APPENDIX A FINAL SAFETY ANALYSIS REPORT SUPPLEMENT
TABLE OF CONTENTS

A.1 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 A.1.1 contains a listing of the aging management programs that correspond to NUREG-1801 Chapter XI programs, including the status of the program at the time the License Renewal Application was submitted.
- Section A.1.2 contains a listing of the plant-specific aging management programs, including the status of the program at the time the License Renewal Application was submitted.
- Section A.1.3 contains a listing of aging management programs that correspond to NUREG-1801 Chapter X programs associated with Time-Limited Aging Analyses, including the status of the program at the time the License Renewal Application was submitted.
- Section A.1.4 contains a listing of the Time-Limited Aging Analyses.
- Section A.1.5 contains a discussion of the Quality Assurance Program and Administrative Controls.
- Section A.2 contains a summarized description of the aging management programs.
- Section A.2.1 contains a summarized description of the NUREG-1801 Chapter XI programs for managing the effects of aging.
- Section A.2.2 contains a summarized description of the plant-specific programs for managing the effects of aging.
- Section A.3 contains a summarized description of the NUREG-1801 Chapter X programs that support the TLAAs.
- Section A.4 contains a summarized description of the Time-Limited Aging Analyses (TLAAs) applicable to the period of extended operation.
- Section A.5 contains the License Renewal Commitment List.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that systems, structures, and 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 current operating license expiration date.

A.1.1 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

The NUREG-1801 Chapter XI Aging Management Programs (AMPs) are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 or require enhancements.

The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in the Appendix A.5 License Renewal Commitment List.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section A.2.1.1) [Existing]
2. Water Chemistry (Section A.2.1.2) [Existing – Requires Enhancement]
3. Reactor Head Closure Studs (Section A.2.1.3) [Existing]
4. Boric Acid Corrosion (Section A.2.1.4) [Existing]
5. Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (Section A.2.1.5) [Existing]
6. Flow-Accelerated Corrosion (Section A.2.1.6) [Existing]
7. Bolting Integrity (Section A.2.1.7) [Existing]
8. Steam Generator Tube Integrity (Section A.2.1.8) [Existing]
9. Open-Cycle Cooling Water System (Section A.2.1.9) [Existing – Requires Enhancement]
10. Closed-Cycle Cooling Water System (Section A.2.1.10) [Existing – Requires Enhancement]
11. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section A.2.1.11) [Existing – Requires Enhancement]
12. Compressed Air Monitoring (Section A.2.1.12) [Existing – Requires Enhancement]
13. Fire Protection (Section A.2.1.13) [Existing – Requires Enhancement]
14. Fire Water System (Section A.2.1.14) [Existing – Requires Enhancement]
15. Aboveground Steel Tanks (Section A.2.1.15) [Existing – Requires Enhancement]
16. Fuel Oil Chemistry (Section A.2.1.16) [Existing – Requires Enhancement]
17. Reactor Vessel Surveillance (Section A.2.1.17) [Existing – Requires Enhancement]

18. One-Time Inspection (Section A.2.1.18) [New]
19. Selective Leaching of Materials (Section A.2.1.19) [New]
20. Buried Piping and Tanks Inspection (Section A.2.1.20) [Existing – Requires Enhancement]
21. External Surfaces Monitoring (Section A.2.1.21) [New]
22. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (Section A.2.1.22) [New]
23. Lubricating Oil Analysis (Section A.2.1.23) [Existing]
24. ASME Section XI, Subsection IWE (Section A.2.1.24) [Existing]
25. ASME Section XI, Subsection IWL (Section A.2.1.25) [Existing]
26. ASME Section XI, Subsection IWF (Section A.2.1.26) [Existing]
27. 10 CFR Part 50, Appendix J (Section A.2.1.27) [Existing]
28. Structures Monitoring Program (Section A.2.1.28) [Existing – Requires Enhancement]
29. Protective Coating Monitoring and Maintenance Program (Section A.2.1.29) [Existing]
30. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.30) [New]
31. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section A.2.1.31) [Existing – Requires Enhancement]
32. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.32) [New]
33. Metal Enclosed Bus (Section A.2.1.33) [Existing – Requires Enhancement]
34. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirement (Section A.2.1.34) [New]

A.1.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

The plant-specific programs are described in the following sections. The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in the Appendix A.5 License Renewal Commitment List.

1. Nickel Alloy Aging Management Program (Section A.2.2.1) [Existing]

A.1.3 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

The NUREG-1801 Chapter X Aging Management Programs associated with Time-Limited Aging Analyses are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 Chapter X or require enhancements. The following list reflects the status of these programs at the time of the License Renewal Application (LRA) submittal. Commitments for program additions and enhancements are identified in the Appendix A.5 License Renewal Commitment List.

1. Metal Fatigue of Reactor Coolant Pressure Boundary (Section A.3.1.1) [Existing – Requires Enhancement]

2. Concrete Containment Tendon Prestress (Section A.3.1.2) [Existing]

3. Environmental Qualification (EQ) of Electrical Components (Section A.3.1.3) [Existing]

A.1.4 TIME-LIMITED AGING ANALYSIS SUMMARIES

Summaries of the Time-Limited Aging Analyses applicable to the period of extended operation are included in the following sections.

1. Neutron Embrittlement of the Reactor Vessel and Internals (Section A.4.2)

2. Metal Fatigue of Piping and Components (Section A.4.3)

3. Leak-Before-Break Analysis of Primary System Piping (Section A.4.4)

4. Fuel Transfer Tube Bellows Design Cycles (Section A.4.5)

5. Crane Load Cycle Limits (Section A.4.6)

6. Loss of Prestress in Concrete Containment Tendons (Section A.4.7)

7. Environmental Qualification EQ of Electrical Components (Section A.4.8)

A.1.5 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2, “Quality Assurance For Aging Management Programs (Branch Technical Position IQMB-1)” of NUREG-1800. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and these elements are applicable to the safety-related and non-safety related systems, structures, and components (SSCs) that are subject to Aging Management Review (AMR). In many cases, existing activities were found adequate for managing aging effects during the period of extended operation.

A.2 AGING MANAGEMENT PROGRAMS

A.2.1 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

This section provides summaries of the NUREG-1801 programs credited for managing the effects of aging.

A.2.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is an existing program that consists of periodic volumetric and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions. The inspections will be implemented in accordance with 10 CFR 50.55(a). These activities include inspections, and monitoring and trending of results to confirm that aging effects are managed.

A.2.1.2 WATER CHEMISTRY

The TMI-1 Water Chemistry aging management program is an existing program that provides activities for monitoring and controlling the chemical environments of the TMI-1 primary cycle and secondary cycle systems such that aging effects of system components are minimized. Aging effects include cracking, denting, loss of material, reduction of heat transfer, and reduction of neutron-absorbing capacity. The primary cycle scope of this program consists of the reactor coolant system and related auxiliary systems containing reactor coolant (borated treated water), including the primary side of the steam generators. The secondary cycle portion of the program consists of the various secondary side systems and the secondary side of the steam generators. Major component types include reactor vessel, reactor internals, heat exchangers, pumps casing, boiler casings, filter housings, tanks, valve bodies, piping, and piping components. The Water Chemistry aging management program is consistent with EPRI 1002884, Pressurized Water Reactor Primary Chemistry Guidelines, Revision 5 and Plant Technical Specification limits for fluorides, chlorides, and dissolved oxygen. The Water Chemistry program will be enhanced to incorporate the continuous monitoring of sodium in steam generator blowdown, making it consistent with EPRI 1008224, Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 6. This enhancement will be implemented prior to the period of extended operation.

A.2.1.3 REACTOR HEAD CLOSURE STUDS

The Reactor Head Closure Studs program is an existing program that provides for condition monitoring and preventive activities to manage stud cracking and loss of material. The program is implemented through station procedures based on the examination and inspection requirements specified in ASME Section XI, Table IWB-2500-1 and preventive measures described in NRC Regulatory Guide 1.65, “Materials and Inspection for Reactor Vessel Closure Studs.”

A.2.1.4 BORIC ACID CORROSION

The Boric Acid Corrosion aging management program is an existing program that manages loss of material due to boric acid corrosion. The program includes provisions to identify, inspect, examine and evaluate leakage, and initiate corrective action. The program relies in part on implementation of recommendations of NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Components in PWR plants” and includes visual examinations of Alloy 600 components for stress corrosion cracking due to boric acid leakage.

A.2.1.5 NICKEL-ALLOY PENETRATION NOZZLES WELDED TO THE UPPER REACTOR VESSEL CLOSURE HEADS OF PRESSURIZED WATER REACTORS

The TMI-1 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors aging management program is an existing program credited for managing primary water stress corrosion cracking (PWSCC) of upper vessel head penetration (VHP) nozzles. The program for Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (Upper Head Nickel Alloy AMP) was developed by TMI-1 to respond to NRC Order EA-03-009. The Upper Head Nickel Alloy AMP manages cracking due to PWSCC in nickel-alloy vessel head penetration nozzles and includes the reactor vessel closure head, upper vessel head penetration nozzles and associated welds.

A.2.1.6 FLOW-ACCELERATED CORROSION

The Flow-Accelerated Corrosion (FAC) aging management program is an existing program based on EPRI guidelines in NSAC-202L, “Recommendations for an Effective Flow Accelerated Corrosion Program.” The program provides for predicting, detecting, and monitoring wall thinning in piping, fittings, valve bodies, and feedwater heaters due to FAC. Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to FAC are used to predict the amount of wall thinning in pipes, fittings, and feedwater heater shells. Program activities include analyses to determine critical locations, baseline inspections to determine the extent of thinning at these critical locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

A.2.1.7 BOLTING INTEGRITY

The Bolting Integrity aging management program is an existing program that incorporates industry recommendations of EPRI NP 5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” and includes periodic visual inspections of closure bolting for loss of material due to general, pitting and crevice corrosion, microbiologically influenced corrosion and loss of preload due to thermal effects, gasket creep, and self-loosening. Inspection of Class 1, 2, and 3 components is conducted in accordance with ASME Section XI. The requirements of ASME Section XI will be implemented in accordance with 10 CFR 50.55(a). Program activities address the guidance contained in EPRI TR 104213, “Bolted Joint Maintenance and Applications Guide”. Non-ASME Class 1, 2 and 3 bolted joint inspections rely on detection of visible leakage during maintenance or routine observation.

A.2.1.8 STEAM GENERATOR TUBE INTEGRITY

The Steam Generator Tube Integrity program is an existing program that establishes the operation, maintenance, testing, inspection and repair of the steam generators to ensure that Technical Specification surveillance requirements, ASME Code requirements and the Maintenance Rule performance criteria are met. The program identifies, maintains and protects the steam generator design and licensing bases and implements NEI 97-06. NEI 97-06 establishes a framework for prevention, inspection, evaluation, repair and leakage monitoring measures.

TMI-1 will replace the original Once-Through Steam Generators (OTSGs) with enhanced OTSGs prior to the period of extended operation. This decision was made based on industry and TMI-1 experience with tube degradation. The new OTSGs have improved design features including Alloy 690 tubes. The new OTSGs will have a design life of 40 years, which along with the Steam Generator Tube Integrity program will be effective in assuring that the intended functions will be maintained consistent with the CLB for the period of extended operation.

A.2.1.9 OPEN-CYCLE COOLING WATER SYSTEM

The TMI-1 Open-Cycle Cooling Water System aging management program is an existing program. The GL 89-13 activities provide for management of aging effects in raw water cooling systems through tests and inspections per the guidelines of NRC Generic Letter 89-13. System and component testing, visual inspections, NDE (RT, UT, and/or ECT-Eddy Current Testing), and chemical treatment are conducted to ensure that aging effects are managed such that system and component intended functions and integrity are maintained.

The TMI-1 Open-Cycle Cooling Water System (OCCWS) aging management program primarily consists of station GL 89-13 activities that include chemical and biocide injection, system testing, periodic inspections and NDE. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the OCCW system or structures and components serviced by the OCCW system. Other activities include station maintenance inspections, component preventive maintenance (PM), plant surveillance testing, ISI, and inspections. These activities provide for management of loss of material (without credit for protective coatings) and buildup of deposit (including fouling from biological, corrosion product, and external sources) aging effects where applicable in system components exposed to a raw water environment.

The program will be enhanced by adding a new river water chemical treatment system to treat the river water systems for biofouling. This enhancement will be implemented prior to the period of extended operation.

A.2.1.10 CLOSED-CYCLE COOLING WATER SYSTEM

The Closed-Cycle Cooling Water System aging management program is an existing program that manages aging of piping, piping components, piping elements and heat exchangers that are included in the scope of license renewal for loss of material and reduction of heat transfer and are exposed to a closed cooling water environment at TMI-1. The Closed-Cycle Cooling Water System aging management program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive examinations (NDE) based on industry-recognized guidelines of EPRI 1007820 for closed-cycle cooling water systems. Station maintenance inspections and NDE provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments.

The program will be enhanced to include a one-time inspection of selected components in stagnant flow areas to confirm the absence of aging effects resulting from exposure to closed cycle cooling water. Also, a one-time inspection of selected CCCW chemical mix tanks and associated piping components will be performed to verify corrosion has not occurred on the interior surfaces of the tanks and associated piping components. These enhancements will be implemented prior to entering the period of extended operation.

A.2.1.11 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program is an existing program that is credited for managing aging effects of cranes and hoists in the scope of license renewal. The program was developed based on guidance contained in ASME/ANSI B30.2, and B30.16 and relies on visual inspection to detect loss of material due to corrosion of cranes and hoists components (including bridge, trolley, rails, girders, and other lifting devices). Structural bolting is monitored for loss of preload by inspecting for loose or missing bolts, or nuts. Inspection frequency is annually for cranes and hoists that are accessible during plant operation and every 2 years for cranes and hoists that are only accessible during refueling outages.

Prior to the period of extended operation, the program will be enhanced to include visual inspection of rails in the rail system for loss of material due to wear, and visual inspection of structural bolting for loss of material. Acceptance criteria will be enhanced to require that significant loss of material due corrosion or wear be evaluated or corrected to ensure the intended function of the crane or hoist is not impacted.

A.2.1.12 COMPRESSED AIR MONITORING

The Compressed Air Monitoring aging management program is an existing program that manages the internal surface aging effects of loss of material due to general, pitting and crevice corrosion, and the reduction of heat transfer due to fouling for piping and components in a compressed air system. The TMI-1 aging management activities consist of preventive and condition-monitoring measures to manage the aging effects.

The Compressed Air Monitoring program will be enhanced to include air quality testing for dew point, particulates, lubricant content, and contaminants to ensure that the contamination standards of ANSI/ISA-S7.0.01-1996, paragraph 5 are met.

In addition the Compressed Air Monitoring program will be enhanced to include air quality sampling on a representative sampling of headers on a yearly basis in accordance with the guidelines of ASME OM-S/G-1998, Part 17 and EPRI TR-108147.

Enhancements will be implemented prior to entering the period of extended operation.

A.2.1.13 FIRE PROTECTION

The Fire Protection aging management program is an existing program that includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The program includes surveillance tests of fuel oil systems for the diesel-driven fire pumps to ensure that the fuel supply lines can perform intended functions. The program also includes visual inspections and periodic operability tests of halon and carbon dioxide fire suppression systems based on NFPA codes.

The Fire Protection aging management program will be enhanced to include additional inspection criteria for degradation of fire barrier walls, ceilings, and floors, and specific fuel supply line inspection criteria for diesel-driven fire pumps during tests.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.14 FIRE WATER SYSTEM

The Fire Water System aging management program is an existing program that provides for system pressure monitoring, fire system header flushing and flow testing, pump performance testing, hydrant flushing, and visual inspection activities. System flow tests measure hydraulic resistance and compare results with previous testing, as a means of evaluating the internal piping conditions. Monitoring system piping flow characteristics ensures that signs of internal piping degradation from significant corrosion or fouling would be detected in a timely manner. Pump performance tests, hydrant flushing and system inspections are performed in accordance with applicable NFPA standards.

The program will be enhanced to include sprinkler head testing in accordance with NFPA 25, “Inspection, Testing and Maintenance of Water-Based Fire Protection Systems.” Samples will be submitted to a testing laboratory prior to being in service 50 years. This testing will be repeated at intervals not exceeding 10 years.

Prior to the period of extended operation, the program will be enhanced to include periodic non-intrusive wall thickness measurements of selected portions of the fire water system at an interval not to exceed every 10 years.

A.2.1.15 ABOVEGROUND STEEL TANKS

The Aboveground Steel Tanks aging management program is an existing program that will manage corrosion of outdoor carbon steel tanks. Paint is a corrosion preventive measure, and periodic visual inspections will monitor degradation of the paint and any resulting metal degradation of carbon steel tanks. Tanks within the scope of this program are the Condensate Storage Tanks, Fire Service Head Tank (Altitude Tank) and Sodium Hydroxide Tank. The Fire Service Head Tank and Sodium Hydroxide Tank are supported by structural steel and the Condensate Storage Tanks are supported by concrete foundations. Therefore, inspection of the sealant at the tank-foundation interface, and UT inspection of inaccessible tank bottoms apply only to the Condensate Storage Tanks, which are located on concrete pads.

The existing TMI-1 Aboveground Steel Tanks program implementing procedures will be enhanced to include one-time thickness measurements of the bottom of the Condensate Storage Tanks, which are supported on concrete foundations. The measurements will be taken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. Existing implementing procedures will also be enhanced to inspect the sealant at the tank-foundation interface. Enhancements will be implemented prior to the period of extended operation.

A.2.1.16 FUEL OIL CHEMISTRY

The Fuel Oil Chemistry aging management program is an existing program that includes preventive activities to provide assurance that contaminants are maintained at acceptable levels in fuel oil for systems and components within the scope of License Renewal. The fuel oil tanks within the scope of License Renewal are maintained by monitoring and controlling fuel oil contaminants in accordance with the guidelines of the American Society for Testing and Materials (ASTM). Fuel oil sampling and analysis is performed in accordance with approved procedures for new fuel oil and stored fuel oil. Fuel oil tanks are periodically drained of accumulated water and sediment, cleaned, and internally inspected. These activities effectively manage the effects of aging by providing reasonable assurance that potentially harmful contaminants are maintained at low concentrations. The TMI-1 Fuel Oil Chemistry aging management program will be enhanced to include:

- The analysis of new fuel oil for specific or API gravity, kinematic viscosity, and water and sediment prior to filling the fuel oil storage tanks followed by full spectrum analysis within 31 days after the addition of the fuel oil into the fuel oil storage tanks.
- The determination of water and sediment and particulate contamination in accordance with ASTM standards.
- The analysis for bacteria in new and stored fuel oil.
- The addition of biocides, stabilizers, or corrosion inhibitors as determined by fuel oil analysis activities.
- Activities to periodically drain water and sediment from tank bottoms, and, activities to periodically drain, clean, and inspect fuel oil tanks.
- Manual sampling in accordance with ASTM standards and required frequencies.
- The use of ultrasonic techniques for determining tank bottom thicknesses should there be any evidence of loss of material due to general, pitting, crevice, and microbiologically influenced corrosion, and fouling found during visual inspection activities.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.17 REACTOR VESSEL SURVEILLANCE

The TMI-1 Reactor Vessel Surveillance program is an existing program that manages the reduction of fracture toughness of the reactor vessel beltline materials due to neutron embrittlement. The program fulfills the intent and scope of 10 CFR 50, Appendix H. The program evaluates neutron embrittlement by projecting upper-shelf energy (USE) for all reactor materials with projected neutron exposure greater than 10^{17} n/cm^2 (E >1MeV) after 60 years of operation and with the development of pressure-temperature limit curves. Embrittlement information is obtained in accordance with RG 1.99, Rev. 2 chemistry tables.

TMI-1 participates in the Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP). The integrated program is feasible because of the similarity of the design and operating characteristics of the affected plants, as required by 10 CFR Part 50, Appendix H, paragraph II.C. The integrated program provides sufficient material data to meet the ASTM E-185-82 capsule program requirement for monitoring embrittlement. The MIRVSP includes all seven operating B&W 177-fuel assembly (FA) plants and six participating Westinghouse-designed plants having B&W-fabricated reactor vessels.

By letter dated December 20, 2005, the Babcock and Wilcox Owners Group Reactor Vessel Working Group, now a part of the PWROG, submitted TR BAW-1543 (NP), Revision 4, Supplement 6, “Supplement to the Master Integrated Reactor Vessel Surveillance Program,” to the NRC staff for review. Supplement 6 revised neutron fluence values and revised the withdrawal schedules of capsules A2 and A4 from the end of the 17th fuel cycle to the end of the 29th fuel cycle to adjust the planned neutron exposure for limiting Linde 80 heats to correspond with 60 and 80 years of operation. The NRC staff determined that the withdrawal schedules were prepared in accordance with ASTM Standard E 185-82 for each of the subject units and that the withdrawal schedules meet the requirements of 10 CFR 50, Appendix H.

The TMI-1 Reactor Vessel Surveillance program will be enhanced to address maintenance of the TMI-1 cavity dosimetry exchange schedule. The program will also be enhanced to clarify that, if future plant operations exceed the limitations or bounds specified in Regulatory Position 1.3 of RG 1.99, Rev. 2, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. Enhancements will be implemented prior to the period of extended operation.

A.2.1.18 ONE-TIME INSPECTION

The TMI-1 One-Time Inspection aging management program is a new program that will provide reasonable assurance that an aging effect is not occurring, or that the aging effect is occurring slowly enough to not affect a components intended function during the period of extended operation, and therefore will not require additional aging management. The program will be credited for cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, (b) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse than that generally expected, or (c) the characteristics of the aging effect include a long incubation period.

This program will be used for the following:

- To confirm the effectiveness of the Water Chemistry program to manage the loss of material, cracking, and the reduction of heat transfer aging effects for steel, stainless steel, copper alloy, nickel alloy, and aluminum alloy in treated water, steam, and reactor coolant environments.

- To confirm the effectiveness of the Fuel Oil Chemistry program to manage the loss of material aging effect for steel, stainless steel, and copper alloy in a fuel oil environment.
- To confirm the effectiveness of the Lubricating Oil Analysis program to manage the loss of material and the reduction of heat transfer aging effects for steel, stainless steel, copper alloy, and aluminum alloy in a lubricating oil environment.
- To confirm the loss of material aging effect is insignificant for stainless steel and copper alloy in an air/gas – wetted environment.

The inspections will be implemented prior to the period of extended operation to manage the effects of aging for selected components within the scope of license renewal.

Inspection methods will include visual examination or volumetric examination. Acceptance criteria are in accordance with industry guidelines, codes, and standards. The One-Time Inspection program provides for the evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation. Should aging effects be detected, the program triggers actions to characterize the nature and extent of the aging effect and determines what subsequent monitoring is needed to ensure intended functions are maintained during the period of extended operation.

A.2.1.19 SELECTIVE LEACHING OF MATERIALS

The Selective Leaching of Materials aging program is a new program that will include inspections of a representative sample of susceptible components to determine if loss of material due to selective leaching is occurring. One-time inspections will include visual examinations, supplemented by hardness tests, and other examinations, as required. If selective leaching is found, the condition will be evaluated to determine the need to expand inspection scope. This new inspection program will be implemented prior to the period of extended operation.

A.2.1.20 BURIED PIPING AND TANKS INSPECTION

The Buried Piping and Tanks Inspection aging management program is an existing program that manages the external surface aging effects of loss of material for piping and components in a soil (external) environment. The TMI-1 buried component activities consist of preventive and condition-monitoring measures to manage the loss of material due to external corrosion for piping and components in the scope of license renewal that are in a soil (external) environment.

External inspections of buried components will occur opportunistically when they are excavated during maintenance. Excavation and component inspection activities of buried piping and components for cast iron, carbon steel, and concrete-coated carbon steel materials have occurred in the ten years prior to the beginning of the period of extended operation. The program will be enhanced to include at least one opportunistic or focused excavation and inspection of stainless steel piping and components prior to entering the period of extended operation. Upon entering the period of extended operation, a focused inspection of an example of each of the above materials shall be performed within ten years, unless an opportunistic inspection occurs within this ten-year period. The program will also be enhanced to include an internal inspection and UT of the buried Diesel Generator Fuel Storage 30,000 Gal Tank within the ten-year period prior to the period of extended operation, and within ten years of entering the period of extended operation. Program enhancements will be implemented prior to entering the period of extended operation.

A.2.1.21 EXTERNAL SURFACES MONITORING

The External Surfaces Monitoring aging management program is a new program that directs visual inspections that are performed during system walkdowns. The program consists of periodic visual inspection of components such as piping, piping components, ducting, and other components within the scope of license renewal. The program manages aging effects through visual inspection of external surfaces for evidence of hardening and loss of strength and loss of material. Visual inspections may be augmented by physical manipulation to detect hardening and loss of strength of elastomers. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion program. The external surfaces of components that are buried and those of above ground tanks are inspected via the Buried Piping and Tanks Inspection program and the Aboveground Steel Tanks program, respectively. This new aging management program will be implemented prior to the period of extended operation.

A.2.1.22 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new program that manages cracking due to stress corrosion cracking; hardening and loss of strength due to elastomer degradation; loss of material due to general, pitting, crevice, and microbiologically influenced corrosion and fouling; and reduction of heat transfer due to fouling. The program includes provisions for visual inspections of the internal surfaces and volumetric testing of components not managed under any other aging management program. Identified deficiencies are evaluated under the Corrective Action Program. This new aging management program will be implemented prior to the period of extended operation.

A.2.1.23 LUBRICATING OIL ANALYSIS

The TMI-1 Lubricating Oil Analysis aging management program is an existing program that provides oil condition monitoring activities to manage the loss of material and the reduction of heat transfer in piping, piping components, piping elements, heat exchangers, and tanks within the scope of license renewal exposed to a lubricating oil environment. Sampling and condition monitoring activities identify specific wear products, contamination and the physical properties of lubricating oil within operating machinery to ensure that intended functions are maintained.

A.2.1.24 ASME SECTION XI, SUBSECTION IWE

The ASME Section XI, Subsection IWE aging management program is an existing program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program consists of periodic inspection of the Reactor Building liner plate, including its integral attachments, penetration sleeves, pressure retaining bolting, personnel airlock and equipment hatch, seals, gaskets, and moisture barrier, and other pressure retaining components for loss of material (general, pitting, and crevice corrosion), loss of pressure retaining bolting preload, cracking due to cyclic loading, loss of sealing, leakage through containment/deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants).

Examination methods include visual and volumetric testing as required by ASME Section XI, Subsection IWE. Observed conditions that have the potential for impacting an intended function are evaluated for acceptability in accordance with ASME requirements or corrected in accordance with corrective action process.

TMI-1 is committed to replacing the existing steam generators with new Once Through Steam Generators (OTSGs) prior to entering the period of extended operation. Repair/replacement of Reactor Building liner plate, removed for access purposes, will be done in accordance with ASME Section XI, Subsection IWE.

A.2.1.25 ASME SECTION XI, SUBSECTION IWL

The ASME Section XI, Subsection IWL aging management program is an existing program based on ASME Code and complies with the provisions 10 CFR 50.55a. The program requires periodic inspection of the reactor building (containment) reinforced concrete surfaces and unbonded post-tensioning system for degradations. Reinforced concrete surfaces are inspected for loss of material, cracking, increase in porosity and permeability, and loss of bond. A sample of each tendon wire type (vertical, hoop, dome) for the post-tensioning system is tested for loss of prestress. One tendon wire of each type is also examined for loss of material and subject to physical testing to determine yield strength, ultimate tensile strength, and elongation. The end anchorage for the post-tensioning system is inspected for loss of material.

Examination methods include visual inspection and physical testing in accordance with ASME Section XI, Subsection IWL. An acceptance criterion is in accordance with ASME Section XI, Subsection IWL. Observed conditions and tests that have the potential for impacting an intended function are evaluated for acceptability in accordance with ASME requirements or corrected in accordance with the corrective action process.

TMI-1 is committed to replacing the existing steam generators with new Once Through Steam Generators (OTSGs) prior to entering the period of extended operation. Repair/replacement of Reactor Building concrete and prestressing system, removed for access purposes, will be done in accordance with ASME Section XI, Subsection IWL.

A.2.1.26 ASME SECTION XI, SUBSECTION IWF

The ASME Section XI, Subsection IWF aging management program is an existing program that consists of periodic visual examinations of ASME Class 1, 2, and 3 piping and component supports for identification of signs of degradation, and establishment of corrective actions. The inspections are in accordance with ASME Section XI, Subsection IWF as approved in 10 CFR 50.55(a). The program activities are relied upon to detect and confirm that aging effects are managed.

A.2.1.27 10 CFR PART 50, APPENDIX J

The 10 CFR Part 50, Appendix J aging management program is an existing program that monitors leakage rates through the containment pressure boundary, including penetrations, fittings and other access openings, in order to detect age related degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Appendix J program also detects age related degradation in material properties of gaskets, o-rings and packing materials for the containment pressure boundary access points. Consistent with the current licensing basis, the containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J Option B, Regulatory Guide 1.163, “Performance-Based Containment Leak-Testing Program,” NEI 94-01 “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J”, and ANSI/ANS 56.8, “Containment System Leakage Testing Requirements.”

A.2.1.28 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program is an existing program that was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93 01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2 and Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2. The program includes elements of the Masonry Wall Program and the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants aging management program.

The program relies on periodic visual inspections to monitor the condition of structures and structural components, structural bolting, component supports, masonry block walls, water control structures. The inspections are conducted on a frequency not greater than 5 years.

Enhancements to the program include the following:

- Include Service Building, UPS Diesel Building, Mechanical Draft Cooling Tower Structures, Miscellaneous Yard Structures (Foundation for condensate storage tank, borated water storage tank, diesel fuel storage tank, altitude tank, duct banks, and manholes) Penetration seals in the scope of the program.
- Monitor the Intake Canal for Loss of material and loss of form
- Monitor electrical panels, junction boxes, instrument panels, and conduits for loss of material due to corrosion
- Monitor ground water chemistry by periodically sampling, testing, and analysis of ground water to confirm that the environment remains non-aggressive for buried reinforced concrete.
- Monitor reinforced concrete submerged in raw water associated with Intake Screen and Pumphouse, Circulating Water Pump House, Mechanical Draft Cooling Tower Structures, Natural Draft Cooling Tower Basins.
- Monitor vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF, for reduction or loss of isolation function.
- Parameters monitored will be enhanced to include plausible aging mechanisms.
- Monitor concrete structures for a reduction in anchor capacity due to local concrete degradation. This will be accomplished by visual inspection of concrete surfaces around anchors for cracking, and spalling.
- Revise acceptance criteria to provide details specified in ACI 349.3R-96.

The enhancements will be implemented prior to entering the period of extended operation.

A.2.1.29 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM

The Protective Coating Monitoring and Maintenance Program is an existing program that provides for aging management of Service Level I coatings inside the containment. Service Level I coatings are used in areas where corrosion protection may be required and where coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. The Protective Coating Monitoring and Maintenance Program provides for inspections, assessment, and repairs for any condition that adversely affects the ability of Service Level I coatings to function as intended.

A.2.1.30 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new program that will be used to manage aging of non-EQ cables and connections during the period of extended operation. A representative sample of accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for indications of accelerated insulation aging such as embrittlement, discoloration, cracking, or surface contamination. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection. This new program will be implemented prior to the period of extended operation.

A.2.1.31 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program is an existing program that will be enhanced to manage the aging of the cable and connection insulation of the in scope radiation monitoring and nuclear instrumentation circuits in the License Renewal Radiation Monitoring and Nuclear Instrumentation and Incore Monitoring Systems. The in scope radiation monitoring and nuclear instrumentation circuits are sensitive instrumentation circuits with low-level signals and are located in areas where the cables and connections could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Calibration testing and system performance monitoring are currently being performed for in scope radiation monitoring circuits. Direct cable testing will be performed as an enhancement to ensure that the cable and connection insulation resistance is adequate for the in scope nuclear instrumentation circuits to perform their intended functions. The enhanced program will be implemented prior to the period of extended operation.

A.2.1.32 INACCESSIBLE MEDIUM VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible medium voltage cables. These cables may at times be exposed to significant moisture simultaneously with significant voltage. The TMI-1 cables in the scope of this aging management program will be tested using a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. The cables will be tested at least once every 10 years. Manholes associated with the cables included in this aging management program will be inspected for water collection at least twice a year, in accordance with existing practices, and drained as required. This new aging management program will be implemented prior to the period of extended operation.

A.2.1.33 METAL ENCLOSED BUS

The Metal Enclosed Bus aging management program is an existing program that will be enhanced to manage the aging of metal enclosed busses at TMI-1. The Metal Enclosed Bus aging management program is an existing program that will be enhanced to manage the aging of metal enclosed busses at TMI-1. A sample of accessible bolted connections is currently checked for loose connections via Thermography. A sample of metal enclosed bus internals is currently visually inspected.

The Metal Enclosed Bus aging management program will be enhanced to specify the following inspection criteria.

1. Internal portion of the metal enclosed bus will be visually inspected for cracks, corrosion, foreign debris, excessive dust build-up and evidence of moisture intrusion.

2. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation.

3. The internal bus supports will be visually inspected for structural integrity and signs of cracks.

The Metal Enclosed Bus aging management program will be enhanced to perform internal visual inspections on the 480V Metal Enclosed Bus and the Station Black Out Metal Enclosed Bus.

This program, including its enhancements, will be implemented prior to the period of extended operation so that the intended functions of components within the scope of license renewal will be maintained during the period of extended operation.

A.2.1.34 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will be used to manage the aging effects of metallic parts of non-EQ electrical cable connections within the scope of license renewal during the period of extended operation. A representative sample of non-EQ electrical cable connections will be selected for one-time testing considering application (medium and low voltage), circuit loading (high loading) and location, with respect to connection stressors. The technical basis for the sample selected is to be documented. The specific type of test performed will be a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate to the application. The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be implemented prior to the period of extended operation.

A.2.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

This section provides summaries of the plant-specific programs credited for managing the effects of aging.

A.2.2.1 NICKEL ALLOY AGING MANAGEMENT PROGRAM

The TMI-1 Nickel Alloy Aging Management program is an existing program that manages cracking due to primary water stress corrosion cracking (PWSCC) for nickel alloy components. The Nickel Alloy Aging Management program uses a number of inspection techniques to detect cracking due to PWSCC, including surface examinations, volumetric examinations and bare metal visual examinations. The Nickel Alloy Aging Management program implements the inspection of components through an augmented In-service Inspection (ISI) program. The augmented program administers component evaluations, examination methods, scheduling, and site documentation as required to comply with regulatory, code or industry commitments related to Nickel Alloy issues. The Nickel Alloy Aging Management program includes mitigation and repair activities and strategies to ensure the long-term operability of nickel alloy components. The Nickel Alloy Aging Management program implements applicable Bulletins and Generic Letters and staff-accepted industry guidelines.

A.3 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

A.3.1 EVALUATION OF CHAPTER X AGING MANAGEMENT PROGRAMS

Aging Management Programs evaluated in Chapter X of NUREG-1801 are associated with Time-Limited Aging Analyses for metal fatigue of the reactor coolant pressure boundary, concrete containment tendon prestress, and environmental qualification (EQ) of electrical components. These programs are evaluated in this section.

A.3.1.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

The TMI-1 Metal Fatigue of Reactor Coolant Pressure Boundary program is an existing program credited for managing fatigue of reactor coolant pressure boundary components and other components. The program tracks the number of occurrences of significant thermal and pressure transients and compares the cumulative cycles to design limits. Several categories of transients are monitored, including reactor trips, heatups and cooldowns, power changes, secondary side temperature changes, hydrostatic tests, high-pressure injection cycles, and others.

The effect of the reactor coolant environment on TMI-1 fatigue usage has been evaluated for the sample components identified in NUREG/CR-6260 applicable for TMI-1 as a Babcock and Wilcox plant. In order to satisfactorily qualify each location, reduced numbers of transient cycles were required for several transient types, and these reduced numbers of cycles will be imposed as transient cycle administrative limits in the Metal Fatigue of Reactor Coolant Boundary aging management program prior to the period of extended operation to assure these environmental fatigue analyses will remain valid through the period of extended operation.

Prior to the period of extended operation, the program will be enhanced to add the statement: “Acceptable corrective actions include: reanalysis of the component to demonstrate that the design code limit will not be exceeded prior to or during the period of extended operation; repair of the component; replacement of the component, or other methods approved by the NRC.” It will be further enhanced to require consideration of environmental fatigue for additional reactor coolant pressure boundary locations if the fatigue usage for one of the environmental fatigue sample locations approaches its design limit.

The continued implementation of the TMI-1 Metal Fatigue of Reactor Coolant Pressure Boundary aging management program provides reasonable assurance that fatigue of reactor coolant pressure boundary components will be managed so that the intended functions of the components within the scope of License Renewal will be maintained during the period of extended operation.

A.3.1.2 CONCRETE CONTAINMENT TENDON PRESTRESS

The TMI-1 Concrete Containment Tendon Prestress aging management program is an existing program that is part of the TMI-1 ASME Section XI, Subsection IWL Program. The program is based on the 1992 Edition, with 1992 Addenda, of the ASME Boiler and Pressure Vessel Code, Section XI and includes confirmatory actions that monitor loss of containment tendon prestressing forces during the current term and will continue through the period of extended operation.

The program requires inspection of a sample of tendons from each group (vertical, hoop, and dome) in each inspection interval to confirm that individual tendon forces are above 95% of the applicable predicted values and that the mean of the normalized forces in each group is above the group mean required value (MRV). The program also requires that extrapolated trends do not fall below the MRVs prior to the time of the next regularly scheduled inspection. The predicted value, or base value, for individual tendons is developed consistent with the guidance presented in NRC Regulatory Guide 1.35.1. Trend line regression analysis is consistent with NRC Information Notice 99-10, Attachment 3. The regression analysis is revised after each inspection period to reflect the newly acquired data. In accordance with the requirements of 10CFR50.55a(b)(2)(viii)(B), an evaluation will be performed if the trend lines predict the prestressing forces in the containment to be below the MRV before the next scheduled inspection.

TMI-1 is committed to replacing the existing steam generators with new OTSGs prior to entering the period of extended operation. Repair/replacement and testing of the Reactor Building prestressing system, removed for access purposes, will be done in accordance with ASME Section XI, Subsection IWL.

A.3.1.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS

The Environmental Qualification (EQ) of Electrical Components is an existing program that manages the aging of electrical equipment within the scope of 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.” The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for equipment within the scope of the program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded. The aging effects are adequately managed so that the intended functions of components within the scope of 10 CFR 50.49 are maintained consistent with the current licensing basis during the period of extended operation.

A.4 TIME-LIMITED AGING ANALYSIS SUMMARIES

A.4.1 INTRODUCTION

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.

A.4.2 NEUTRON EMBRITTLEMENT OF THE REACTOR VESSEL AND INTERNALS

The reactor vessel embrittlement calculations for TMI-1 that evaluated reduction of fracture toughness of the TMI-1 reactor vessel beltline materials for 40 years are reported in the NRC RVID2 database and are based upon a predicted End of License fluence of 29 Effective Full Power Years (EFPY). These analyses are considered Time-Limited Aging Analyses (TLAAs) as defined in 10 CFR 54.21(c) and they must be evaluated for the increased neutron fluence associated with 60 years of operation.

A.4.2.1 NEUTRON FLUENCE ANALYSIS

End-of-life fluence is based on an assumed value of effective full power years over the licensed life of the plant. TMI-1 began operation in April 1974 but was shut down for six years from 1979 to 1985. As of the end of fuel cycle 15 in October 2005, the plant had been operated for approximately 21.1 EFPY. If the plant were operated at the maximum current licensed power level at a 100% capacity factor through the end of the period of extended operation, the plant would reach approximately 49.6 EFPY. Since this value does not include allowances for plant outages or periods at reduced power levels, a fluence projection that meets or exceeds 49.6 EFPY is bounding for 60 years of operation.

A 52 EFPY best-estimate analytical fluence update was prepared for Reactor Vessel beltline plates, forgings, and welds that included a benchmark comparison to measured cavity dosimetry test results. These projections were determined to meet the uncertainty requirements of Regulatory Guide 1.190, Revision 2. The NRC has reviewed this methodology and has issued a Safety Evaluation Report accepting it. Therefore, these 52 EFPY fluence values are suitable inputs for 60-year neutron embrittlement analyses for TMI-1.

A.4.2.2 CHARPY UPPER-SHELF ENERGY (USE) FOR BELTLINE PLATES AND FORGINGS

Charpy Upper Shelf Energy (USE) calculations were prepared for TMI-1 reactor vessel beltline plate and forging materials with projected neutron fluence greater than 10^{17} n/cm^2 after 60 years. The calculations were performed in accordance with NRC Regulatory Guide 1.99, Rev. 2, Radiation Embrittlement of Reactor Vessel Materials, using Regulatory Position 1.2, based on material chemistry and neutron fluence values valid for 52 EFPY. The 60-year USE values were demonstrated to be above 50 ft-lbs, which is acceptable. Position 2.2 was not used because there were no credible data for the TMI-1 surveillance materials (base metal C2789-2 and weld metals 72105 and 299L44).

A.4.2.3 CHARPY UPPER-SHELF ENERGY (USE) FOR BELTLINE WELDS

The Linde 80 beltline welds were all determined to have 60-year USE values below 50 ft lbs, necessitating an equivalent margins analysis (EMA) in accordance with the requirements of 10 CFR 50.60 for the period of extended operation. In addition, an evaluation was performed to determine if additional welds or components should be considered beltline materials due to the increased fluence levels.

An equivalent margins analysis was prepared with fluence values valid for 52 EFPY to determine the associated fracture toughness properties for the limiting welds after 60 years of operation. The results demonstrate that welds WF-25 and SA-1526 have adequate upper shelf energy toughness values to satisfy the requirement of 10 CFR 50, Appendix G, Section IV.A.1.a at 52 EFPY of reactor operation. It is further concluded that welds WF-25 and SA-1526 satisfy the acceptance criteria of Appendix K of ASME Code Section XI. Therefore, these welds have been demonstrated to have an equivalent margin of safety to that of ASME Section XI, Appendix G.

In addition to the beltline materials, adjacent components were reviewed to determine if they would meet the definition of reactor vessel beltline material as a result of the increased fluence associated with 60 years of operation. The outlet nozzle-to-shell attachment welds were evaluated and it was concluded that they are not limiting with regard to neutron embrittlement damage and do not meet the definition of beltline material for the period of extended operation. Since the outlet nozzle-to-shell welds are closer to the top of active fuel than the inlet nozzle-to-shell welds, the core flood nozzle-to-shell welds, inlet nozzle forgings, outlet nozzle forgings, or core flood nozzle forgings, it is bounding with respect to embrittlement. Since these additional welds and components have lower fluence than the outlet nozzle-to-shell welds, none of them meet the definition of reactor vessel beltline material.

A.4.2.4 PRESSURIZED THERMAL SHOCK LIMITS (RT_{PTS}) FOR REACTOR VESSEL MATERIALS DUE TO NEUTRON EMBRITTLEMENT

10 CFR 50.61(b)(1) provides rules for the protection of pressurized water reactors against pressurized thermal shock. Licensees are required to assess the projected values of reference temperature whenever a significant change occurs in projected values of RT_{PTS} or upon request for a change in the expiration date for the facility-operating license. RT_{PTS} was analyzed for 29 EFPY for the 40-year operating period, which is a TLAA.

For license renewal, 60-year RT_{PTS} values were calculated based upon fluence projections valid for 52 EFPY and were found to be less than the applicable screening criteria from 10 CFR 50.61(b)(2): 270°F for plates, forgings, and axial welds and 300°F for circumferential welds, which is acceptable. The limiting beltline material for TMI-1 is circumferential weld WF-70, with a RT_{PTS} value of 263.8°F, which is well below the applicable limit of 300°F.

A.4.2.5 REACTOR VESSEL OPERATING PRESSURE–TEMPERATURE LIMITS, INCLUDING ADJUSTED REFERENCE TEMPERATURES AND LOW TEMPERATURE OVERPRESSURIZATION LIMITS

10 CFR 50 Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. The P-T limits must account for the anticipated reactor vessel fluence. The current TMI-1 P-T and Low Temperature Overpressure Protection (LTOP) system limits are effective through 29 EFPY. These calculations associated with generation of the P-T limit curves satisfy the criteria of 10 CFR 54.3(a) and are considered TLAAs.

Updated Pressure-Temperature limits were calculated using fluence values valid for 52 EFPY for the TMI-1 reactor vessel beltline region, inlet and outlet nozzles, and closure head flange locations for normal heatup, normal cooldown, and inservice leak and hydrostatic (ISLH) test conditions. These P-T limits are expressed in the form of a curve of allowable pressure versus temperature. In addition, the minimum temperature for core criticality was determined to satisfy the regulatory requirements of 10 CFR 50, Appendix G. Temperature differences between the reactor coolant in the downcomer region and the 1/4 T wall location were also calculated for plant heatup and cooldown transients to support the development of LTOP system limits.

These updated P-T limits were developed in accordance with the analytical methods and flaw acceptance criteria of topical report BAW-10046A and ASME Section XI, Appendix G, 2003 Addendum. These updated P-T limits and LTOP limits show that an operating window will be available between the pressure-temperature limits and the net positive suction curves for the RC pumps at 60 years, providing a sufficient operating window for conduct of normal heatup and cooldown operations. Updated P-T limits and LTOP limits will be submitted to the NRC for approval prior to exceeding the 29 EFPY fluence values upon which the current P-T limits and LTOP limits are based.

A.4.2.6 NEUTRON EMBRITTLEMENT OF REACTOR VESSEL INTERNALS

The effect of irradiation on the material properties and deformation limits for the reactor vessel internals was evaluated for the current licensing basis. This analysis concluded that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits. This analysis is a TLAA that will be managed by the PWR Reactor Vessel Internals program during the period of extended operation.

TMI-1 commits to the following activities for the PWR Vessel Internals program:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals.

2. Evaluate and implement the results of the industry programs as applicable to the reactor internals.

3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, TMI-1 will submit an inspection plan for reactor internals to the NRC for review and approval.

A.4.3 METAL FATIGUE OF PIPING AND COMPONENTS

Metal fatigue was considered explicitly in the design process for pressure boundary components designed in accordance with ASME Section III, Class A or Class 1 requirements or USAS B31.7 requirements. Metal fatigue was evaluated implicitly for components designed in accordance with ASME Section III, Class 2 or 3 requirements or USAS B31.1 requirements. Each of these fatigue analyses and evaluations are considered to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the period of extended operation in accordance with 10 CFR 54.21(c).

A.4.3.1 ASME CLASS 1 AND USAS B31.7 PIPING AND COMPONENT FATIGUE ANALYSES

The TMI-1 components with an ASME Section III, Class A or Class 1 fatigue analysis or a USAS B31.7 fatigue analysis are addressed in this section, which includes the Reactor Vessel, Reactor Vessel Internals, Pressurizer, Reactor Coolant Pumps, Steam Generators, Class 1 piping and other components. Each of these fatigue analyses has a Cumulative Usage Factor (CUF) value less than 1.0, the code design limit, based upon assumed exposure to the full numbers of applicable 40-year design transients. TMI-1 will continue to manage fatigue of all Class 1 components using the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program during the period of extended operation, assuring that fatigue usage does not exceed 1.0 in 60 years. This program is further described in Appendix A, Section A.3.1.1.

A.4.3.2 EFFECTS OF REACTOR WATER ENVIRONMENTAL EFFECTS ON FATIGUE LIFE OF COMPONENTS AND PIPING (GSI-190)

The effect of the reactor coolant environment on TMI-1 fatigue usage has been satisfactorily evaluated for the sample components applicable for a Babcock and Wilcox plant in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.” The methodology from EPRI MRP-47, Revision 1, “Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,” was used for computing environmental fatigue correction factors, using the appropriate equations from NUREG/CR-6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels” and NUREG/CR-5704, “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels.” The methodology used to compute the environmental correction factor for nickel-alloy materials is based upon a paper entitled “Status of Fatigue Issues at Argonne National Laboratory,” presented by Omesh K. Chopra at the EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, August 1996.

In order to satisfactorily qualify each location, reduced numbers of transient cycles were required for several transient types, and these reduced numbers of cycles will be imposed as transient cycle administrative limits in the Metal Fatigue of Reactor Coolant Boundary aging management program prior to the period of extended operation to assure these environmental fatigue analyses will remain valid through the period of extended operation.

A.4.3.3 ASME CLASS 2 AND 3 AND USAS B31.1 PIPING AND COMPONENT FATIGUE ANALYSES

The non-Class 1 piping and components at TMI-1 were designed in accordance with USAS B31.1.0 -1967 or ASME Section III, Class 2 or 3 requirements. These codes require determination of the overall number of thermal design cycles for the component and application of stress range reduction factors if the total number of cycles exceeds 7,000 cycles. Since these analyses were based upon the number of cycles expected to occur during the original license period, which could potentially increase during 60 years of operation, these analyses are considered to be TLAAs.

In order to evaluate these TLAAs for 60 years, the numbers of cycles expected to occur in 60 years were determined and were found to be less than the 7,000 cycle threshold for stress range reduction factors, so the current implicit fatigue evaluations remain valid for the period of extended operation. Since they are based upon applicable design transients for Class 1 components, the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program will continue to be used to manage fatigue. The program monitors the number of occurrences of the significant design transients and assures they are not permitted to exceed the design limits, thus managing the potential for cracking resulting from fatigue.

A.4.3.4 REACTOR VESSEL INTERNALS FATIGUE

The Reactor Vessel Internals were designed and constructed prior to the development of ASME Code requirements for core support structures, but the reactor coolant system functional design requirements were considered in the design. Design cyclic loadings and thermal conditions for the B&W-designed reactor coolant system Class 1 components are defined by the component design specifications. These design transient cycles were used implicitly in the original design analyses for the reactor vessel internals, which have been identified as TLAAs.

Since these implicit fatigue evaluations for reactor vessel internals are based upon the design transient limits for Class 1 components, the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program will continue to be used to manage fatigue. The program monitors the number of occurrences of the significant design transients and assures they are not permitted to exceed the design limits, thus managing the potential for cracking resulting from fatigue.

A.4.3.5 REACTOR VESSEL INTERNALS FLOW-INDUCED VIBRATION ANALYSIS

The Flow-Induced Vibration (FIV) analysis for the reactor vessel internals components showed that the maximum alternating stresses for each component was below the applicable alternating stress endurance limits (derived from the ASME Section III fatigue curves) and would therefore not experience fatigue cracking. The components analyzed include the stainless steel incore instrumentation nozzles, the incore instrumentation guide tubes, the flow distributor, the flow distributor assembly support plate, the thermal shield, the inlet baffle, and bolting. This FIV analysis has been identified as a TLAA.

In the analysis, the highest zero-to-peak alternating stresses due to FIV were compared with an endurance limit for the material applicable for 10^{12} cycles, which was the number of cycles postulated for a 40-year plant life. Since the alternating stress was below the endurance limit for the material, fatigue cracking was not predicted to occur during the 40-year period.

In order to project these analyses for 60 years, the endurance limit curves have been extended to 10^{13} cycles accommodate the higher number of cycles that could potentially occur during the period of extended operation. Each of the alternating stress values was determined to be less than the 60-year endurance limit, so the TLAA has been satisfactorily projected for 60 years and fatigue cracking is not predicted to occur for Reactor Vessel Internals due to flow-induced vibration during the period of extended operation.

A.4.3.6 UNDERCLAD CRACKING EVALUATION FOR REACTOR VESSEL

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld cladding were detected in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged-arc welding processes. BAW-10013A contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to fatigue design cycles. The flaw growth analysis was initially performed for 40-year cyclic loading, and a 40-year end-of-life fluence value of 3×10^{19} n/cm^2 (E>1.0MeV) was used to determine fracture toughness properties. The report concluded that the intergranular separation found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the Atomic Energy Commission.

To cover the period of extended operations, a generic license renewal program analysis was performed using current ASME Code requirements. This analysis is fully described in BAW-2274-A, which updates and supersedes the fracture mechanics analysis for underclad cracking as originally reported in BAW-10013-A based upon 48 EFPY fluence values. The revised analysis concluded that the evaluation of postulated underclad cracking in the reactor vessel meets the acceptance criteria of the 1989 Edition of the ASME Code, Section XI, IWB-3612. The fracture toughness margin for emergency and faulted conditions was 2.42, which is greater than the required toughness margin of 1.41.

Since TMI-1 is expected to have fluence values that exceed 48 EFPY, the analysis was reevaluated using fluence values valid for 52 EFPY. BAW-2274-A was determined to remain bounding for TMI-1 at 52 EFPY, including the conclusion that intergranular separation will not lead to vessel failure at TMI-1 during the period of extended operation.

A.4.3.7 REACTOR COOLANT PUMP MOTOR FLYWHEEL FATIGUE CRACK GROWTH ANALYSIS

Westinghouse report WCAP-14535A, "Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination," includes a fatigue crack growth analysis that has been identified as a TLAA. The report was submitted for NRC review in January 1996 and the NRC issued a Safety Evaluation Report in September 1996. The purpose of the report was to provide an engineering basis for elimination of flywheel inservice inspection requirements for all operating Westinghouse plants and for certain Babcock and Wilcox plants with Westinghouse reactor coolant pumps, specifically including TMI-1.

The analysis addresses crack growth of a postulated flaw and compares this growth to a critical flaw size to determine whether or not a failure would occur under maximum overspeed conditions. A peak LOCA speed of 1500 rpm was used in evaluation of the Babcock and Wilcox units. To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10 percent of the flywheel thickness was assumed. Since the maximum stress intensity range occurs between RCP shutdown (zero rpm) and the normal operating speed of approximately 1200 rpm, the number of cycles is the same as the number of RCP starts and stops. 6000 cycles of RCP starts and stops were assumed in the analysis. Crack growth was shown to be negligible, with the bounding crack growth of 0.08 inches resulting from exposure to 6000 cycles.

RCP flywheel cycles are associated with pump starts and stops resulting from plant heatups and cooldowns, which are limited to 240 cycles by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program. If this is multiplied by a factor of 1.5 to project the potential cycles for 60 years of operation, the result is 360 cycles in 60 years. Since the projected number of RCP starts and stops is not expected to exceed the 6000 cycles analyzed in the topical report during the period of extended operation, the analysis has been projected for 60 years. Therefore, reactor coolant pump flywheel crack growth will be negligible during the period of extended operation.

A.4.4 LEAK-BEFORE-BREAK ANALYSIS OF PRIMARY SYSTEM PIPING

The Leak-Before-Break (LBB) analysis for the RCS primary piping of TMI-1 NSS systems is contained in topical report BAW-1847, Revision 1, “B&W Owners Group Leak-Before Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W-Designed NSS,” November 1985, and BAW-1999, “TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping,” October 1987. These analyses were reviewed and approved by the NRC staff for the current licensing period. These reports successfully demonstrated the application of LBB to the TMI-1 RCS primary system. The LBB evaluations included fatigue flaw growth analyses, flaw stability analyses, and limit load analyses. In addition, the report qualitatively addressed thermal aging of Reactor Coolant Pump (RCP) casings for the current period.

The portions of the analysis that are considered to be time-limited aging analyses (TLAAs) are 1) the fatigue flaw growth analysis contained in topical report BAW-1847, Revision 1 (and referenced in topical report BAW-1999) and 2) the evaluation of thermal embrittlement of cast austenitic stainless steel (CASS).

A.4.4.1 FATIGUE FLAW GROWTH ANALYSIS

A surface flaw was postulated at selected locations of the piping system (i.e. highest stress coincident with the lower bound of materials properties for base metal, welds, and safe ends). A fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions. The fatigue flaw growth is based upon design transient inputs, including 240 heatup/cooldown cycles and 22 safe shutdown earthquake (SSE) events, originally postulated to bound 40 years of operation.

The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program limits heatup and cooldown cycles to 240 cycles. Transient occurrence data was reviewed and projections were made to determine the total number of transients that would occur in 60 years. The results showed that these limits will not be exceeded in 60 years at the average rate of occurrence in the past. The original design allowance of 22 SSE events remains bounding for the period of extended operation since no earthquake cycles of magnitudes assumed for SSE events have occurred to-date. Therefore, the flaw growth analysis in BAW-1847, Revision 1, is considered applicable for 60 years of operation since the transient cycles used as design inputs will not be exceeded during the period of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program will continue to be used during the period of extended operation to assure these limits are not exceeded.

A.4.4.2 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL REACTOR COOLANT PUMP CASINGS

The leak-before-break analyses described above use material property assumptions that account for the reduction in fracture toughness properties of cast austenitic stainless steel (CASS) associated with thermal embrittlement. The relevant aging effect is the reduction in the fracture toughness of the material as a function of time at temperature above 482°F. This analysis has been identified as a TLAA that requires evaluation for the period of extended operation.

An updated flaw stability analysis has been performed to demonstrate that thermal embrittlement of the CASS nozzles will not prevent these components from performing their intended functions during the period of extended operation. The analysis demonstrated that the CASS RCP casing materials meet all safety margin requirements of Standard Review Plan (SRP) 3.6.3, which provided the acceptance standards for the leak-before-break analysis for commercial nuclear reactor piping. Based upon the results of this analysis, it is concluded that the TMI-1 RCP CASS components meet all safety margin requirements of SRP 3.6.3 with consideration of thermal aging and are acceptable for the period of extended operation.

A.4.5 FUEL TRANSFER TUBE BELLOWS DESIGN CYCLES

The fuel transfer tube connects the fuel transfer canal (inside the primary containment building) to the spent fuel pool (inside the fuel handling building). The transfer tube passes through the primary containment wall and through exterior wall of the fuel handling building.

The fuel handling building penetration is comprised of a penetration sleeve through the wall, a flexible bellows outside the wall that connects the penetration sleeve to the transfer tube, and a second flexible bellows inside the wall that connects the penetration sleeve to the transfer tube. The penetration sleeve and the two bellows perform a fuel handling building leakage boundary intended function, and are within the scope of license renewal.

Inside the containment building, a flexible bellows is located where the fuel transfer tube penetrates the fuel transfer canal. It performs a leakage boundary function preventing refueling water from leaking inside containment. This bellows does not perform a primary containment pressure boundary function since the penetration sleeve, closure plate and fuel transfer tube perform that function.

Each of these three bellows was designed for a minimum of 5,000 cycles of expansion and contraction cycles for 40 years, so these design analyses are TLAAs requiring evaluation for the period of extended operation.

In order to determine if the design analyses remain valid for 60 years of operation, the number of cycles for 60 years has been conservatively projected. For each of these components, one thermal cycle occurs during each refueling operation. The cycle begins when the transfer canal is filled with water for refueling and ends when the canal is drained at the end of the refueling operation. The number of refueling operations in 60 years is conservatively estimated to be 40 cycles, based upon one refueling operation every 18 months. This is conservative because refueling operations are now conducted once every 24 months. In addition to these cycles, the fuel transfer canal penetration assembly is exposed to pressurization cycles during Integrated Leak Rate Tests, conservatively projected to occur once every 5 years, compared to a maximum interval of once per 10 years. This contributes 12 additional cycles in 60 years. These penetrations would also be exposed to up to 20 Safe Shutdown Earthquake cycles. Therefore, the total cycles projected for 60 years is 40 + 12 + 20 = 72 cycles.

Therefore, since the number of cycles projected to occur in 60 years is well below the 5,000 design cycles analyzed for these bellows, these design analyses remain valid for the period of extended operation.

A.4.6 CRANE LOAD CYCLE LIMITS

The load cycle limits for cranes was identified as a potential TLAA. The following TMI-1 Nuclear Station cranes are in the scope of License Renewal and have been identified as having a TLAA, which requires evaluation for 60 years:

- Reactor Building Crane
- Fuel Handling Building Crane

The method of review applicable to the crane cyclic load limit TLAA involves reviewing the existing 40-year design basis to determine the number of load cycles considered in the design of each of the cranes in the scope of License Renewal and developing 60-year projections for load cycles for each of the cranes in the scope of License Renewal for comparison with the number of design cycles for 40 years used in the design.

A.4.6.1 REACTOR BUILDING CRANE

The purchasing specification for the 185-ton Reactor Building Crane at TMI-1 required that the crane conform to the design requirements of EOCI-61, "Specifications for Electric Overhead Traveling Cranes – 1961," prior to the issuance of the Crane Manufacturers Association of America (CMAA) Specification 70. However, the design of this crane corresponds to the cyclic loading requirements of CMAA 70, Class A1, which is a minimum of 20,000 load cycles. As stated in the TMI-1 response dated February 21, 1984 to NUREG-0612, Control of Heavy Loads in Nuclear Power Plants, the total number of lift cycles for any of the crane members will be less than 2000 over the original 40-year life of the plant. This can be multiplied by a factor of 1.5 to determine the number of cycles for 60-year life. Therefore, the total number of cycles is projected to be less than 3000 for the total 60-year life of the plant. This is considerably less than the allowable design value of 20,000 cycles. Therefore, the Reactor Building Crane load cycle fatigue analysis has been successfully projected for 60 years of plant operation.

A.4.6.2 FUEL HANDLING BUILDING CRANE

The purchasing specification for the 110-ton Fuel Handling Building Crane at TMI-1 required the crane to conform to the design requirements of EOCI-61, "Specifications for Electric Overhead Traveling Cranes – 1961," prior to the issuance of the Crane Manufacturers Association of America (CMAA) Specification 70. However, the design of this crane corresponds to the cyclic loading requirements of CMAA 70, Class A1, which is minimum of 20,000 load cycles. As stated in the TMI-1 response dated February 21, 1984 to NUREG-0612, Control of Heavy Loads in Nuclear Power Plants, the total number of lift cycles for any of the crane members will be less than 2000 over the original 40-year life of the plant. This can be multiplied by a factor of 1.5 to determine the number of cycles for 60-year life. Therefore, the number of load cycles projected for 60-year life is less than 3000. This is considerably less than the 20,000 permissible cycles and is therefore acceptable. Therefore, the Fuel Handling Building Crane load cycle fatigue analysis has been successfully projected for 60 years of plant operation.

A.4.7 LOSS OF PRESTRESS IN CONCRETE CONTAINMENT TENDONS

The TMI-1 Reactor Building (Containment) is a reinforced and post-tensioned concrete structure composed of a cylindrical wall with a flat foundation mat and a shallow dome roof. A massive ring girder provides a transition between the wall and dome. The foundation mat is reinforced with conventional mild reinforcing steel. The cylindrical wall and dome are post-tensioned by an un-grouted BBRV (parallel, button headed wires) pre-stressing system. The wall is pre-stressed by 166 vertical tendons anchored at the top of the ring girder and the bottom of the base mat and 330 hoop tendons anchored at six vertical buttresses equally spaced around the cylinder wall. The hoop tendons, which span just over 120 degrees of arc, are arranged into 6 overlapping sub-groups. The dome is pre-stressed by 147 tendons that anchor at the vertical face of the ring girder. The dome tendons are arranged into 3 sub-groups of 49 parallel (in plan) tendons; the groups intersect at 60 degrees.

The tendons consist of 169 wires of ¼ inch diameter with a specified minimum ultimate tensile strength of 240 ksi and they are enclosed in galvanized steel conduits filled with a corrosion protection medium (grease). Tendons were initially tensioned to a force of approximately 1,400 kip. The original design included a calculation of expected loss of prestress for plant life in accordance with ACI 318-63. The calculation accounted for pre-stressing force loss due to elastic shortening during initial stressing operations as well as subsequent time dependent losses resulting from concrete shrinkage, concrete creep and tendon stress relaxation. The time dependent losses were calculated for 40 years and documented in vendor manual VM-TM-2485, as referenced in UFSAR Section 5.7.5.2.3b, which is a TLAA.

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Periodic tendon surveillance activities are implemented in accordance with the TMI-1 Concrete Containment Tendon Prestress program as required by Technical Specification 4.4.2. The program is based on ASME Section XI, Subsection IWL, as incorporated by reference in 10 CFR 50.55a, and guidance of Regulatory Guide 1.35. The forces in randomly sampled tendons are measured periodically to verify that long-term losses are following an acceptable trend. Regression analyses incorporating current and prior surveillance measurements show that trended vertical, hoop and dome group mean forces will not fall below the minimum required value (MRV) specified in UFSAR Section 5.7.5.2.3 prior to the deadline for completion of the subsequent surveillance.

The program predicts 95% lower confidence limits (LCL's) of the lift-off force for each tendon group (vertical, hoop, dome) by regression analysis of individual tendon surveillance data, and maintains trend lines of the tendons surveyed. The program also requires inspection of a sample of tendons from each group during each inspection interval to confirm that the trend lines remain above the 95% LCL, and therefore that tendon prestress will remain above their respective minimum required value (MRV) for the succeeding inspection interval. The program requires initiation of corrective actions if surveillance data indicate that a trend line may cross its MRV prior to the next regularly scheduled inspection interval. See Appendix B, Concrete Containment Tendon Prestress (B.3.1.2), for a more detailed description of the aging management program.

A.4.8 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL COMPONENTS

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components, developed to meet 10 CFR 50.49 requirements, have been identified as time-limited aging analyses (TLAAs) for TMI-1. The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50.49 and 10 CFR 50, Appendix A, Criterion 4. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Harsh environments are defined as those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant accident (LOCA), high energy line break (HELB), or post-LOCA radiation. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

The TMI-1 EQ Program will manage the effects of aging effects for the components associated with the environmental qualification TLAA. This program implements the requirements of 10 CFR 50.49 (as further defined and clarified by NUREG-0588, and RG 1.89, Rev. 1). Reanalyses of component aging evaluations are performed on a routine basis to extend the qualifications of components as part of the TMI-1 EQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). The TMI-1 EQ Program methodology is further described in Appendix B, Section B.3.1.3.

Under the TMI-1 EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful.

A.5 License Renewal Commitment List

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
1.	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Existing program is credited.	A.2.1.1	Ongoing	Section B.2.1.1
2.	Water Chemistry	Existing program is credited. The program will be enhanced to incorporate the continuous monitoring of sodium in steam generator blowdown, making it consistent with EPRI 1008224, Pressurized Water Reactor Secondary Water Chemistry Guidelines, Revision 6.	A.2.1.2	Prior to the period of extended operation.	Section B.2.1.2
3.	Reactor Head Closure Studs	Existing program is credited.	A.2.1.3	Ongoing	Section B.2.1.3
4.	Boric Acid Corrosion	Existing program is credited.	A.2.1.4	Ongoing	Section B.2.1.4
5.	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors	Existing program is credited.	A.2.1.5	Ongoing	Section B.2.1.5
6.	Flow-Accelerated Corrosion	Existing program is credited.	A.2.1.6	Ongoing	Section B.2.1.6
7.	Bolting Integrity	Existing program is credited.	A.2.1.7	Ongoing	Section B.2.1.7
8.	Steam Generator Tube Integrity	Existing program is credited.	A.2.1.8	Ongoing	Section B.2.1.8
9.	Open-Cycle Cooling Water System	Existing program is credited. The program will be enhanced by adding a new river water chemical treatment system to treat the river water systems for biofouling.	A.2.1.9	Prior to the period of extended operation.	Section B.2.1.9

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
10.	Closed-Cycle Cooling Water System	Existing program is credited. The program will be enhanced to include a one-time inspection of selected components in stagnant flow areas to confirm the absence of aging effects resulting from exposure to closed cycle cooling water. Also, a one-time inspection of selected CCCW chemical mix tanks and associated piping components will be performed to verify corrosion has not occurred on the interior surfaces of the tanks and associated piping components.	A.2.1.10	Prior to the period of extended operation.	Section B.2.1.10
11.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Existing program is credited. The program will be enhanced to include visual inspection of rails in the rail system for loss of material due to wear, and visual inspection of structural bolting for loss of material. Acceptance criteria will be enhanced to require that significant loss of material due to wear will be evaluated or corrected to ensure the intended function of the crane or hoist is not impacted.	A.2.1.11	Prior to the period of extended operation.	Section B.2.1.11
12.	Compressed Air Monitoring	Existing program is credited. The program will be enhanced to include air quality testing for dew point, particulates, lubricant content, and contaminants to ensure that the contamination standards of ANSI/ISA-S7.0.01-1996, paragraph 5 are met. In addition the program will be enhanced to include air quality sampling on a representative sampling of headers on a yearly basis in accordance with the guidelines of ASME OM-S/G-1998, Part 17 and EPRI TR-108147.	A.2.1.12	Prior to the period of extended operation.	Section B.2.1.12
13.	Fire Protection	Existing program is credited. The program will be enhanced to include additional inspection criteria for degradation of fire barrier walls, ceilings, and floors, and specific fuel supply line inspection criteria for diesel-driven fire pumps during tests.	A.2.1.13	Prior to the period of extended operation.	Section B.2.1.13

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
14.	Fire Water System	Existing program is credited. The program will be enhanced to include sprinkler head testing in accordance with NFPA 25, "Inspection, Testing and Maintenance of Water-Based Fire Protection Systems." Samples will be submitted to a testing laboratory prior to being in service 50 years. This testing will be repeated at intervals not exceeding 10 years. Prior to the period of extended operation, the program will be enhanced to include periodic non-intrusive wall thickness measurements of selected portions of the fire water system at an interval not to exceed every 10 years.	A.2.1.14	Prior to the period of extended operation.	Section B.2.1.14
15.	Aboveground Steel Tanks	Existing program is credited. The program will be enhanced to include one-time thickness measurements of the bottom of the Condensate Storage Tanks, which are supported on concrete foundations. The measurements will be taken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. The program will also be enhanced to inspect the sealant at the tank-foundation interface.	A.2.1.15	Prior to the period of extended operation.	Section B.2.1.15

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
16.	Fuel Oil Chemistry	Existing program is credited. The program will be enhanced to include: • The analysis of new fuel oil for specific or API gravity, kinematic viscosity, and water and sediment prior to filling the fuel oil storage tanks followed by full spectrum analysis within 31 days after the addition of the fuel oil into the fuel oil storage tanks. • The determination of water and sediment and particulate contamination in accordance with ASTM standards. • The analysis for bacteria in new and stored fuel oil. • The addition of biocides, stabilizers, or corrosion inhibitors as determined by fuel oil analysis activities. • Activities to periodically drain water and sediment from tank bottoms, and, activities to periodically drain, clean, and inspect fuel oil tanks. • Manual sampling in accordance with ASTM standards and required frequencies. • The use of ultrasonic techniques for determining tank bottom thicknesses should there be any evidence of loss of material due to general, pitting, crevice, and microbiologically influenced corrosion, and fouling found during visual inspection activities.	A.2.1.16	Prior to the period of extended operation.	Section B.2.1.16

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
17.	Reactor Vessel Surveillance	Existing program is credited. The program will be enhanced to address maintenance of the TMI-1 cavity dosimetry exchange schedule. The program will also be enhanced to clarify that, if future plant operations exceed the limitations or bounds specified in Regulatory Position 1.3 of RG 1.99, Rev. 2, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified.	A.2.1.17	Prior to the period of extended operation.	Section B.2.1.17
18.	One-Time Inspection	Program is new. The program will be used to provide reasonable assurance that an aging effect is not occurring, or that the aging effect is occurring slowly enough to not affect a components intended function during the period of extended operation, and therefore will not require additional aging management. The program will be credited for cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, (b) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse than that generally expected, or (c) the characteristics of the aging effect include a long incubation period. This program will be used for the following: • To confirm the effectiveness of the Water Chemistry program to manage the loss of material, cracking, and the reduction of heat transfer aging effects for steel, stainless steel, copper alloy, nickel alloy, and aluminum alloy in treated water, steam, and reactor coolant environments. • To confirm the effectiveness of the Fuel Oil Chemistry program to manage the loss of material	A.2.1.18	Prior to the period of extended operation.	Section B.2.1.18

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
		aging effect for steel, stainless steel, and copper alloy in a fuel oil environment. • To confirm the effectiveness of the Lubricating Oil Analysis program to manage the loss of material and the reduction of heat transfer aging effects for steel, stainless steel, copper alloy, and aluminum alloy in a lubricating oil environment. • To confirm the loss of material aging effect is insignificant for stainless steel and copper alloy in an air/gas – wetted environment. Inspection methods will include visual examination or volumetric examinations. Acceptance criteria will be in accordance with industry guidelines, codes, and standards. The One-Time Inspection program provides for the evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation. Should aging effects be detected, the program triggers actions to characterize the nature and extent of the aging effect and determines what subsequent monitoring is needed to ensure intended functions are maintained during the period of extended operation.			

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
19.	Selective Leaching of Materials	Program is new. The program will be used to manage the loss of material due to selective leaching. The program includes inspection of a representative sample of susceptible components to determine if loss of material due to selective leaching is occurring. One-time inspections will include visual examinations, supplemented by hardness tests, and other examinations, as required. If selective leaching is found, the condition will be evaluated to determine the need to expand inspection scope.	A.2.1.19	Prior to the period of extended operation.	Section B.2.1.19
20.	Buried Piping and Tanks Inspection	Existing program is credited. The program will be enhanced to include: • Inspection of buried stainless steel piping and components prior to entering the period of extended operation • Inspection of buried cast iron, carbon steel, concrete-coated carbon steel, and stainless steel piping and components within ten years of entering the period of extended operation • Internal inspection and UT of the D.G. Fuel Storage 30,000 Gallon Tank prior to the period of extended operation, and within ten years of entering the period of extended operation	A.2.1.20	Prior to the period of extended operation.	Section B.2.1.20

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
21.	External Surfaces Monitoring	Program is new. The program will be used to manage aging effects through visual inspection of external surfaces for evidence of hardening and loss of strength and loss of material. The program directs visual inspections that are performed during system walkdowns. The program consists of periodic visual inspection of components such as piping, piping components, ducting, and other components within the scope of license renewal. Visual inspections may be augmented by physical manipulation to detect hardening and loss of strength of elastomers.	A.2.1.21	Prior to the period of extended operation.	Section B.2.1.21
22.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Program is new. The program will be used to manage cracking due to stress corrosion cracking; hardening and loss of strength due to elastomer degradation; loss of material due to general, pitting, crevice, and microbiologically influenced corrosion and fouling; and reduction of heat transfer due to fouling. The program includes provisions for visual inspections of the internal surfaces and volumetric testing of components not managed under any other aging management program.	A.2.1.22	Prior to the period of extended operation.	Section B.2.1.22
23.	Lubricating Oil Analysis	Existing program is credited.	A.2.1.23	Ongoing	Section B.2.1.23
24.	ASME Section XI, Subsection IWE	Existing program is credited.	A.2.1.24	Ongoing	Section B.2.1.24
25.	ASME Section XI, Subsection IWL	Existing program is credited.	A.2.1.25	Ongoing	Section B.2.1.25
26.	ASME Section XI, Subsection IWF	Existing program is credited.	A.2.1.26	Ongoing	Section B.2.1.26
27.	10 CFR Part 50, Appendix J	Existing program is credited.	A.2.1.27	Ongoing	Section B.2.1.27

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
28.	Structures Monitoring Program	Existing program is credited. The program will be enhanced to include: • Inspection of penetration seals in the Service Building, UPS Diesel Building, Mechanical Draft Cooling Tower Structures, and Miscellaneous Yard Structures (Foundation for condensate storage tank, borated water storage tank, diesel fuel storage tank, altitude tank, duct banks, manholes). • Monitoring of Penetration Seals. • Monitoring of the Intake Canal for Loss of material and loss of form. • Monitoring of electrical panels, junction boxes, instrument panels, and conduits for loss of material due to corrosion. • Monitoring of ground water chemistry by periodically sampling, testing, and analysis of ground water to confirm that the environment remains non-aggressive for buried reinforced concrete. • Monitoring of reinforced concrete submerged in raw water associated with Intake Screen and Pumphouse, Circulating Water Pump House, Mechanical Draft Cooling Tower Structures, Natural Draft Cooling Tower Basins, and Dike/Flood Control System. • Monitoring of vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF, for reduction or loss of isolation function. • Monitoring of HVAC duct supports for loss of material.	A.2.1.28	Prior to the period of extended operation.	Section B.2.1.28

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
		• Parameters monitored will be enhanced to include plausible aging effects and mechanisms. • Monitoring of concrete structures for a reduction in anchor capacity due to local concrete degradation. This will be accomplished by visual inspection of concrete surfaces around anchors for cracking, and spalling. • Revised acceptance criteria to provide details specified in ACI 349.3R-96.			
29.	Protective Coating Monitoring and Maintenance Program	Existing program is credited.	A.2.1.29	Ongoing	Section B.2.1.29
30.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Program is new. The program will be used to manage aging of non-EQ cables and connections during the period of extended operation. A representative sample of accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for indications of accelerated insulation aging such as embrittlement, discoloration, cracking, or surface contamination. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection.	A.2.1.30	Prior to the period of extended operation.	Section B.2.1.30

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
31.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Existing program is credited. The program will be enhanced to manage the aging of the cable and connection insulation of the in scope radiation monitoring and nuclear instrumentation circuits in the Radiation Monitoring and Nuclear Instrumentation and Incore Monitoring Systems. The in scope radiation monitoring and nuclear instrumentation circuits are sensitive instrumentation circuits with low-level signals and are located in areas where the cables and connections could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Calibration testing and performance monitoring are currently being performed for in scope radiation monitoring circuits. Direct cable testing will be performed as an enhancement to ensure that the cable and connection insulation resistance is adequate for the nuclear instrumentation circuits to perform their intended functions.	A.2.1.31	Prior to the period of extended operation.	Section B.2.1.31

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
32.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Program is new. The program will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible medium voltage cables. These cables may at times be exposed to significant moisture simultaneously with significant voltage. The TMI-1 cables in the scope of this aging management program will be tested using a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. The cables will be tested at least once every 10 years. Manholes associated with the cables included in this aging management program will be inspected for water collection at least twice a year, in accordance with existing practices, and drained as required.	A.2.1.32	Prior to the period of extended operation.	Section B.2.1.32

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
33.	Metal Enclosed Bus	Existing program is credited. The program will be enhanced to include the following inspection criteria: • The internal portion of the metal enclosed bus will be visually inspected for cracks, corrosion, foreign debris, excessive dust build-up and evidence of moisture intrusion. • The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. • The internal bus supports will be visually inspected for structural integrity and signs of cracks. The program will also be enhanced to perform internal visual inspections on the 480V Metal Enclosed Bus and the Station Black Out Metal Enclosed Bus.	A.2.1.33	Prior to the period of extended operation.	Section B.2.1.33
34.	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Program is new. The program will be used to manage the aging effects of metallic parts of non-EQ electrical cable connections within the scope of license renewal during the period of extended operation. A representative sample of non-EQ electrical cable connections will be selected for one-time testing considering application (medium and low voltage), circuit loading (high loading) and location, with respect to connection stressors. The technical basis for the sample selected is to be documented. The specific type of test performed will be a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate to the application.	A.2.1.34	Prior to the period of extended operation.	Section B.2.1.34

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
35.	Nickel Alloy Aging Management Program	Existing program is credited. TMI-1 commits to implement applicable Bulletins, Generic Letters, and staff-accepted industry guidelines.	A.2.2.1	Ongoing	Section B.2.2.1
36.	PWR Vessel Internals	TMI-1 commits to the following activities for the PWR Vessel Internals program: • Participate in the industry programs for investigating and managing aging effects on reactor internals. • Evaluate and implement the results of the industry programs as applicable to the reactor internals. • Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	N/A	Prior to the period of extended operation.	N/A
37.	Metal Fatigue of Reactor Coolant Pressure Boundary	Existing program is credited. The program will be enhanced to add the statement: "Acceptable corrective actions include: reanalysis of the component to demonstrate that the design code limit will not be exceeded prior to or during the period of extended operation; repair of the component; replacement of the component, or other methods approved by the NRC." In addition, the program will be enhanced to require a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit.	A.3.1.1	Prior to the period of extended operation.	Section B.3.1.1
38.	Concrete Containment Tendon Prestress	Existing program is credited.	A.3.1.2	Ongoing	Section B.3.1.2

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA Appendix A)	Enhancement or Implementation Schedule	Source
39.	Environmental Qualification (EQ) of Electrical Components	Existing program is credited.	A.3.1.3	Ongoing	Section B.3.1.3
40.	Steam Generators	Install new Once Through Steam Generators (OTSGs) prior to the period of extended operation.	N/A	Prior to the period of extended operation.	N/A
41.	New P-T Curves	Revised pressure-temperature (P-T) limits and low-temperature overpressurization (LTOP) limits for a 60-year operating life have been prepared and will be submitted to the NRC for approval.	A.4.2.5	Prior to exceeding the 29 EFPY fluence values upon which the current P-T limits and LTOP limits are based.	Section 4.2.5