scope of license renewal that are exposed to an adverse localized environment (but not subject to the environmental qualification requirements of 10 CFR 50.49) are capable of performing their intended function. Adverse localized environments may be caused by heat, radiation or moisture.

The acceptance criterion for the visual inspections of accessible non-EQ cable jackets and connector coverings is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following actions will be implemented prior to the period of extended operation:

- Program Implementation

  The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program has been established.

  This completes the action required to complete Commitment Item 5 in Table 15.6-1, License Renewal Commitments.

- Inclusion of In-Scope Fuse Holders

  Fuse holders meeting the requirements have been evaluated. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The aging management review of fuse holders considered the aging stressors for metallic clips and concluded that there were no aging effects that require management.

  This completes the actions required to complete Commitment Item 6 in Table 15.6-1, License Renewal Commitments.

15.2.1.9 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Program Description

Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits corresponds to NUREG-1801, Section XI.E2, "Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits" and the program as modified in draft NRC ISG-15 (Reference 15.2-5). This program manages the aging effects of cracking and embrittlement for electrical cables within the scope of license renewal that are used in circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation (but not subject to the environmental qualification requirements of 10 CFR 50.49), and are installed in adverse localized environments caused by heat, radiation or moisture.

The acceptance criterion for the calibration readings is the loop-specific tolerances established in Technical Specifications and surveillance procedures. Where calibration of the instrumentation is not performed in situ, the acceptance criteria for each test are defined by the specific type of test performed and the specific cable tested. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Testing of Cables for Instruments That Are Not Calibrated In Situ

  Procedures have been developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ. The first tests have been completed. The frequency of subsequent tests will be based on Engineering evaluation and will not exceed a 10 year interval. This completes the action required to complete commitment Item 7 in Table 15.6-1, License Renewal Commitments.

- Review of Surveillance Test Results for Cables Tested In Situ

  Calibration results for cables tested in situ have been reviewed to detect severe aging degradation of the cable insulation, and include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years. This completes the action required to complete commitment Item 32 in Table 15.6-1, License Renewal Commitments.

15.2.1.10 Fire Protection Program

Program Description

The Fire Protection Program is an existing program and corresponds to NUREG-1801, Sections XI.M26, “Fire Protection” and XI.M27, “Fire Water System” and to the revised XI.M27, “Fire Water System” program described in NRC Interim Staff Guidance (ISG)-04 (Reference 15.2-6). The program manages the aging effects of loss of material, cracking, and change of material properties for plant fire protection features and components. The program manages these aging effects through the use of periodic inspections and tests.

The program also manages the aging effects for the diesel-driven fire pump fuel supply line, the reactor coolant pump oil collection systems, and Appendix R support equipment.

Visual inspection of fire protection piping internal surfaces that are exposed to water is performed when the system is opened for maintenance and/or repair. The Work Control Process provides guidance for the performance of internal inspections of fire protection piping and components whenever the system is opened for maintenance or repair.

The acceptance criteria for the Fire Protection Program are:

- For visual inspections, the absence of anomalous indications that are signs of degradation.

- For fire barriers and fire doors, the sizes for breaks, holes, cracks, spalling gaps, and/or clearances are in accordance with the limits established in the inspection procedures.

- For fire protection equipment performance tests (i.e., flow and pressure tests), acceptance criteria are provided in the appropriate surveillance procedures.

Additionally, the fire protection water system pressure is continuously monitored to be above the minimum setpoint. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Baseline Fire Protection Inspections

  A baseline visual inspection has been performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation. These inspections complete the action required for Commitment Item 8 in Table 15.6-1.

The following program enhancement will be implemented prior to the sprinkler heads achieving 50 years of service life:

- Testing or Replacement of Sprinkler Heads

  Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years has been included in the Fire Protection Program. The first tests will be completed prior to the sprinkler heads achieving 50 years of service life. The frequency of subsequent tests will not exceed a 10 year interval. This change completes the action required for Commitment Item 9 in Table 15.6-1, License Renewal Commitments.

15.2.1.11 Flow-Accelerated Corrosion

Program Description

Flow-Accelerated Corrosion Program corresponds to NUREG-1801, Section XI.M17, "Flow-Accelerated Corrosion." The program manages the aging effect of loss of material in accordance with the EPRI guidelines in NSAC-202L (Reference 15.2-7). It includes procedures or administrative controls to assure that the structural integrity of carbon steel and low-alloy steel piping and components, such as valves, steam traps, and feedwater heaters, is maintained.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.12 Fuel Oil Chemistry

Program Description

Fuel Oil Chemistry corresponds to NUREG-1801, Section XI.M30, "Fuel Oil Chemistry." The program manages the aging effect of loss of material by monitoring and controlling fuel oil quality to ensure that it is compatible with the materials of construction for in-scope components containing diesel fuel oil.

The Fuel Oil Chemistry program uses the following industry standards as the basis for the program:

ASTM Standard D 1796 (Reference 15.2-8),

ASTM Standard D 2276 (unmodified) (Reference 15.2-9), and

ASTM Standard D 4057 (Reference 15.2-10).

The acceptance criterion is adherence to the specific guidelines and limits defined in related plant procedures for parameters that have been shown to contribute to component degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.13 General Condition Monitoring

Program Description

General Condition Monitoring is a plant specific program that manages the aging effects of loss of material, change of material properties, and cracking on the external surfaces of components. It is performed in accessible plant areas for components and structures including those within the scope of license renewal and involves visual inspections for evidence of age related degradation. General Condition Monitoring is implemented by Health Physics technicians, System Engineers, and Plant Equipment Operators while performing their routine in-plant activities.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Procedure and Training Enhancements

  The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns have been enhanced to provide expectations that identify the requirements for the inspection of aging effects. These actions complete the requirements for Item 10 in Table 15.6-1.

15.2.1.14 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E3, “Inaccessible Medium-Voltage Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements.” This program manages the aging effect of formation of water trees and ensures that inaccessible medium-voltage (2 kV to 15 kV) electrical cables within the scope of license renewal (but not subject to the environmental qualification requirements of 10 CFR 50.49) that have been submerged, remain capable of performing their intended function. The program considers the combined effects of submergence, simultaneous with a significant voltage exposure. Significant voltage exposure is defined as being subjected to system voltage for more the twenty-five percent of the time.

The acceptance criterion for the inspections performed under the Structures Monitoring Program is to confirm that in-scope, medium-voltage cables have not become submerged. In-scope cable found to be submerged in standing water for an extended period of time will be subject to an engineering evaluation and corrective action. The evaluation will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The test will use a proven methodology for detecting deterioration of the insulation due to wetting. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Verification Testing

  In scope cable found to be submerged has been subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time was based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation also addresses the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test used a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Cables, or other appropriate testing. Testing has acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality was entered into the Corrective Actions Program. This completes the actions required to complete commitment Item 11 in Table 15.6-1, License Renewal Commitments.

- Testing of Inaccessible Medium Voltage Cables

  The in-scope cables in Unit 3 duct lines number 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and number 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C, 35D) have been tested to demonstrate that water treeing will not prevent the cables from performing their intended function. Subsequent testing has been scheduled to be performed on a frequency not to exceed a 10 year interval.

  This completes the actions required to complete commitment Item 33 in Table 15.6-1, License Renewal Commitments.

- Sample Testing of Inaccessible Medium Voltage Cables

  In addition to the testing specified in Commitment 33, a representative sample of in-scope medium voltage cables has been tested to demonstrate that water treeing will not prevent the cables from performing their intended function. Subsequent testing has been scheduled to be performed on a frequency not to exceed a 10 year interval.

15.2.1.15 Infrequently Accessed Areas Inspection Program

Program Description

Infrequently Accessed Areas Inspection Program is a plant-specific program that manages the aging effects of loss of material, change of material properties, and cracking. The program uses visual inspections of the external surfaces of in-scope structures and components located in infrequently accessed areas of the plant.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements have been implemented prior to the period of extended operation:

- Program Implementation

  The Infrequently Accessed Inspection Program has been established.

  The establishment of the Infrequently Accessed Inspection Program completes the action required for Commitment Item 12 in Table 15.6-1.

15.2.1.16 Inservice Inspection Program: Containment Inspections

Program Description

Inservice Inspection Program: Containment Inspections corresponds to the following NUREG-1801 program descriptions:

- Section XI.S1, “ASME Section XI, Subsection IWE”,
- Section XI.S2, “ASME Section XI, Subsection IWL”, and
- Section XI.S4, “10 CFR Part 50, Appendix J.”

The program manages the aging effects of loss of material, change of material properties, and cracking. The program is consistent with ASME Section XI, Subsections IWE and IWL, and 10 CFR 50.55a(b)(2), which provide the criteria for ISI Containment inspections.

Appendix J Leakage Rate Testing is included as part of the Inservice Inspection Program: Containment Inspections. The Containment Appendix J Leakage Rate Test Program implements Type A and B tests to measure the overall primary Containment integrated leakage rate.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Containment Inspections are based on the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.17 Inservice Inspection Program: Reactor Vessel Internals

Program Description

Inservice Inspection Program: Reactor Vessel Internals corresponds to the following NUREG-1801 program descriptions:

- Section XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”.
- Section XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”.
- Section XI.M16, “PWR Vessel Internals”.

The Inservice Inspection Program: Reactor Vessel Internals manages the effects of aging for those reactor internals that are susceptible to loss of material, cracking, loss of preload, change in dimension and loss of fracture toughness (which presents itself as cracking due to embrittlement).

Industry groups are in place whose objectives include the investigation of the aging effects applicable to reactor vessel internals regarding such items as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC and IASCC), and loss of preload for baffle and former-assembly bolts.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Reactor Vessel Internals are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be implemented at least two years prior to period of extended operation:

- Reactor Vessel Internals Inspections

  Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), and stress corrosion cracking (PWSCC and IASCC) and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, has been submitted to the NRC for approval.

The submission of the program description to the NRC completes the required actions for Commitment Item 13.

15.2.1.18 Inservice Inspection Program: Systems, Components and Supports

Program Description

Inservice Inspection Program: Systems, Components and Supports corresponds to the following NUREG-1801 program descriptions:

- Section XI.M1, “ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD”,
- Section XI.M3, “Reactor Head Closure Studs”,
- Section XI.M12, “Thermal Aging Embrittlement of Cast Austentic Stainless Steel (CASS)”, and
- Section XI.S3, “ASME Section XI, Subsection IWF.”

The Inservice Inspection Program: Systems, Components and Supports is an existing program that was developed to comply with the requirements of ASME Boiler and Pressure Vessel Code, Section XI (Reference 15.2-11). The ASME program provides the requirements for ISI, repair, and replacement for all Class 1, 2 and 3 components and the associated component supports. For license renewal, the Millstone program has been credited to manage the effects of aging for only Class 1 and specific Class 2 components (on the secondary side of the steam generators as determined through the aging management review process) and for Class 1, 2, and 3 components supports. Inservice Inspection Program: Systems, Components and Supports manages the aging effects of cracking, loss of fracture toughness, loss of material and loss of preload.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Systems, Components and Supports are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be taken prior to the period of extended operation:

- Monitoring Fracture Toughness

For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3.

An engineering evaluation determined that CASS components susceptible to thermal aging embrittlement exist only in the pressurizer surge line. The evaluation determined that the surge line is highly flaw tolerant but recommended volumetric examinations to confirm that no flaws exist that are deeper than the allowable flaw depths. Successful inspections of the pressurizer surge line during the M2R22 refueling outage confirmed that no adverse flaws exist. Those inspections complete the actions required for Commitment Item 27, Table 15.6-1.

- Pressurizer Replacement

  Dominion has replaced the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC. Replacing the pressurizer completes the action required for Commitment Item 36.

15.2.1.19 Inspection Activities: Load Handling Cranes and Devices

Program Description

Inspection Activities: Load Handling Cranes and Devices corresponds to NUREG-1801, Section XI. M23, “Inspection of Overhead Heavy Load [Related to Refueling] Handling Systems.” The program manages the aging effect of loss of material for the load handling cranes and devices within the scope of license renewal. The in-scope load handling cranes and devices are either safety-related or seismically designed to ensure that they will not adversely impact safety-related components during or subsequent to a seismic event.

Inspection Activities: Load Handling Cranes and Devices addresses the overall condition of the crane or device, including checking the condition of the structural members (i.e., rails, girders, etc.) and fasteners on the crane or device, the runways along which the crane or device moves, and the base plates and anchorages for the runways and monorails.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements have been implemented prior to the period of extended operation:

- Inclusion of In-Scope Lifting Devices

  The existing inspection program has been modified to include all lifting devices that require monitoring for license renewal. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 15.

- Inspection Criteria

  Implementing procedures and documentation have been modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 16.

15.2.1.20 Reactor Vessel Surveillance

Program Description

Reactor Vessel Surveillance corresponds to NUREG-1801, Section XI.M31 “Reactor Vessel Surveillance”. The Reactor Vessel Surveillance program manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy subcomponents in the beltline region of the reactor vessel. Neutron dosimetry and material properties data derived from the reactor vessel materials’ irradiation surveillance program are used in calculations and evaluations that demonstrate compliance with applicable regulations. This program ensures compliance with Technical Requirements Manual requirements that surveillance specimens are removed and examined at predetermined intervals established in the Technical Specification to monitor the changes in the material properties and the results of the examinations used to update the Technical Specification operating limits.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.21 Service Water System (Open-Cycle Cooling)

Program Description

The Service Water System (Open-Cycle Cooling) program corresponds to NUREG-1801, Section XI.M20, “Open Cycle Cooling Water System.” The program manages the aging effects of loss of material and buildup of deposits. The program implements the NRC guidelines in Generic

Letter 89-13 (Reference 15.2-12), which includes (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion (including microbiologically influenced corrosion), erosion, protective coating failure, silting, and biofouling do not degrade the performance of safety-related systems serviced by Service Water System; (d) a system walkdown inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures. In lieu of thermal performance testing, Millstone Unit 2 relies on frequent, regular inspection and cleaning of heat exchangers to preclude fouling.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.22 Steam Generator Structural Integrity

Program Description

Steam Generator Structural Integrity corresponds to NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity Program.” This program manages the aging effects of loss of material and cracking and adopts the performance criteria and guidance for monitoring and maintaining steam generator tubes as defined in NEI 97-06 (Reference 15.2-13). The program incorporates performance criteria for structural integrity, accident-induced leakage, and operational leakage. The program includes preventive measures to mitigate degradation through the control of primary and secondary side water chemistry; assessment of degradation mechanisms; inservice inspection of the steam generator tubes to detect degradation; evaluation and plugging or repair, as needed; and leakage monitoring to ensure the structural and leakage integrity of the pressure boundary.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.23 Structures Monitoring Program

Program Description

Structures Monitoring Program corresponds to the following NUREG-1801 program descriptions:

- Section XI.S5 “Masonry Wall Program”,
- Section XI.S6 “Structures Monitoring Program”, and

- Section XI.S7 “R. G. 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants”.

The Structures Monitoring Program manages the aging effects of loss of material, change of material properties, and cracking by the monitoring of structures and structural support systems that are in the scope of license renewal. The majority of these structures and structural support systems are monitored under 10 CFR 50.65 (Reference 15.2-14). Other structures in the scope of license renewal (such as non-safety related buildings and enclosures, duct banks, valve pits and trenches, HELB barriers, and flood gates) are also monitored to ensure there is no loss of intended function.

The scope includes all masonry walls and water-control structures identified as performing intended functions in accordance with 10 CFR 54.4.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Modification of Structures Monitoring Program procedures

  NUREG-1801 recommends the use of ACI 349.3R-96 and ANSI/ASCE 11-90, as a reference for recommendations for the development of an evaluation procedure for nuclear safety-related concrete structures and existing buildings. These documents were not used or referenced as a standard for establishing the Structures Monitoring Program. The implementing procedure for the Structures Monitoring Program has been modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program. Modification of the implementing procedure completes the action required for Commitment Item 17 in Table 15.6-1.

- Addition of Structures to the Structures Monitoring Program

  The Structures Monitoring Program did not initially monitor all structures in-scope for license renewal. The Structures Monitoring Program and the implementing procedure have been modified to include all in-scope structures. Modification of the Program and the implementing procedure completes the actions required for Commitment Item 18 in Table 15.6-1.

- Sampling of Groundwater

Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade. The implementing procedure for the Structures Monitoring Program has been revised to direct groundwater sampling. Modification of the implementing procedure completes the action required for Commitment Item 19 in Table 15.6-1.

- Engineering Notification of Submerged Medium Voltage Cables

  The Structures Monitoring Program and implementing procedures have been modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged. Modification of the implementing procedure for the Structures Monitoring Program completes the action required for Commitment Item 20 in Table 15.6-1.

- Inspection of Normally Inaccessible Areas That Become Accessible

  Initially, the license renewal implementation plans stated that the maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason. The procedure revision task is accomplished by a modification of the Structures Monitoring implementing procedure for the situation of an inaccessible area becoming accessible by excavation or installation of shielding. Modification of the implementing procedure completes the action required for Commitment Item 21 in Table 15.6-1.

15.2.1.24 Tank Inspection Program

Program Description

Tank Inspection Program corresponds to NUREG-1801, Section XI.M29, “Aboveground Carbon Steel Tanks.” The program manages the aging effect of loss of material through periodic internal and external tank inspections. The program includes inspections of the sealant and caulking in and around the tank and the concrete foundation and evaluations to monitor the condition of coatings, linings, and structural elements, to prevent deterioration of the tanks to unacceptable levels. The program also includes volumetric examination of inaccessible locations, such as the external surfaces of tank bottoms.

The acceptance criterion for visual inspections of paint, coatings, sealant, caulking, and structural elements is the absence of anomalous indications that are signs of degradation. Thickness measurements of the tank walls and bottoms are evaluated against design thickness, established baseline values, or loss of material allowances. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inspection of sealants and caulkings

  Appropriate inspections of sealants and caulkings used for moisture intrusion prevention in and around aboveground tanks have been performed. These inspections complete the actions required for Commitment Item 22 in Table 15.6-1.

- Non-destructive Volumetric Examination of Inaccessible Tank Bottoms

  Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, have been performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities. These examinations and scheduled subsequent inspections complete the actions required for Commitment Item 23 in Table 15.6-1.

- Tanks Being Added to Tank Inspection Program

  The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and have been included in the respective Tank Inspection Program inspection plan. These changes complete the actions required for Commitment Item 24 in Table 15.6-1.

15.2.1.25 Work Control Process

Program Description

Work Control Process is a plant specific program that integrates and coordinates the combined efforts of Maintenance, Engineering, Operations, and other support organizations to manage maintenance activities. The Work Control Process is utilized to manage the aging effects of loss of material, change of material properties, cracking, and buildup of deposits for components and plant commodities within the scope of license renewal. Performance testing and maintenance activities, both preventive and corrective, are planned and conducted in accordance with the Work Control Process. The Work Control Process also provides opportunities to collect oil and engine coolant fluid samples for subsequent analysis of contaminants and chemical properties, which could either indicate or affect aging.

In addition to visual inspections, the field inspection for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping.

The acceptance criterion for visual inspections is the absence of anomalous signs of degradation. The acceptance criteria for testing or sampling are specified in the various station procedures and/or vendor technical manuals or recommendations. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Performance of Inspections During Maintenance Activities

  Changes have been made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities. These changes complete the actions required for Commitment Item 25 in Table 15.6-1.

- Selective Leaching Inspection

  Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching have been performed on a representative sample of buried pipe locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation. Susceptible aluminum-bronze valves have been replaced using a material (AL6XN stainless steel) that is more resistant to selective leaching. The buried pipe inspections and valve replacements complete the actions required for Commitment Item 30 in Table 15.6-1.

- Verification of Scope

  A review of the Work Control Process inspection opportunities for each material and environment group supplemental to the initial review conducted during the development of the LRA has been performed. No additional baseline inspections were needed to be performed for the material and environment combinations inspected as part of the Work Control Process. This action completes the requirements for Item 31 in Table 15.6-1.

15.2.1.26 Bolting Integrity Program

Program Description

The Bolting Integrity Program corresponds to NUREG-1801, Section XI.M18, “Bolting Integrity”. The program manages the aging effects of cracking, loss of material and loss of preload.

This is accomplished by establishing good bolting practices in accordance with EPRI NP-5067, Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual, and Volume 2: Small Bolts and Threaded Fasteners and EPRI TR-04213, Bolted Joint Maintenance and Application Guide. For ASME Class bolting, aging effects are additionally managed by the performance of inservice examinations in accordance with ASME Section XI, Subsections IWB, IWC, IWD, and IWF.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.27 Alloy 600 Management Program

Program Description

The Alloy 600 Management Program manages the aging effect of primary water stress corrosion cracking (PWSCC) in Alloy 600 base metal and Alloy 82/182 dissimilar metal welds. The Alloy 600 Management Program is consistent with the Nickel-Alloy Nozzles and Penetrations aging management program described in Chapter XI of NUREG-1801, Rev. 0 (GALL Report). The program monitors susceptible components for indications of PWSCC before there is a loss of intended function. Since the program is based on more recent industry guidance, EPRI TR-1009561, MRP-126 (Reference 15.2-21), the program is enhanced from the program described in Rev. 0 of the GALL. The program scope has been increased according to industry guidelines to include all Alloy 600/82/182 components in the primary system. The inspection requirements have been updated to the requirements of ASME Section XI (Reference 15.2-11) Code Cases N-722-1, N-729-1 and N-770-1.

The Alloy 600 Management Program credits inspections performed under existing inspection processes. Inspection requirements for each type of component are listed in the Alloy 600 Management Plan. Defects found during inspections will be dispositioned in accordance with applicable regulatory and code requirements, utilizing the corrective action system.

Commitments

The following action will be implemented at least two years prior to the period of extended operation:

- PWSCC of Nickel Based Alloys

Millstone will follow the industry efforts investigating the aging effects applicable to nickel based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description was submitted two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 54.21(a)(3).

The submission of the Alloy 600 Management Program description to the NRC (Reference 15.6-3) completes the required actions for Commitment Item 14 in Table 15.6-1.

15.2.2 REFERENCES FOR SECTION 15.2

15.2-1 TR-105714, PWR Primary Water Chemistry Guidelines, Technical Report, Revision 3, Electric Power Research Institute.

15.2-2 TR-102134, PWR Secondary Water Chemistry Guidelines, Technical Report, Revision 3, Electrical Power Research Institute.

15.2-3 EPRI TR-107396, Closed Cooling Water Chemistry Guideline, Technical Report, Electrical Power Research Institute, Palo Alto, CA, November 1997.

15.2-4 NRC Interim Staff Guidance (ISG)-05, The Identification And Treatment Of Electrical Fuse Holders For License Renewal, U.S. Nuclear Regulatory Commission, March 10, 2003.

15.2-5 Letter from Pao-Tsin Kuo, Nuclear Regulatory Commission, to Alex Marion, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, Proposed Interim Staff Guidance (ISG)-15: Revision of Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) X1.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits”, August 12, 2003.

15.2-6 NRC Interim Staff Guidance (ISG)-04, “Aging Management of Fire Protection Systems for License Renewal”, U.S. Nuclear Regulatory Commission, December 3, 2002.

15.2-7 NSAC-202L-R4 Recommendations for an Effective Flow Accelerated Corrosion Program, Electric Power Research Institute, November, 2013.

15.2-8 ASTM D 1796, Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method, American Society for Testing Materials, West Conshohocken, PA.

15.2-9 ASTM D 2276, Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling, American Society for Testing Materials, West Conshohocken, PA.

15.2-10 ASTM D 4057, Standard Practice for Manual Sampling of Petroleum and Petroleum Products, American Society for Testing Materials, West Conshohocken, PA.

15.2-11 ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers.

15.2-12 Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment, Nuclear Regulatory Commission, July 18, 1989 (Supplement 1 dated 4/4/90).

15.2-13 NEI 97-06, Steam Generator Program Guidelines, Technical Report, Nuclear Energy Institute.

15.2-14 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.2-15 NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, U.S. Nuclear Regulatory Commission, August 3, 2001.

15.2-16 NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, March 18, 2002.

15.2-17 NRC Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs, U.S. Nuclear Regulatory Commission, August 9, 2002.

15.2-18 NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, 08/21/03.

15.2-19 NRC Order EA-03-009, Issuance Of Order Establishing Interim Inspection Requirements For Reactor Pressure Vessel Heads At Pressurized Water Reactors, U.S. Nuclear Regulatory Commission, February 11, 2003."

15.2-20 NRC Bulletin 2004-01, Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, May 28, 2004.

15.2-21 EPRI Report 1009561, Materials Reliability Program: Generic Guidance for Alloy 600 Management (MRP-126), Electric Power Research Institute, Palo Alto, CA November 2004.

15.3 TIME-LIMITED AGING ANALYSIS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

15.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in FSAR Section 4.3.1. Time-limited aging analyses (TLAAs) applicable to the reactor vessel are:

- Upper-shelf energy (USE)
- Pressurized thermal shock (PTS)
- Pressure-temperature limits

The Reactor Vessel Surveillance program manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time dependent parameters used in the aging analyses for pressurized thermal shock, upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

The reactor vessel neutron embrittlement evaluations have been based on 54 effective full power years of operation. 54 effective full power years would be reached at the end of the period of extended operation (60 years) assuming a capacity factor of 90% for the lifetime of the unit.

15.3.1.1 Upper Shelf Energy

10 CFR 50, Appendix G contains screening criteria that establish limits on how far the upper shelf energy values for a reactor pressure vessel material may be allowed to drop due to neutron irradiation exposure. The regulation requires the initial upper shelf energy value to be greater than 75 ft-lbs in the unirradiated condition and for the value to be greater than 50 ft-lbs in the fully irradiated condition as determined by Charpy V-notch specimen testing throughout the licensed life of the plant. Upper shelf energy values of less than 50 ft-lbs may be acceptable to the NRC if it can be demonstrated that these lower values will provide margins of safety against brittle fracture equivalent to those required by ASME Section XI, Appendix G.

Acceptable upper shelf energy values have been calculated in accordance with Regulatory Guide 1.99, Revision 2 to the end of the period of extended operation. Calculated upper shelf energy values for the most limiting reactor pressure vessel beltline plate and weld materials remain greater than 50 ft-lbs.

15.3.1.2 Pressurized Thermal Shock

Reactor pressure vessel beltline fluence is one of the factors used to determine the margin to reactor pressure vessel pressurized thermal shock as a result of radiation embrittlement. The margin is the difference between the maximum nil ductility reference temperature in the limiting beltline material (RT_{PTS}) and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forging and axial weld materials, and 300°F for beltline circumferential weld materials.

Acceptable RT_{PTS} values have been calculated in accordance with Regulatory Guide 1.99, Revision 2, requirements to the end of the period of extended operation.

15.3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits identify the maximum allowable pressure as a function of reactor coolant temperature. As the pressure vessel becomes irradiated and its fracture toughness is reduced, the allowable pressure at low temperatures is reduced. Therefore, in order to heatup and cooldown the Reactor Coolant System, the reactor coolant temperature and pressure must be maintained within the limits of Appendix G as defined by the reactor vessel fluence.

Heatup and cooldown limit curves have been calculated using the adjusted reference temperature nil-ductility transition or adjusted reference temperature (adjusted RT_{NDT} or ART) corresponding to the limiting beltline material of the reactor pressure vessel for the current period of licensed operation. Current low temperature overpressure protection (LTOP) system heatup and cooldown limit curves were approved in License Amendment 218.

In accordance with 10 CFR 50, Appendix G, updated pressure-temperature limits for entering the period of extended operation have been developed and implemented prior to the period of extended operation. Low temperature overpressure protection system enable temperature requirements were found to remain valid and will ensure that the pressure-temperature limits will not be exceeded for postulated plant transients during the period of extended operation. Millstone Unit 2 will calculate USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation.

Commitments

The following actions will be taken prior to the period of extended operation.

Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by regulatory guidance changes, were required to be submitted to the NRC for review at least two years prior to the period of extended operation. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 37.

Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements were submitted to the NRC for review in 2005 (Reference 15.6-4) and approved in 2006 (Reference 15.6-5).

The submittal of the updated USE, RT_{PTS}, and P-T limits completes the required actions for Commitment Item 37 in Table 15.6-1.

15.3.2 METAL FATIGUE

Fatigue is defined as structural deterioration that can occur through repeated stress or strain cycles resulting from fluctuations in loads and/or temperatures. After repeated cyclic loading of sufficient magnitude, micro-structural damage can accumulate leading to microscopic crack initiation at the most highly affected locations. Fatigue cracks typically initiate at points of maximum local stress ranges and minimum local strength. Further cyclic mechanical and/or thermal loading can lead to crack growth.

Fatigue represents an aging mechanism. As such, fatigue evaluations represent a time-limited aging analysis even though the system, structure and component design limits are based upon the number of cycles and the associated fatigue (cumulative) usage factors rather than specific time limits.

15.3.2.1 Millstone Unit 2 Class 1 Components

Components within the Millstone Unit 2 nuclear steam supply system are subject to a wide variety of varying mechanical and thermal loads that contribute to fatigue accumulation. The Reactor Coolant System components are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III (Reference 15.3-1) and ANSI Standard B31.7 (Reference 15.3-2). Use of these codes requires that design analyses for Class A (Class 1) systems and components address fatigue and the establishment of load limits to preclude initiation of fatigue cracks.

The type and number of Reactor Coolant System design transients have been identified. In all instances, the number of Reactor Coolant System design transients assumed in the original design were found to be acceptable for the period of extended operation.

NRC Bulletin 88-08 identified a concern regarding potential temperature stratification or temperature oscillations in unisolable sections of piping attached to the Reactor Coolant System. Based upon the Millstone Unit 2 response (Reference 15.3-3) and supplemental communications, the NRC concluded that Millstone Unit 2 meets the requirements of Bulletin 88-08 (Reference 15.3-4).

Pressurizer surge line thermal stratification was a concern raised by the NRC in Bulletin 88-11. One of the requirements of this bulletin was to analyze the effects of thermal stratification on surge line integrity. These analyses were collectively performed as a Combustion Engineering Owners Group task (Reference 15.3-5) supplemented by additional unit specific inspections and activities. Based upon the Combustion Engineering Owners Group task, the NRC concluded that the owners group analysis CEN-387-P is bounding for thermal stratification and thermal striping

(Reference 15.3-7). Confirmation that this analysis is bounding for Millstone Unit 2 and confirmation that the actions required by Bulletin 88-11 have been completed was provided to the NRC (Reference 15.3-6).

Thermal aging refers to changes in the microstructure and properties of a susceptible material due to prolonged exposure to elevated temperatures above approximately 480 °F. Reactor Coolant System temperatures exceed this threshold. At these temperatures, the hardness of potentially susceptible Cast Austenetic Stainless Steel (CASS) materials increase while their ductility, impact strength and more importantly, their fracture toughness, decrease. Fracture toughness is one of the more important design inputs in a leak-before-break and a flaw tolerance evaluation, performed to ensure protection of the reactor coolant system against guillotine pipe breaks throughout plant life. The degree of change in fracture toughness (thermal embrittlement) is dependent on the time of exposure to these elevated temperatures.

Acceptable thermal and pressure transients, and operating cycles have been projected for ASME Section III Class A, Class 1 and ANSI Standard B31.7 Class 1 components though the period of extended operation.

15.3.2.2 Non-Class 1 Components

Non-Class 1 components can include ASME Section III Classes 2 and 3, ANSI Standard B31.7 Classes 2 and 3, and ANSI Standard B31.1 (Reference 15.3-8) piping and tubing. Piping systems designed to these requirements (e.g., sample lines) incorporate a stress range reduction factor to conservatively address the effects of thermal cycling on fatigue. For those sample lines projected to experience greater than 7,000 equivalent full-temperature thermal cycles, actual expansion stresses did not exceed allowable expansion stresses.

Acceptable numbers of thermal cycles and acceptable expansion stresses have been projected to the end of the period of extended operation.

15.3.2.3 Environmentally Assisted Fatigue

The effect of reactor coolant environment on fatigue is generally referred to as environmentally assisted fatigue. As part of an industry effort to address environmental effects on operating nuclear power plants during the current 40 year licensing term, Idaho National Engineering Laboratories evaluated fatigue-sensitive component locations at plants designed by all four domestic nuclear steam supply system vendors. These evaluations are presented and discussed in NUREG/CR-6260 (Reference 15.3-9). The evaluations associated with the newer-vintage Combustion Engineering plants are applicable, since the majority of the Millstone Unit 2 Class 1 systems and components were designed to ASME Section III/ANSI B31.7 requirements.

The influence of the reactor water environment on the cumulative usage factor was evaluated for the following representative components identified in NUREG/CR-6260 for the period of extended operation, using the most recent laboratory data and methods:

- Reactor vessel shell and lower head.

- Reactor vessel inlet and outlet nozzles.
- Surge line.
- Charging System nozzle.
- Safety Injection System nozzle.
- Shutdown Cooling line.

These six fatigue-sensitive locations have been evaluated using the methods identified in NUREG/CR-6583 (Reference 15.3-10), and NUREG/CR-5704 (Reference 15.3-11).

Utilizing Millstone Unit 2 cyclic and transient information, all six fatigue sensitive component locations were determined to be acceptable for the period of extended operation.

Commitments

The following actions will be implemented prior to the period of extended operation:

- Millstone follows industry efforts that provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will implement the appropriate recommendations resulting from this guidance. This action completes the requirements for Commitment Item 28 in Table 15.6-1.

15.3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Electrical Equipment Qualification (EEQ) program is an integral part of the design, construction and operation of nuclear power generating stations. A description of this program is provided in Section 15.4.1, Electrical Equipment Qualification.

10 CFR Part 50 requires that certain categories of systems, structures and components be designed to accommodate the effects of both normal and accident environmental conditions, and that design control measures be employed to ensure the adequacy of these designs. Specific requirements pertaining to the environmental qualification of these categories of electrical equipment are embodied within 10 CFR 50.49 (Reference 15.3-12). The categories include safety-related (Class 1E) electrical equipment, non-safety-related electrical equipment whose failure could prevent the satisfactory accomplishment of a safety function by safety-related equipment, and certain post-accident monitoring equipment. As required by 10 CFR 50.49, electrical equipment not qualified for the current license term is to be refurbished, replaced or have its qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical equipment that specify a qualification of 40 years or greater are considered to represent a time-limited aging analysis. Guidance relating to the methods and procedures for implementing the requirements of 10 CFR 50.49 is contained within Regulatory

Guide 1.89 (Reference 15.3-13). Further guidance for post-accident monitoring equipment is contained within Regulatory Guide 1.97 (Reference 15.3-14).

Environmental qualification of electrical equipment will be adequately managed for the period of extended operation.

15.3.4 CONCRETE CONTAINMENT TENDON PRESTRESS

The Millstone Unit 2 Containment consists of a pre-stressed, reinforced concrete cylinder and dome, and a flat, reinforced concrete mat foundation supported on unweathered bedrock. The cylindrical portion of the Containment is prestressed by a post-tensioning system composed of horizontal and vertical tendons, with the horizontal tendons placed in three 240 degree systems that use three buttresses as support for the anchorages. The dome has a three-way post tensioning system. Prestress on the containment tendons is expected to decrease over the life of the unit as a result of such factors as elastic deformation, creep and shrinkage of concrete, anchorage seating losses, tendon wire friction, stress relaxation and corrosion.

Containment tendon surveillance results has been performed and indicate acceptable losses in Containment tendon prestress have been projected through the period of extended operation. Inspection results, Table 15.3-1, were used to project acceptable dome, vertical and horizontal forces through the current and extended period of operation. Prestress values are projected to remain above the minimum value of 1308 kips (7.03 kips per wire) for each of the tendon groups. Since containment tendon examinations are performed on a 5 year interval, force projections will be refined and updated as necessary following each examination.

The evaluation of containment tendon examination and surveillance test results involves the use of time-limited assumptions such as corrosion rates, losses of tendon prestress, and changes in material properties. Regression analysis incorporating the most recent 25 years.

15.3.5 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

15.3.5.1 Containment Liner Plate

Millstone Unit 2 has a prestressed, post-tensioned concrete Containment surrounded by an enclosure building. A welded carbon steel liner plate is attached to the inside surface of the concrete. Both the liner plate and the penetration sleeves are designed to serve as the primary Containment leakage barrier. Components of the liner plate include penetration sleeves, access openings, piping penetrations, and electrical penetrations.

Evaluations of the Containment liner plate involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses. Acceptable Containment liner plate integrity has been projected to the end of the period of extended operation.

15.3.5.2 Containment Penetrations

All Millstone Unit 2 Containment penetrations are pressure resistant, leak-tight, welded assemblies, fabricated, installed, inspected, and tested in accordance with ASME Nuclear Vessel Code, Section III (Reference 15.3-1) and ANSI Nuclear Piping Code B31.7 (Reference 15.3-2).

Evaluations of Containment liner plate components involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses.

Acceptable Containment penetration integrity has been projected to the period of extended operation.

15.3.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

15.3.6.1 Crane Load Cycle Limit

The containment polar crane, spent fuel crane, and monorails are examples of the types of cranes within the scope of license renewal. These cranes meet the guidance contained in NUREG-0612.

The evaluation of crane loads represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of a time-limited assumption, load cycles. The most frequently used crane is the spent fuel crane. Considering all uses, the spent fuel crane is expected to conservatively experience a total number of load cycles over a 60 year period, that is well below the number of cycles allowed in Crane Manufacturers Association of America, Inc. Specification Number 70.

Acceptable crane load cycles have been projected to the end of the period of extended operation.

15.3.6.2 Reactor Coolant Pump Flywheel

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of primary coolant flow to the core in the event that pump power is lost. During normal operation, the reactor coolant pump flywheels develop sufficient kinetic energy to produce high energy missiles in the event of failure. Conditions that may result in overspeed of the pump increase both the potential for failure and the kinetic energy of the flywheel.

An evaluation was performed of the likelihood of flywheel failure over a 60 year period of operation and a justification was developed for relaxation of Regulatory Guide 1.14, Revision 1, Regulatory Position C.4.b(1), requirements to those identified in Regulatory Position C.4.b(2) (Reference 15.3-15). Using this evaluation, the NRC issued Amendment Number 264 to the unit Technical Specifications, consistent with Regulatory Guide 1.14, Revision 1, Regulatory Position C.4.b(2), to allow the examination of each reactor coolant pump flywheel at least once every 10 years, coinciding with the ASME Section XI inservice inspection program schedule.

The evaluation of reactor coolant pump flywheels represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as thermal cycles and crack growth rates. This evaluation, which indicates a low likelihood of flywheel fatigue failure over a 60 year period, along with implementation of the Inservice Inspection Program: Systems, Components and Supports, provides reasonable assurance that flywheel cracking will be adequately managed for the period of extended operation.

Reactor coolant pump flywheel fatigue cracking will be adequately managed for the period of extended operation.

15.3.6.3 Reactor Coolant Pump Code Case N-481

ASME Boiler and Pressure Vessel Code, Section XI, specifies that a volumetric inspection of the reactor coolant pump casing welds and a visual inspection of pump casing internal surfaces be performed on a reactor coolant pump within each ten year inspection period. These 10 year volumetric inspections are significant for a number of reasons, including; the reactor coolant pumps are welded to the piping, and the pumps must be disassembled in order to gain access to the inside surface of the cast austenitic stainless steel casings. In recognition of these difficulties, ASME Code Case N-481, “Alternative Examination Requirements for Cast Austenitic Pump Casings”, was developed to allow for the replacement of volumetric examinations with a fracture mechanics-based evaluation, supplemented by specific visual inspections.

The evaluation of reactor coolant pump casings represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as thermal cycles and crack growth rates. This evaluation, which indicates a low likelihood of casing fatigue failure over a 60 year period, along with implementation of the Inservice Inspection Program: Systems, Components and Supports, provides reasonable assurance that cracking of cast stainless steel reactor coolant pump casing welds will be adequately managed throughout the period of extended operation.

15.3.6.4 Leak-Before-Break

The Leak-Before-Break (LBB) analyses were evaluated as time-limited aging analyses (TLAAs) to determine that the analyses remain valid for the period of extended operation. The systems and components that have been analyzed for LBB include the reactor coolant loop piping (hot leg, cold leg, and crossover piping), the pressurizer surge line, and portions of the safety injection and shutdown cooling systems.

The LBB analyses were determined to remain valid for the period of extended operation by evaluating their time-based inputs. Thermal aging of cast austenitic stainless steel (CASS) materials and fatigue crack growth calculations were determined to be time-based inputs as defined in 10 CFR 54.3 and required evaluation for the period of extended operation.

The metal fatigue TLAA evaluations described in FSAR Section 15.3.2.1 conclude that design basis limits are not exceeded for ASME Class 1 components (which envelopes the components evaluated for LBB) through the period of extended operation.

Thermal aging of CASS materials for components that have been evaluated for LBB has been evaluated as a TLAA since long term exposure of CASS materials to reactor coolant system operating temperatures results in an increase in material hardness while its ductility, impact strength and fracture toughness decrease. Fracture toughness represents one of the more important design inputs in a LBB evaluation. The degree of reduction in CASS fracture toughness is dependent on the time of thermal exposure. However, the change in material properties due to thermal aging reaches a saturation value, after which material property changes resulting from additional thermal exposure are not significant. The evaluation of the thermal aging of CASS material for the LBB evaluations consisted of a review to determine whether the fracture toughness value used in the analyses was conservative relative to the fully aged value for fracture toughness for the CASS components. The review concluded that the analysis values were either equal to or lower than the worst-case saturation (fully aged) values for fracture toughness in all cases. Therefore, since the CASS material property values used in current design basis LBB evaluations represent fully aged (saturation) values, and since these values would not change with further exposure time, the LBB evaluations remain valid for the thermal aging of CASS materials throughout the period of extended operation.

The LBB analyses have been projected to remain valid through the end of the period of extended operation.

15.3.7 REFERENCES FOR SECTION 15.3

15.3-1 ASME Section III, “Rules for Construction of Nuclear Vessels”, ASME Boiler and Vessel Pressure Code, American Society of Mechanical Engineers, 1971.

15.3-2 ANSI Standard B31.7, Nuclear Power Piping Code, American Society of Mechanical Engineers.

15.3-3 Letter from E. J. Mroczka to NRC, Response to NRC Bulletin No. 88-08, Thermal Stresses in Piping Connected to Reactor Coolant System, September 20, 1988.

15.3-4 Letter from G.S. Vissing to E. J. Mroczka, NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems (TAC 69652), September 30, 1991.

15.3-5 Letter from E. J. Mroczka to NRC, NRC Bulletin Number 88-11, Pressurizer Surge Line Thermal Stratification, February 28, 1989.

15.3-6 Letter from J. F. Opeka to NRC, Response to NRC Staff Request for Additional Information on Pressurizer Surge Line Thermal Stratification Evaluation, September 30, 1993.

15.3-7 Letter from G. S. Vissing to J. F. Opeka, Safety Evaluation for Combustion Engineering Owners Group Report CEN-387-P, Revision 1 - Pressurizer Surge Line Thermal Stratification Evaluation (Bulletin 88-11)(TAC No. M72144), July 6, 1993.

15.3-8 ANSI B31.1, Power Piping Code, American Society of Mechanical Engineers, 1967.

15.3-9 NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, U.S. Nuclear Regulatory Commission.

15.3-10 NUREG/CR-6583, Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels, U.S. Nuclear Regulatory Commission.

15.3-11 NUREG/CR-5704, Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steel, U.S. Nuclear Regulatory Commission.

15.3-12 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.3-13 Regulatory Guide 1.89, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.3-14 Regulatory Guide 1.97, Instrumentation of Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, U. S. Nuclear Regulatory Commission.

15.3-15 SIR-94-080-A, Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements, Structural Integrity Associates, Inc., September 1997.

TABLE 15.3-1 CONTAINMENT TENDON PRESTRESS

Inspection Year	Dome Tendon Projected (kips)	Dome Minimum Value (kips)	Vertical Tendon Projected (kips)	Vertical Minimum Value (kips)	Horizontal Tendon Projected (kips)	Horizontal Minimum Value (kips)
40	1453	1343	1521	1339	1467	1325
60	1435	1343	1509	1339	1449	1325

15.4 TLAA SUPPORT PROGRAMS

15.4.1 ELECTRICAL EQUIPMENT QUALIFICATION

The Electrical Equipment Qualification program corresponds to the Time-Limited Aging Analysis (TLAA) support program described in NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.” The program applies to certain electrical components that are important to safety and could be exposed to post-accident environmental conditions, as defined in 10 CFR 50.49. The EEQ program ensures the continued qualification of this equipment during and following design basis accidents. The program determines the necessity for, and frequency of, component replacement or refurbishment in order to maintain the qualification of the equipment. Performance of preventive maintenance and surveillance activities, and monitoring of normal ambient conditions, ensure that components remain within the bounds of their original qualification and provide a basis for extending qualified life through re-analysis.

The acceptance criterion is that the equipment remains within the bounds of its qualified life such that after maximum normal service conditions, the equipment retains sufficient capacity to perform its required safety function during design basis accident conditions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.4.2 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary program mitigates fatigue cracking caused by cyclic strains in metal components of the reactor coolant pressure boundary. This is accomplished by monitoring and tracking the number of critical thermal and pressure transients for selected Reactor Coolant System components to ensure that the number of design transient cycles is not exceeded during the plant operating life.

The acceptance criterion is the fatigue usage factors bounded by the design usage factors. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.5 EXEMPTIONS

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analyses as defined in 10 CFR 54.3 have been identified.

15.6 LICENSE RENEWAL COMMITMENTS

Table 15.6-1, License Renewal Commitments, provides a listing of the license renewal commitments.

15.6.1 REFERENCES FOR SECTION 15.6

15.6-1 Letter from Leslie N. Hartz to NRC, Millstone Power Station Units 2 and 3, Response to Request for Additional Information License Renewal Applications, August 13, 2004 (Serial Number.: 04-398).

15.6-2 Letter from Daniel G. Stoddard to NRC, Dominion Nuclear Connecticut Inc, Millstone Power Station Unit 2, Information in Support of License Renewal Commitment #13, Program Description for Reactor Vessel Internals Inspections, July 31, 2013 (Serial Number 13-398).

15.6-3 Letter from Daniel G. Stoddard to NRC, Dominion Nuclear Connecticut Inc, Millstone Power Station Unit 2, Information in Support of License Renewal Commitment #14, Program Description for Alloy 600 Aging Management Program, July 31, 2013 (Serial Number 13-399).

15.6-4 Letter S/N 05-307, Millstone Power Station Unit 2, Proposed Revision to Technical Specifications (LBDCR 05-MP2-003) Reactor Coolant System Heatup/Cooldown Limits, July 14, 2005.

15.6-5 Letter S/N 06-420, U.S. Nuclear Regulatory Commission, Millstone Power Station, Unit 2, - Issuance of Amendment Re: Revision to Technical Specifications Pertaining to the Reactor Coolant System Heatup and Cooldown Limits (TAC No. MC7593) May 3, 2006.

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS

Item	Commitment	Source	Schedule [a]
1	The existing inspection program will be modified to include those battery racks that require monitoring for license renewal, but are not already included in the program.	Battery Rack Inspections	Complete
2	Implementing procedures will be modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks.	Battery Rack Inspections	Complete
3	A baseline inspection of the in-scope buried piping located in a damp soil environment will be performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Complete
4	The maintenance and work control procedures will be revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures will include the inspection for the loss of material due to selective leaching which will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Complete
5	The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be established.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
6	Fuse holders meeting the requirements will be evaluated prior to the period of extended operation for possible aging effects requiring management. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The program (if needed for fuse holders) will consider the aging stressors for the metallic clips.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Complete
7	Procedures will be developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	"Prior to Period of Extended Operation." Complete. Subsequent testing will not exceed a 10 year frequency.
8	A baseline visual inspection will be performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation.	Fire Protection Program	Complete
9	Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years has been included in Fire Protection Program.	Fire Protection Program	Prior to The Sprinkler Heads Achieving 50 Years Of Service Life Not to Exceed a 10 Year Interval Thereafter
10	The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns will be enhanced to provide expectations that identify the requirements for the inspection of aging effects.	General Condition Monitoring	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
11	In-scope cable found to be submerged will be subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation will also address the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test will use a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables, or other appropriate testing. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.
12	The Infrequently Accessed Areas Inspection Program will be established.	Infrequently Accessed Areas Inspection Program	Complete
13	Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), and stress corrosion cracking (PWSCC and IASCC) and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, will be submitted to the NRC for approval.	ISI Program: Reactor Vessel Internals (Reference 15.6-2)	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
14	Millstone will follow the industry efforts investigating the aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description will be submitted at least two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 54.21(a)(3).	Alloy 600 Management Program	At Least Two Years Prior to the Period of Extended Operation Complete
15	The existing inspection program will be modified to include those lifting devices that require monitoring for license renewal, but are not already included in the program.	Inspection Activities: Load Handling Cranes and Devices	Complete
16	Implementing procedures and documentation will be modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal in Commitment 15.	Inspection Activities: Load Handling Cranes and Devices	Complete
17	The implementing procedures will be modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program.	Structures Monitoring Program	Complete
18	The Structures Monitoring Program and implementing procedures will be modified to include all in-scope structures.	Structures Monitoring Program	Complete
19	Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade.	Structures Monitoring Program	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
20	The Structures Monitoring Program and implementing procedures will be modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged.	Structures Monitoring Program	Complete
21	The maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason.	Structures Monitoring Program	Complete
22	Appropriate inspections of sealants and caulkings used for moisture intrusion prevention in and around aboveground tanks will be performed.	Tank Inspection Program	Complete
23	Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, will be performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities.	Tank Inspection Program	Applicable inspections have been performed. Subsequent inspections will be performed on a frequency consistent with scheduled tank internal inspection activities.
24	The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and will be included on the respective Tank Inspection Program inspection plan.	Tank Inspection Program	Complete
25	Changes will be made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities.	Work Control Process	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
26	Dominion actively participates in a comprehensive industry initiative, in response to NRC Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." Dominion is following the industry efforts on this issue and will implement the appropriate recommendations resulting from this guidance prior to the period of extended operation.	Environmental Report - SAMA Analysis	Complete
27	For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally-embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3.	Inservice Inspection Program: Systems, Components and Supports	Complete
28	Millstone will follow industry efforts that will provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will also implement the appropriate recommendations resulting from this guidance.	Environmentally Assisted Fatigue	Complete
29	A baseline visual inspection will be performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one: • Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger, • Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and • Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.	Closed-Cycle Cooling Water System	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
30	Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching will be performed on a representative sample of locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation.	Work Control Process	Complete
31	A review of the Work Control Process inspection opportunities for each material and environment group, supplemental to the initial review conducted during the development of the LRA, will be performed. Baseline inspections will be performed for the material and environment combinations that have not been inspected as part of the Work Control Process.	Work Control Process	Complete
32	Calibration results for cables tested in situ will be reviewed to detect severe aging degradation of the cable insulation. The initial review will be completed prior to entering the period of extended operation and will include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Prior to period of extended operation complete. Subsequent reviews will not exceed a 10 year frequency.
33	The in scope cables in Unit 3 duct lines number 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and number 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C and 35D) will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.
34	In addition to the testing specified in Commitment 33, a representative sample of in-scope medium voltage cables will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Prior to Period of Extended Operation Not to Exceed a 10 Year Frequency Thereafter

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
35	Millstone Unit 2 will complete the SAMA evaluation of the capability to flash the Diesel Generator field in the event of extended loss of DC power with a loss of offsite power. If this SAMA is cost beneficial (i.e., can be accomplished without a hardware modification), a Severe Accident Management Guideline (SAMG) addressing this mitigation strategy will be developed.	Severe Accident Mitigation Alternatives (SAMA) Analysis (Reference 15.6-1)	Complete
36	Dominion will replace the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC.	Inservice Inspection Program: Systems, Components and Supports	Complete
37	Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, will be submitted to the NRC for review at least two years prior to the period of extended operation.	Reactor Vessel Neutron Embrittlement	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.

a. The Period of Extended Operation is the period of 20 years beyond the expiration date of the unit's previous 40 year operating license.