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A0 INTRODUCTION

As required by 10 CFR 54.21(d), this appendix contains summary descriptions of the Aging Management Program activities and the Time-Limited Aging Analyses (TLAAs) for the period of extended operation. Following the issuance of the renewed operating licenses, Appendix A1 will be incorporated into the NMP1 Updated Final Safety Analysis Report (UFSAR) and Appendix A2 will be incorporated into the NMP2 Updated Safety Analysis Report (USAR).

A1 APPENDIX A1 – NMP1 UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT

A1.1 AGING MANAGEMENT PROGRAMS

A1.1.1 10 CFR 50 APPENDIX J PROGRAM

The 10 CFR 50 Appendix J Program detects degradation of the containment structure and components that comprise the containment pressure boundary, including seals and gaskets. Containment leak rate tests are performed to assure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage limits specified in the Technical Specifications. This program complies with Option B requirements of 10 CFR 50 Appendix J with plant-specific exceptions approved by the NRC as part of license amendments, and implements the guidelines provided in NRC Regulatory Guide (RG) 1.163 and NEI 94-01.

A1.1.2 ASME SECTION XI INSERVICE INSPECTION (SUBSECTION IWE) PROGRAM

The American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection (Subsection IWE) Program (referred to herein as the IWE ISI Program) manages aging effects due to (1) corrosion of carbon steel components comprising the containment pressure boundary; and (2) degradation of containment pressure-retaining polymers. Program activities include visual examination, with limited surface or volumetric examinations when augmented examination is required. The IWE ISI Program is based on the 1998 edition of the ASME Boiler and Pressure Vessel Code, Section XI (Subsection IWE) for containment inservice inspection with plant-specific exceptions approved by the NRC. This is an exception to the evaluation in NUREG-1801 (which covers ASME Section XI requirements from both the 1992 edition with the 1992 addenda and the 1995 edition with the 1996 addenda).

A1.1.3 ASME SECTION XI INSERVICE INSPECTION (SUBSECTION IWF) PROGRAM

The ASME Section XI Inservice Inspection (Subsection IWF) Program (referred to herein as the IWF ISI Program) manages aging of carbon steel component and piping supports due to general corrosion and wear. Program activities include visual examination to determine the general mechanical and structural condition of components and their supports. The IWF ISI Program is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code, Section XI (Subsection IWF) for inservice inspection of supports and implements the alternate examination requirements of ASME Code Case N-491-1. These are exceptions to the evaluation in NUREG-1801 (which covers ASME Section XI requirements from the 1989 edition through the 1995 edition and addenda through the 1996 addenda).

A1.1.4 ASME SECTION XI INSERVICE INSPECTION (SUBSECTIONS IWB, IWC, IWD) PROGRAM

The ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program manages aging of Class 1, 2, or 3 pressure-retaining components and their integral attachments. Program activities include periodic visual, surface, and/or volumetric examination and pressure tests of Class 1, 2 and 3 pressure-retaining components. The ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, Section XI (Subsections IWB, IWC, and IWD) for inservice inspection of pressure-retaining components and their integral attachments, with the risk-informed requirements of ASME Code Case N-578-1 implemented for examination of welds in Class 1 and 2 piping as approved by the NRC in plant-specific exemptions. These are exceptions to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda).

A1.1.5 BORAFLEX MONITORING PROGRAM

The Boraflex Monitoring Program is an existing program that manages degradation of neutron absorbing material in spent fuel pool storage racks resulting from radiation exposure and possible water ingress. Program activities include (1) correlation of measured levels of silica in the spent fuel pool with analysis using a predictive code (e.g., RACKLIFE) to estimate boron loss from Boraflex panels; and (2) neutron attenuation testing to measure the boron areal density of the short-length test coupons. The Boraflex Monitoring Program is based on existing technology and methods for testing and evaluating material properties necessary to ensure the required 5% margin to criticality in the spent fuel pool is maintained. The Boraflex Monitoring Program takes exception to certain NUREG-1801, Section XI.M22 (Boraflex Monitoring) evaluation elements. Specifically, exception is taken to performing neutron attenuation testing and measurement of boron areal density of the spent fuel pool storage racks directly.

A1.1.6 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program is a new program that will manage the aging effects on the external surfaces of carbon steel, low-alloy steel, and cast iron components (e.g. tanks, piping) that are buried in soil. Program activities will include visual inspections of external coatings and wrappings to detect damage and degradation. Periodicity of inspections will be based on plant operating experience and opportunities for inspection due to maintenance. This program will be implemented prior to the period of extended operation.

A1.1.7 BWR FEEDWATER NOZZLE PROGRAM

The BWR Feedwater Nozzle Program manages cracking of critical regions of the BWR feedwater nozzle. Program activities are implemented as augmented examinations through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The recommendations of General Electric (GE) NE-523-A71-0594 are applicable and certain system modifications have been made to mitigate nozzle cracking. The program includes inspection of the NMP1 feedwater nozzles per Table IWB 2500-1.

A1.1.8 BWR PENETRATIONS PROGRAM

The BWR Penetrations Program manages the effects of cracking in the various penetrations of the reactor pressure vessels at NMPNS. The BWR Penetrations Program is based on guidelines issued by the BWR Vessel and Internals Project (BWRVIP) and approved by the NRC. This program is implemented by the BWR Vessel Internals Program for managing specific aging effects. The attributes of the BWR Penetrations Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A1.1.9 BWR REACTOR WATER CLEANUP SYSTEM PROGRAM

The BWR Reactor Water Cleanup System Program manages the effects of intergranular stress corrosion cracking on the intended function of austenitic stainless steel piping in the reactor water cleanup system. This program is based on industry guidelines approved by the NRC. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The attributes of the BWR Reactor Water Cleanup System Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A1.1.10 BWR STRESS CORROSION CRACKING PROGRAM

The BWR Stress Corrosion Cracking (SCC) Program mitigates intergranular SCC in stainless steel reactor coolant pressure boundary components and piping four inches and greater nominal pipe size. This program is based on industry guidelines approved by the NRC. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The attributes of the BWR SCC Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A1.1.11 BWR VESSEL ID ATTACHMENT WELDS PROGRAM

The BWR Vessel ID Attachment Welds Program manages the effects of cracking in reactor pressure vessel inside diameter attachment welds. This program is based on industry guidelines issued by the BWRVIP and approved by the NRC. The BWR Vessel ID Attachment Welds Program is implemented by the BWR Vessel Internals Program for managing specific aging effects. The attributes of the BWR Vessel ID Attachment Welds Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A1.1.12 BWR VESSEL INTERNALS PROGRAM

The BWR Vessel Internals Program manages aging of materials inside the reactor vessel. Program activities include (1) inspections for the presence and effects of cracking; and (2) monitoring and control of water chemistry. This program is based on guidelines issued by the BWRVIP and approved (or pending approval[1]) by the NRC. Relevant BWRVIP-required inspections have been implemented as augmented examinations through the Inservice Inspection Program. The attributes of the BWR Vessel Internals Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A1.1.13 CLOSED-CYCLE COOLING WATER SYSTEM PROGRAM

The Closed-Cycle Cooling Water System (CCCWS) Program manages loss of material and fouling of components exposed to closed-cycle cooling water environments. The applicable piping systems include the Reactor Building Closed Loop Cooling System, Control Room HVAC System, the heat exchanger jacket water cooling portions of the Emergency Diesel Generator System, and a portion of the Turbine Building Closed Loop Cooling System. Program activities include chemistry monitoring, surveillance testing, and component inspections. The CCCWS Program implements the guidelines for controlling system performance and aging effects described in Electric Power Research Institute (EPRI) Report TR-107396.

The CCCWS Program takes exceptions to NUREG-1801, Section XI.M21 (Closed-Cycle Cooling Water System) evaluation elements. Specifically, corrosion inhibitors are not part of the system chemistry for the applicable NMP1 closed cycle cooling water systems. However, pure demineralized water chemistry is addressed by EPRI TR-107396, and the jacket water cooling portion of the Emergency Diesel Generator System utilizes a chromate water treatment regime based on vendor recommendations.

Enhancements to the CCCWS Program include the following revisions to existing activities that are credited for license renewal:

- Include additional information on sampling frequencies for Control Room Chilled Water.
- Direct periodic inspections to monitor for loss of material and fouling in the Reactor Water Cleanup System and Shutdown Cooling heat exchangers.

[1] NRC review of BWRVIP-76 is not yet complete.

- Implement a corrosion monitoring program for larger bore CCCW piping not subject to inspection under another program.

- Establish a five year (minimum) frequency to inspect for degradation of components in CCCW Systems.

The enhancements are scheduled for completion prior to the period of extended operation.

A1.1.14 COMPRESSED AIR MONITORING PROGRAM

The Compressed Air Monitoring Program manages aging effects for portions of the Compressed Air Systems within the scope of license renewal, including cracking and loss of material due to general corrosion, by controlling the internal environment of systems and components. Program activities include air quality checks at various locations to detect contaminants that would affect the system's intended function. Additional visual inspections are credited for identification and monitoring of degradation for air compressors, receivers, and air dryers. The Compressed Air Monitoring Program is based on Generic Letter (GL) 88-14 and recommendations presented in INPO Significant Operating Event Report 88-01.

Enhancements to the Compressed Air Monitoring Program include the following revisions to existing activities that are credited for license renewal:

- Develop new activities to manage the loss of material, stress corrosion cracking, and perform periodic system leak checks.

- Expand the scope, periodicity, and inspection techniques to ensure that the aging of certain sub-components of the dryers and compressors (e.g., valves, heat exchangers) is managed.

- Establish activities that manage the aging of the internal surfaces of carbon steel piping and that require system leak checks to detect deterioration of the pressure boundaries.

- Expand the acceptance criteria to ensure that the aging of certain sub-components of the dryers and compressors (e.g., valves, heat exchangers) is managed.

The enhancements are scheduled for completion prior to the period of extended operation.

A1.1.15 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification (EQ) Program manages thermal, radiation, and cyclical aging for electrical equipment important to safety and located in harsh plant environments at NMPNS. Program activities (1) identify applicable equipment and environmental requirements; (2) establish, demonstrate, and document the level of qualification (including configuration, maintenance, surveillance, and replacement requirements); and (3) maintain (or preserve) qualification. The EQ Program employs aging evaluations based on 10 CFR 50.49(f) qualification methods. Components in the EQ Program must be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

A1.1.16 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program (FMP) is an existing program that manages the fatigue life of reactor coolant pressure boundary components by tracking and evaluating key plant events. This program monitors operating transients to date, calculates fatigue usage factors to date, and permits implementation of corrective measures in order not to exceed the design limit on fatigue usage.

The FMP will be enhanced with guidance for the use of the FatiguePro software package and updated methodology for environmental fatigue factors in establishing updated fatigue life calculations for components. The enhancement is scheduled for completion prior to the period of extended operation.

A1.1.17 FIRE PROTECTION PROGRAM

The Fire Protection Program provides guidance for performance of periodic visual inspections to manage aging of the various materials comprising rated fire barriers. These include (a) sealants in rated penetration seals (subject to shrinkage due to weathering); (b) concrete and steel in fire rated walls, ceilings, and floors (subject to loss of material due to flaking and abrasion; separation and concrete damage due to relative motion, vibration, and shrinkage); and (c) steel in rated fire doors (subject to loss of material due to corrosion and wear or mechanical damage). In addition this program requires testing of the diesel-driven fire pump to verify that it is performing its intended function. This activity manages aging of the fuel oil supply line to the diesel engine, which may experience loss of material due to corrosion. Inspection and testing is performed in accordance with the guidance of applicable standards.

A1.1.18 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program manages aging of water-based fire protection systems due to loss of material and biofouling. Program activities include periodic maintenance, testing, and inspection of system piping and components containing water (e.g., sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes). Inspection and testing is performed in accordance with the guidance of applicable National Fire Protection Association (NFPA) Codes and Standards and the Nuclear Electric Insurance Limited (NEIL) Members' Manual.

Enhancements to the Fire Water System Program include the following revisions to existing activities that are credited for license renewal:

- Incorporate inspections to detect and manage loss of material due to corrosion into existing periodic test procedures.
- Specify periodic component inspections to verify that loss of material is being managed.
- Add procedural guidance for performing visual inspections to monitor internal corrosion and detect biofouling.
- Add requirements to periodically check the water-based fire protection systems for microbiological contamination.
- Measure fire protection system piping wall thickness using non-intrusive techniques (e.g., volumetric testing) to detect loss of material due to corrosion.
- Establish an appropriate means of recording, evaluating, reviewing, and trending the results of visual inspections and volumetric testing.
- Define acceptance criteria for visual inspections and volumetric testing.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.19 FLOW-ACCELERATED CORROSION PROGRAM

The Flow-Accelerated Corrosion (FAC) Program (also referred to as the Erosion/Corrosion Program) manages aging effects due to flow-accelerated corrosion in carbon steel and low alloy steel piping containing single-phase and two-phase high-energy fluids. Program activities include (1) analysis using a predictive code (CHECWORKS) to determine critical locations, (2) baseline inspections to determine the extent of thinning at the selected locations, (3) follow-up inspections to confirm the predictions, and (4) repair or replacement of components, as necessary. The program considers the recommended actions in NRC Bulletin 87-01 and Information Notice 91-18, and implements the guidelines for an effective FAC Program presented in EPRI Report NSAC-202L-R2. The program also implements the recommendations provided in GL 89-08, *Erosion/Corrosion Induced Pipe Wall Thinning*.

A1.1.20 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program manages loss of material due to corrosion that may result from introduction of contaminants into the plant's fuel oil storage tanks. Program activities include (1) sampling and chemical analysis of the fuel oil inventory at the plant, (2) sampling, testing, and analysis of new fuel oil as it is unloaded at the plant, and (3) cleaning and inspection of fuel oil storage tanks. This program is based on maintaining fuel oil quality in accordance with the guidelines of applicable American Society for Testing Materials (ASTM) Standards.

Enhancements to the Fuel Oil Chemistry Program include the following revisions to existing activities that are credited for license renewal:

- Add requirements to periodically check diesel fuel oil for microbiological organisms.
- Add requirements to periodically inspect fuel oil storage tanks for evidence of significant degradation, including a requirement that the tank thickness be determined.
- Provide guidelines for the appropriate use of biocides, corrosion inhibitors, and fuel stabilizers to maintain fuel oil quality. Additionally, specify that water be removed from tanks if found during routine sampling.
- Incorporate periodic tests for microbiological organisms into Fuel Oil Chemistry Program procedures.

- Include evaluation of microbiological organisms in fuel oil as a measure of contamination.

- Add a requirement for multilevel sampling.

- Establish a quarterly (minimum) frequency to check fuel oil for microbiological organisms, and a requirement for quarterly trending of analysis results.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.21 FUSE HOLDER INSPECTION PROGRAM

The Fuse Holder Inspection Program is a new plant-specific program that applies to fuse holders located outside of active devices that have aging effects requiring management. This program requires testing to detect deterioration of metallic clamps that would affect the ability of in-scope fuse holders to perform their intended function. The Fuse Holder Inspection Program includes the following aging stressors: moisture, fatigue, ohmic heating, mechanical stress, vibration, thermal cycling, electrical transients, chemical contamination, oxidation, and corrosion.

Analytical trending will not be included in this activity because the parameters monitored may vary depending upon the test method selected. This is an exception to the “Monitoring and Trending” element in Appendix A.1.2.3.5 to NUREG-1800. This program will be implemented prior to the period of extended operation.

A1.1.22 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD HANDLING SYSTEMS PROGRAM

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program (referred to herein as the Crane Inspection Program) manages loss of material due to corrosion of cranes within scope of license renewal (WSLR). Program activities include (1) performance of various maintenance activities on a specified frequency; and (2) pre-operational inspections of equipment prior to lifting activities. Crane inspection activities are based on the mandatory requirements of applicable industry standards and implement the guidance of NUREG-0612.

The Crane Inspection Program will be enhanced to add specific direction for performance of pre-lift corrosion inspections of certain hoist lifting assembly components. The enhancement is scheduled for completion prior to the period of extended operation.

A1.1.23 MASONRY WALL PROGRAM

The Masonry Wall Program manages aging effects so that the evaluation basis established for each masonry wall WSLR remains valid through the period of extended operation. The Masonry Wall Program is based on the structures monitoring requirements of 10 CFR 50.65. The Masonry Wall Program is implemented by the Structures Monitoring Program for managing specific aging effects.

A1.1.24 NON-EQ ELECTRICAL CABLES AND CONNECTIONS PROGRAM

The Non-EQ Electrical Cables and Connections Program is a new program that manages aging of cables and connectors WSLR exposed to adverse localized temperature, moisture, or radiation environments. Program activities include periodic visual inspection of susceptible cables for evidence of cable and connection jacket surface anomalies. This program will be implemented prior to the period of extended operation.

A1.1.25 NON-EQ ELECTRICAL CABLES USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Non-EQ Electrical Cables Used in Instrumentation Circuits Program manages aging of cables exposed to adverse localized temperature and radiation environments that could result in loss of insulation resistance. It applies to accessible and inaccessible electrical cables that are not in the EQ Program and are used in circuits with sensitive, high-voltage, low-level signals such as radiation monitoring, nuclear instrumentation, and other non-nuclear instrumentation that are WSLR. Activities include routine calibration tests of instrumentation loops and are implemented through the Surveillance Testing and Preventive Maintenance Programs. Testing is based on requirements of the plant technical specifications and implemented through the work control system.

Enhancements to the Non-EQ Electrical Cables Used in Instrumentation Circuits Program include the following revisions to existing activities that are credited for license renewal:

- Implement reviews of cable calibration or surveillance data for indications of aging degradation affecting instrument circuit performance. The first reviews will be completed prior to the period of extended operation and every ten years thereafter.

- In cases where a calibration or surveillance program does not include the cabling system in the testing circuit, or as an alternative to the review of calibration results described above, provide requirements and procedures to perform cable testing to detect deterioration of the insulation system, such as insulation resistance tests or other testing judged to be effective in determining cable insulation condition. The first test will be completed prior to the period of extended operation and every 10 years thereafter.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.26 NON-EQ INACCESSIBLE MEDIUM VOLTAGE CABLES PROGRAM

The Non-EQ Inaccessible Medium Voltage Cable Inspection Program is a new program that provides reasonable assurance that the intended functions of inaccessible medium voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49, and are potentially exposed to adverse localized environments caused by moisture and significant voltage, are maintained consistent with the current licensing basis through the period of extended operation. Program activities include visual inspections of accessible areas of ducts and banks via manholes or inspection covers, and testing if warranted by an engineering evaluation. The program considers the technical information and guidance provided in applicable industry publications. This program will be implemented prior to the period of extended operation.

A1.1.27 NON-SEGREGATED BUS INSPECTION PROGRAM

The Non-Segregated Bus Inspection Program is a new plant-specific program that manages aging effects for components and materials internal to the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V buses required for the recovery of offsite power following a Station Blackout (SBO) event. These normally-energized components are not subject to the environmental qualification requirements of 10 CFR 50.49, but can be affected by elevated temperatures prior to the end of the period of extended operation. Program activities include visual inspection of normally-energized non-segregated bus duct internal components for surface anomalies, such as embrittlement, discoloration, cracking, chipping, or surface contamination, which could affect the life of the component. The program considers the technical information and guidance provided in applicable industry publications.

Analytical trending will not be included in this activity because the parameters inspected are not readily quantifiable in an appropriate form. This is an exception to the “Monitoring and Trending” element in Appendix A.1.2.3.5 to NUREG-1800. This program will be implemented prior to the period of extended operation.

A1.1.28 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program is a new program that manages aging effects with potentially long incubation periods for susceptible components WSLR. Program activities include visual, volumetric, and other established inspection techniques consistent with industry practice to provide a means of verifying that an aging effect is either (1) not occurring, or (2) progressing so slowly that it has a negligible effect on the intended function of the structure or component. The program also provides measures for verifying the effectiveness of existing aging management programs. This program is a new program that will be implemented prior to the period of extended operation.

A1.1.29 OPEN-CYCLE COOLING WATER SYSTEM PROGRAM

The Open-Cycle Cooling Water System Program manages aging of components exposed to raw, untreated (e.g., service) water. Program activities include (a) surveillance and control of biofouling (including biocide injection), (b) verification of heat transfer capabilities for components cooled by the Service Water System, (c) inspection and maintenance, (d) walkdown inspections, and (e) review of maintenance, operating and training practices and procedures. Inspections may include visual, Ultrasonic Testing (UT), and Eddy Current Testing (ECT) methods. This program is based on the recommendations of GL 89-13.

A1.1.30 PREVENTIVE MAINTENANCE PROGRAM

The scope of the Preventive Maintenance (PM) Program includes, but is not limited to, valve bodies, heat exchangers, expansion joints, tanks, ductwork, fan/blower housings, dampers, and pump casings. This program provides for performance of various maintenance activities on a specified frequency based on vendor recommendations and operating experience. These activities provide opportunities for component condition monitoring to manage the effects of aging for many SSCs WSLR.

Enhancements to the PM Program include the following revisions to existing activities that are credited for license renewal:

- Expand the PM Program to encompass activities for certain additional components identified in the license renewal aging management reviews.
- Explicitly define the aging management attributes, including the systems and the component types/commodities included in the program.
- Specifically list activities credited for aging management, parameters monitored, and the aging effects detected.
- Establish a requirement that inspection data be monitored and trended.
- Establish detailed parameter-specific acceptance criteria.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.31 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program manages cracking of and loss of material from the reactor pressure vessel closure studs. This program implements guidance provided in Regulatory Guide 1.65. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda).

A1.1.32 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is an existing program that manages loss of fracture toughness due to neutron irradiation embrittlement in the Reactor Pressure Vessel (RPV) beltline material. Program activities include (1) periodic withdrawal and testing of surveillance capsules from the RPV; (2) use of test results and allowable stress loadings for the ferritic RPV materials to determine operating limits; and (3) comparison with a large industry data set to confirm validity of test results. Analysis and testing are based on the requirements of 10 CFR 50, Appendix H, and ASTM Standard

E-185. NMPNS has committed to implement the Integrated Surveillance Program (ISP) described in BWRVIP-116 (currently under review by the NRC staff). Since the proposed ISP for license renewal has no provision to store tested specimens, such storage is considered unnecessary. This is an exception to the program described in NUREG-1801.

Enhancements to the Reactor Vessel Surveillance Program include the following revisions to existing activities that are credited for license renewal:

- Incorporate the requirements and elements of the ISP, published by the BWRVIP, into the Reactor Vessel Surveillance Program.

- Project analyses of upper shelf energy and pressure-temperature limits to 60 years using methods prescribed by Regulatory Guide 1.99, Revision 2, and include the applicable bounds of the data, such as operating temperature and neutron fluence.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.33 SELECTIVE LEACHING OF MATERIALS PROGRAM

The Selective Leaching of Materials Program is a new program that manages aging of components susceptible to selective leaching. The potentially susceptible components include valve bodies, valve bonnets, pump casings, and heat exchanger components in various systems. This program will be implemented through the One Time Inspection Program prior to the period of extended operation.

A1.1.34 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program manages aging of structures, structural components, and structural supports WSLR. The program provides for periodic visual inspections, surveys, and examination of all safety related buildings (including the primary containment and substructures within the primary containment) and various other buildings WSLR. Program activities identify degradation of materials of construction, which include structural steel, concrete, masonry block, sealing materials. While not credited for mitigation of aging, protective coatings are also inspected under this program. The Structures Monitoring Program, which was initially developed to meet the regulatory requirements of 10 CFR 50.65, implements guidance provided in Regulatory Guide 1.160, NUMARC 93-01 and NEI 96-03.

Enhancements to the Structures Monitoring Program include the following revisions to existing activities that are credited for license renewal:

- Expand the parameters monitored during structural inspections to include those relevant to aging effects requiring management identified for structural bolting.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.35 SYSTEMS WALKDOWN PROGRAM

The Systems Walkdown Program manages aging effects for accessible external surfaces. The specific aging effect of concern is loss of material from external surfaces of pumps, valves, piping, bolts, heat exchangers, tanks, expansion joints, electrical penetrations, electrical enclosures and cabinets, HVAC components, and other carbon steel components. Program activities include system engineer walkdowns (i.e., field evaluations of system components to assess system performance and material condition), evaluation of inspection results, and appropriate corrective actions.

Enhancements to the Systems Walkdown Program include the following revisions to existing activities that are credited for license renewal:

- Explicitly state the aging management attributes, including the systems and component types/commodities included in the program.

- Specifically list parameters monitored, and provide guidance for assessment of identified deterioration. This will include extent of corrosion (loss of material), condition of coatings (material degradation), leakage or indications of leakage, etc.

- Components WSLR included in the Systems Walkdown Program will be visually inspected for loss of material, material degradation and leakage per the upgraded program documentation. The frequency of inspections will be at least once per refuel cycle for each structure and system.

- Define a methodology that specifies consistent criteria for program data collection.

- Specify the parameters and data that will be monitored and trended.

- Specify acceptance criteria for visual inspections and guidelines for using the criteria to assess observed material condition against intended function for inspected SSCs.

- Upgrade program documents so that they fall under the administrative controls program.

- Include previous operating experience and ensure future operating experience is properly incorporated.

Enhancements are scheduled for completion prior to the period of extended operation.

A1.1.36 TORUS CORROSION MONITORING PROGRAM

The Torus Corrosion Monitoring Program manages corrosion of the NMP1 suppression chamber (torus) through inspection and analysis. This program provides for (1) determination of torus shell thickness through ultrasonic measurement; (2) determination of corrosion rate through analysis of material coupons; and (3) visual inspection of accessible external surfaces of the torus support structure for corrosion. The Torus Corrosion Monitoring Program ensures that the NMP1 torus shell and support structure thickness limits are not exceeded.

A1.1.37 WATER CHEMISTRY CONTROL PROGRAM

The Water Chemistry Control Program manages aging effects by controlling the internal environment of the reactor water, feedwater, condensate, and control rod drive systems, and related auxiliaries (such as the torus, condensate storage tank, and spent fuel pool). The aging effects/mechanisms of concern include (1) stress corrosion cracking; (2) loss of material due to corrosion; and (3) fouling. Program activities include monitoring and controlling concentrations of known detrimental chemical species below the levels known to cause degradation. The Water Chemistry Control Program implements the guidelines for BWR water chemistry presented in EPRI Reports TR-103515-R1 and TR-103515-R2. This is an exception to the program described in NUREG-1801 (which identifies the February 1994 version of BWRVIP-29 as the basis for BWR water chemistry programs).

A1.2 TIME-LIMITED AGING ANALYSIS SUMMARIES

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.

A1.2.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. The evaluation of reactor vessel neutron embrittlement is a TLAA. The following TLAA discussions are related to the issue of neutron embrittlement:

- Upper Shelf Energy
- Pressure-Temperature (P-T) Limits
- Elimination of Circumferential Weld Inspection
- Axial Weld Failure Probability

A1.2.1.1 UPPER-SHELF ENERGY

Ferritic Reactor Pressure Vessel (RPV) materials undergo a transition in fracture behavior from brittle to ductile as the temperature of the material is increased. Charpy V-notch tests are conducted in the nuclear industry to monitor changes in the fracture behavior during irradiation. Neutron irradiation to fluences above approximately $1x10^{17}$ n/cm^2 causes an upward shift in the ductile-to-brittle transition temperature and a drop in upper-shelf energy (USE). To satisfy the acceptance criteria for USE contained in 10 CFR 50 Appendix G, the RPV beltline materials must have a Charpy USE of no less than 50 ft-lbs throughout the life of the RPV unless it can be demonstrated that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The USE for the limiting beltline weld materials for NMP1 is predicted to remain above 50 ft-lbs throughout the period of extended operation, based on projected fluence values. The USE of the limiting plate material for NMP1 is below 50 ft-lbs but is predicted to remain above the value required by an equivalent margins analysis, based on projected fluence values. Therefore, the USE for the NMP1 RPV beltline materials has been projected (reevaluated) for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A1.2.1.2 PRESSURE - TEMPERATURE (P – T) LIMITS

10 CFR 50 Appendix G requires that the RPV be operated within established pressure-temperature (P-T) limits during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. NMP1 Technical Specifications contain P-T limit curves for heatup, cooldown, inservice leakage testing, and hydrostatic testing, and limit the maximum rate of change of reactor coolant temperature.

The P-T limit curves are periodically revised to account for changes in fracture toughness of the RPV components due to anticipated neutron embrittlement effects for higher accumulated fluences. Calculation of P-T limit curves using the projected fluence at the end of the period of extended operation would result in unnecessarily restrictive operating curves. However, projection of the Adjusted Reference Temperature (ART), which is used in development of the curves, to the end of the period of extended operation provides assurance that development of P-T limit curves will be feasible up to the maximum predicted effective full power years (EFPY).

Projections of the ART values for the beltline materials have been made for the period of extended operation, providing reasonable assurance that it will be possible to prepare P-T curves that will permit continued plant operation. The P-T curves (and the related Technical Specifications) will continue to be updated either as required by 10 CFR 50, Appendix G, to assure the operational limits remain valid at the current cumulative neutron fluence levels, or on an as-needed basis to provide appropriate operational flexibility.

A1.2.1.3 ELIMINATION OF CIRCUMFERENTIAL WELD INSPECTION

Relief from reactor vessel circumferential weld examination requirements under GL 98-05, *Boiling Water Reactor Licensees Use of the BWRVIP-05 Report To Request Relief From Augmented Examination Requirements On Reactor Pressure Vessel Circumferential Shell Welds*, is based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period. NMP1 has received relief from reactor vessel circumferential weld examination requirements under GL 98-05, for the remainder of its current 40-year license term (Reference A1.3.1).

Projected values of mean and upper bound reference temperature nil ductility transition temperature (RT_{NDT}) for the limiting circumferential welds at NMP1 are below the bounding mean RT_{NDT} determined by the NRC staff in the SER for BWRVIP-05 (Reference A1.3.7). Thus, there is reasonable assurance the conditional probability of vessel failure due to NMP1 RPV circumferential weld failure is bounded by the NRC analysis.

NMP1 will apply for relief from circumferential weld inspections for the period of extended operation. Supporting analyses, procedural controls, and operator training will be completed prior to the period of extended operation to support and confirm that the RPV circumferential weld failure probability remains acceptable for the period of extended operation. Based on the scoping evaluation discussed above, there is reasonable assurance the failure probability will remain acceptable for the period of extended operation.

A1.2.1.4 AXIAL WELD FAILURE PROBABILITY

In the safety evaluation presented in *Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report* (Reference A1.3.8), the NRC staff indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than $5x10^{-6}$ per reactor year, given the assumptions on flaw density, distribution, and location described in the SER. Projected values of mean RT_{NDT} and upper bound RT_{NDT} for the limiting axial welds at NMP1 are below the bounding mean RT_{NDT} value determined by the NRC staff in the SER for BWRVIP-74-A (Reference A1.3.2). Thus, there is reasonable assurance that the RPV failure frequency due to failure of the limiting axial weld is expected to remain less than $5x10^{-6}$ per reactor year for NMP1.

Inspection of the axial welds in accordance with the ASME XI code requirements will continue at NMP1 during the period of extended operation. Supporting analyses will be completed prior to the period of extended operation to confirm that the RPV axial weld failure probability for the limiting NMP1 axial weld remains bounded for the period of extended operation. Based on the scoping evaluation discussed above, there is reasonable assurance the failure probability will remain acceptable for the period of extended operation.

A1.2.2 METAL FATIGUE ANALYSIS

ASME Section III requires calculation of cumulative usage factors (CUFs) to demonstrate fatigue-tolerant design for reactor vessels, vessel internals, Class 1 piping and components, metal containments, and penetrations. These values are indexed to the number of transients anticipated over the design life of the component (usually 40 years).

Designated plant events have been counted and categorized to ensure that the number of actual operational transient cycles does not exceed the number of transients assumed in the plant design for fatigue. For certain events that affect fatigue usage, linear projections of the actual data to the end of the period of extended operation will exceed the analyzed number of design basis transients. For those locations where additional fatigue analysis is required to take advantage of the implicit margin (and to more accurately determine CUFs), the EPRI FatiguePro fatigue monitoring software will be implemented.

The following thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs:

- Reactor Vessel Fatigue Analysis
- Feedwater (FWS) Nozzle and Control Rod Drive Return Line (CRDRL) Nozzle Fatigue and Cracking Analyses
- Non-ASME Section III Class 1 Piping and Components Fatigue Analysis
- Reactor Vessel Internals Fatigue Analysis
- Environmentally Assisted Fatigue
- Fatigue of the Emergency Condenser

A1.2.2.1 REACTOR VESSEL FATIGUE ANALYSIS

The original design of RPV pressure boundary components included analyses of fatigue resistance. Components were evaluated by calculating the alternating stresses associated with applicable design transients and determining a CUF based on the number of anticipated transients for the original 40-year life of the plant. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

For the critical RPV component locations, transients contributing to fatigue usage will be tracked by the Fatigue Monitoring Program (FMP) (Appendix A1.1.16) with additional usage added to the baseline CUF. The FMP provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

A1.2.2.2 FEEDWATER (FWS) NOZZLE AND CONTROL ROD DRIVE RETURN LINE (CRDRL) NOZZLE FATIGUE AND CRACKING ANALYSES

BWRs have experienced fatigue crack initiation and growth in Feedwater System (FWS) and Control Rod Drive Return Line (CRDRL) nozzles. Rapid thermal cycling (occurring as a result of bypass leakage past loose-fitting thermal sleeves, or in nozzles lacking thermal sleeves) initiated fatigue cracks that propagated due to larger (in terms of the magnitude of temperature and pressure change) thermal cycles resulting from plant transients. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking* identifies interim and long-term procedural and design changes to minimize thermal fatigue cracking, as well as inspection requirements.

Various calculations were prepared in response to NUREG-0619 (e.g., to support enhanced inspection intervals, to incorporate updated fatigue crack growth curves, etc.), and CUFs were determined on the basis of anticipated transients for the original 40-year life of the plant. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

The NMP1 FWS nozzles require continued monitoring (including analysis using FatiguePro) to demonstrate compliance over the period of extended operation. Transients contributing to fatigue usage of the FWS nozzles will be tracked by the FMP (Appendix A1.1.16) with additional usage added to the baseline CUF. Additionally, the NMP1 FWS nozzles will be periodically inspected in accordance with NMP1 commitments related to NUREG-0619. The fatigue usage of the NMP1 CRDRL nozzle has been calculated to be significantly below the allowable fatigue usage of 1.0 over the life of the plant, including a 20-year license extension. However, NMP1 will continue to perform enhanced inspections of the CRDRL nozzle in accordance with NMP1 commitments to NUREG-0619.

A1.2.2.3 NON-ASME SECTION III CLASS 1 PIPING AND COMPONENTS FATIGUE ANALYSIS

Piping and components WSLR were designed to codes other than ASME Section III Class 1. Applicable codes include ASA B31.1-1955 and ASME Section III Class 2 or 3. These codes do not require explicit fatigue analyses. Instead, the effects of cyclic loading are accounted for through application of stress range reduction factors based on the anticipated number of equivalent full temperature thermal expansion cycles over the original 40-year life of the plant.

The original design for cyclic loading is expected to remain valid for the period of extended operation for the majority of Non-ASME Class 1 systems and components. However, Non-ASME Class 1 locations meeting one or more of the following criteria, require development of fatigue analyses (similar to those performed for ASME Class 1 piping):

- The location experiences high fatigue usage due to significant thermal transients due primarily to on/off flow, stratification, and local thermal cycling effects;
- The location experiences high fatigue usage due to structural or material discontinuities that result in high stress indices (e.g., at thickness transitions);
- The location has been identified in NUREG/CR-6260 (Reference A1.3.3) for the older-vintage BWRs (i.e., locations equivalent to the recirculation line at the RHR return line tee, the RHR line at the tapered transition, and the feedwater line at the RCIC tee).

Based on the above criteria, portions of the following NMP1 systems were identified for further analysis:

- Feedwater/High Pressure Coolant Injection System,
- Core Spray System,
- Reactor Water Cleanup System (piping inside the reactor coolant pressure boundary), and

- Reactor Recirculation System (and associated Shutdown Cooling System lines).

Prior to the period of extended operation, a baseline CUF (based on a conservative analysis of the fatigue usage to-date) will be determined for the specified portions of the NMP1 systems listed above. If the baseline CUF for a specified portion of a system exceeds 0.4 (considered a general threshold of significance), the limiting location may require monitoring to demonstrate compliance over the period of extended operation. For the limiting locations, those transients contributing to fatigue usage will be tracked by the FMP with additional usage added to the baseline CUF.

A1.2.2.4 REACTOR VESSEL INTERNALS FATIGUE ANALYSIS

Determination of CUFs was not a design requirement for reactor vessel internals at NMP1. However, mechanical clamps installed as a repair for cracked vertical core shroud welds were evaluated for fatigue using ASME Section III methods to calculate alternating stresses and determine CUF values. Fatigue-tolerant design is demonstrated for the mechancial clamps with CUFs less than 1.0.

The potential for cracking of components comprising the reactor vessel internals, both due to fatigue and (more significantly) intergranular stress corrosion cracking (IGSCC), is managed by the BWR Vessel Internals Program (Appendix A1.1.12), which incorporates comprehensive inspection and evaluation guidelines issued by the BWRVIP and approved by the NRC. These activities provide assurance that any unexpected degradation resulting from fatigue in the reactor vessel internals for the current license period and the period of extended operation will be identified and corrected. Therefore, the effects of fatigue on the intended function(s) of the reactor vessel internals will be adequately managed for the period of extended operation.

A1.2.2.5 ENVIRONMENTALLY ASSISTED FATIGUE

Generic Safety Issue (GSI) 190, *Fatigue Evaluation of Metal Components for 60-year Plant Life*, was established to address NRC concerns regarding environmental effects on fatigue of pressure boundary components for 60 years of plant operation. The NRC staff studied the probability of fatigue failure for selected metal components based on the increased CUFs determined in NUREG/CR-6260 (Reference A1.3.3) and a 60-year plant life. The NRC closed this GSI, and concluded that environmental effects did not substantially affect core damage frequency. However, since the nature of age-related degradation indicated the potential for an increase in the frequency of pipe leaks as plants continue to operate, licensees are required to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

NMP1 will assess the impact of the reactor coolant environment on a sample of critical component locations, including locations equivalent to those identified in NUREG/CR-6260 as part of the FMP (Appendix A1.1.16). These locations will be evaluated by applying environmental correction factors (F_{en}) to existing and future fatigue analyses. Evaluation of the sample of critical components will be completed prior to the period of extended operation.

A1.2.2.6 FATIGUE OF THE EMERGENCY CONDENSER

The Emergency Cooling System (ECS) provides for decay heat removal from the reactor fuel in the event that reactor feedwater capability is lost and the main condenser is unavailable. The tube and shell sides of the emergency condensers were designed in accordance with ASME Section III Class 2 and 3, respectively. The original tubing has experienced thermal fatigue resulting from leakage past the condensate return valve to the RPV. As part of the subsequent modification and repair, fatigue loading was evaluated by calculating the alternating stresses associated with applicable design transients and determining a CUF based on the number of anticipated transients for the life of the condensers. Fatigue-tolerant design is demonstrated for components with CUFs less than 1.0.

While the CUFs were shown to be less than 1.0, certain locations in the NMP1 emergency condensers require continued monitoring (including analysis using FatiguePro) to demonstrate compliance over the period of extended operation. The FMP (Appendix A1.1.16) will track transients specific to the ECS with additional usage added to the baseline CUF for the condensers.

A1.2.3 ENVIRONMENTAL QUALIFICATION (EQ)

The following EQ analysis has been identified as a TLAA:

- Electrical Equipment EQ

A1.2.3.1 ELECTRICAL EQUIPMENT EQ

10 CFR 50.49 requires that certain safety related and non-safety related electrical equipment remain functional during and after identified Design Basis Events. To establish reasonable assurance that this equipment can function when exposed to postulated harsh environmental conditions, licensees are required to determine the equipment's qualified life and to develop a program that maintains the qualification of that equipment.

For components within the scope of the EQ Program (Appendix A1.1.15), analyses of thermal exposure, radiation exposure, and mechanical cycle aging that cannot be shown to remain valid for the period of extended operation will be projected to extend the qualification of components before reaching the aging limits established in the applicable evaluation, or the components will be refurbished or replaced.

A1.2.4 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

The following containment liner plate, metal containments, and penetrations fatigue analyses have been identified as TLAAs:

- Torus Shell and Vent System Fatigue Analysis
- Torus Attached Piping Analysis
- Torus Wall Thickness

A1.2.4.1 TORUS SHELL AND VENT SYSTEM FATIGUE ANALYSIS

Large-scale testing of the Mark III containment and in-plant testing of Mark I primary containment systems identified additional hydrodynamic loads that were not considered in the original design of the Mark I containment used at NMP1. To provide the bases for generic load definition and structural assessment techniques, GE initiated the Mark I Containment Program. NUREG-0661, *Safety Evaluation Report, Mark I Containment Long Term*

Program, *Resolution of Generic Technical Activity A*-7, requires a plant-unique analysis for each Mark I configuration to evaluate the effects of the hydrodynamic stresses resulting from a loss of coolant accident (LOCA) and safety relief valve (SRV) discharge.

The 60-year CUF values for the controlling locations in the torus shell are less than 1.0. Therefore, the NMP1 torus shell has been evaluated and is qualified for the period of extended operation.

A1.2.4.2 TORUS ATTACHED PIPING ANALYSIS

As a result of the Mark I Containment Program, modifications were performed at NMP1, including changes to the configuration of SRV piping and other piping penetrating the suppression chamber (torus) (generically referred to herein as torus-attached piping). As part of the generic Mark I Containment Program, fatigue analyses were performed considering the design loads identified in NUREG-0661 and its supplements. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

The bounding 40-year CUFs for the subject piping and associated penetrations are less than 0.5; therefore, the 60-year CUF values for all controlling locations can be demonstrated to remain less than 1.0. Therefore, the NMP1 torus-attached piping has been evaluated and is qualified for the period of extended operation.

A1.2.4.3 TORUS WALL THICKNESS

The NMP1 suppression chamber (torus) is constructed of A201 Grade B (Firebox) steel plates with a certified minimum thickness of 0.460 inches. This value included an original corrosion allowance of 0.0625 inches, which was added to the minimum wall thickness required by the applicable design codes. However, subsequent addition of hydrodynamic loads (resulting from LOCA and safety relief valve actuation) to the containment design bases resulted in a reduction of the corrosion allowance. To establish reasonable assurance that the revised minimum wall thickness of 0.431 inches is not reached, NMP1 is required to monitor torus wall thickness and corrosion rate (Reference A1.3.4). Determination of torus corrosion rates is an ongoing activity that considers inspection results and the remaining corrosion allowance.

The NMP1 Torus Corrosion Monitoring Program (Appendix A1.1.36) has been developed to monitor the torus shell material thickness and ensure it is maintained within the bounds of the qualification bases. Therefore, the effects of loss of material on the intended function(s) of the torus shell will be adequately managed during the period of extended operation.

A1.2.5 OTHER PLANT-SPECIFIC TLAAS

The following Plant-Specific TLAA has been identified for NMP1:

- Reactor Vessel and Reactor Vessel Closure Head Weld Flaw Evaluations

A1.2.5.1 REACTOR VESSEL AND REACTOR VESSEL CLOSURE HEAD WELD FLAW EVALUATIONS

During RFO15, augmented examinations identified unacceptable flaw indications in two RPV shell welds (Reference A1.3.5). During RFO17, UT examinations identified an unacceptable flaw indication in a closure head meridional weld (Reference A1.3.6). Structural evaluations of these flaws (performed in accordance with ASME Section XI, Subsection IWB-3600) compared the flaw characteristics to pre-determined acceptability criteria to justify continued operation without repair of the flaw. Since the acceptability criteria were based on an assumed number of transient cycles applicable to the original 40-year license term, the subject evaluations satisfy the criteria of 10 CFR 54.3(a).

The number of cycles from the time of inspection to the end of the evaluation period is used to determine crack growth. With the addition of the period of extended operation (20 years), the NMP1 RPV can be expected to accumulate fatigue usage for no more than 25 additional years. During this interval, it is unlikely that 240 additional startup/shutdown cycles will occur. Therefore, the RPV closure head weld flaw evaluation remains valid for the period of extended operation.

Prior to the period of extended operation, the RPV weld flaw evaluation will be revised to consider additional fatigue crack growth and the effects of additional irradiation embrittlement associated with operation for an additional 20 years. The flaws will be reexamined in accordance with ASME Section XI as necessary.

A1.3 REFERENCES

A1.3.1 Letter from U.S. Nuclear Regulatory Commission to Niagara Mohawk Power Corporation dated April 7, 1999, *Subject: Alternatives for Examination of Reactor Pressure Vessel Shell Welds, Nine Mile Point Nuclear Station, Unit 1 (TAC No. MA4383).*

A1.3.2 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated October 18, 2001, *Subject: Acceptance for Referencing of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74-A)" and Appendix A, "Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10CFR54.21)"*.

A1.3.3 NUREG/CR-6260, INEL-95/0045, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, February 1995.*

A1.3.4 Letter from U.S. Nuclear Regulatory Commission to Niagara Mohawk Power Corporation dated August 11, 1994, *Subject: Approval of Reduction Factors for Condensation Oscillation Loads in Nine Mile Point Nuclear Station Unit No. 1 (NMP1) Torus (TAC No. M85003)*.

A1.3.5 Letter from Niagara Mohawk Power Corporation (NMP1L 1467) to U.S. Nuclear Regulatory Commission dated September 14, 1999, *Subject: Submittal of 1999 Inservice Inspection Summary Report and Flaw Indication Evaluations.*

A1.3.6 Letter from Nine Mile Point Nuclear Station (NMP1L 1776) to U.S. Nuclear Regulatory Commission dated September 19, 2003, *Subject: Nine Mile Point Unit 1, Docket No. 50-220, Facility Operating License No. DPR-63 – Reactor Pressure Vessel Flaw Evaluation*.

A1.3.7 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated July 28, 1998, *Subject: Final Safety Evaluation of the BWR Vessel and Internal Project BWRVIP-05 Report (TAC No. M93925).*

A1.3.8 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated March 7, 2000, *Subject: Supplement to Final Safety Evaluation of the BWR Vessel and Internal Project BWRVIP-05 Report (TAC No. MA3395).*

A2 APPENDIX A2 – NMP2 UPDATED SAFETY ANALYSIS REPORT (USAR) SUPPLEMENT

A2.1 AGING MANAGEMENT PROGRAMS

A2.1.1 10 CFR 50 APPENDIX J PROGRAM

The 10 CFR 50 Appendix J Program detects degradation of the containment structure and components that comprise the containment pressure boundary, including seals and gaskets. Containment leak rate tests are performed to assure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage limits specified in the Technical Specifications. This program complies with Option B requirements of 10 CFR 50 Appendix J with plant-specific exceptions approved by the NRC as part of license amendments, and implements the guidelines provided in NRC Regulatory Guide (RG) 1.163 and NEI 94-01.

A2.1.2 ASME SECTION XI INSERVICE INSPECTION (SUBSECTION IWE) PROGRAM

The American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection (Subsection IWE) Program (referred to herein as the IWE ISI Program) manages aging effects due to corrosion of carbon steel components comprising the containment pressure boundary. Program activities include visual examination, with limited surface or volumetric examinations when augmented examination is required. The IWE ISI Program is based on the 1998 edition of the ASME Boiler and Pressure Vessel Code, Section XI (Subsection IWE) for containment inservice inspection with plant-specific exceptions approved by the NRC. This is an exception to the evaluation in NUREG-1801 (which covers ASME Section XI requirements from both the 1992 edition with the 1992 addenda and the 1995 edition with the 1996 addenda).

A2.1.3 ASME SECTION XI INSERVICE INSPECTION (SUBSECTION IWF) PROGRAM

The ASME Section XI Inservice Inspection (Subsection IWF) Program (referred to herein as the IWF ISI Program) manages aging of carbon steel component and piping supports due to general corrosion and wear. Program activities include visual examination determine the general mechanical and structural condition of components and their supports. The IWF ISI Program is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code, Section XI (Subsection IWF) for inservice inspection of supports and implements the alternate examination requirements of ASME Code Case N-491-1. This is an exception to the evaluation in NUREG-1801 (which covers ASME Section XI requirements from the 1989 edition through the 1995 edition and addenda through the 1996 addenda).

A2.1.4 ASME SECTION XI INSERVICE INSPECTION (SUBSECTION IWL) PROGRAM

The ASME Section XI Inservice Inspection (Subsection IWL) Program (referred to herein as the IWL ISI Program) manages aging of concrete in the NMP2 containment wall, base mat, and drywell floor. Program activities include general visual examination of all accessible concrete surface areas, with provisions for detailed visual examination when deterioration and distress of suspect areas is detected. The IWL ISI Program is based on the 1998 edition of the ASME Boiler and Pressure Vessel Code, Section XI (Subsection IWL) for containment inservice inspection with plant-specific exceptions approved by the NRC. This is an exception to the evaluation in NUREG-1801 (which covers ASME Section XI requirements from both the 1992 edition with the 1992 addenda and the 1995 edition with the 1996 addenda).

A2.1.5 ASME SECTION XI INSERVICE INSPECTION (SUBSECTIONS IWB, IWC, IWD) PROGRAM

The ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program manages aging of Class 1, 2, or 3 pressure-retaining components and their integral attachments. Program activities include periodic visual, surface, and/or volumetric examination and pressure tests of Class 1, 2 and 3 pressure-retaining components. The ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, Section XI (Subsections IWB, IWC, and IWD) for inservice inspection of pressure-retaining components and their integral attachments, with the risk-informed requirements of ASME Code Case N-578-1 implemented for examination of welds in Class 1 and 2 piping as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda).

A2.1.6 BORAFLEX MONITORING PROGRAM

The Boraflex Monitoring Program is an existing program that manages degradation of neutron absorbing material in spent fuel pool storage racks resulting from radiation exposure and possible water ingress. Program activities include (1) visual inspection of the NMP2 full-length test coupon to detect gap formation; (2) correlation of measured levels of silica in the spent fuel pool with analysis using a predictive code (e.g., RACKLIFE) to estimate boron loss from Boraflex panels; and (3) neutron attenuation testing to mesaure the boron areal density of the short-length test coupons. The Boraflex Monitoring Program is based on existing technology and methods for testing and evaluating material properties necessary to ensure the required 5% margin to criticality in the spent fuel pool is maintained. The Boraflex Monitoring Program takes exception to certain NUREG-1801, Section XI.M22 (Boraflex Monitoring) evaluation elements. Specifically, exception is taken to performing neutron attenuation testing and measurement of boron areal density of the spent fuel pool storage racks directly.

A2.1.7 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program is a new program that will manage the aging effects on the external surfaces of carbon steel, low-alloy steel, and cast iron components (e.g. tanks, piping) that are buried in soil. Program activities will include visual inspections of external coatings and wrappings to detect damage and degradation. Periodicity of inspections will be based on plant operating experience and opportunities for inspection due to maintenance. This program will be implemented prior to the period of extended operation.

A2.1.8 BWR FEEDWATER NOZZLE PROGRAM

The BWR Feedwater Nozzle Program manages cracking of critical regions of the BWR feedwater nozzle. Program activities are implemented as augmented examinations through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The NMP2 feedwater nozzles were redesigned by General Electric (GE) prior to operation and are not susceptible to cracking. The program includes inspection of the NMP2 feedwater nozzles per Table IWB 2500-1.

A2.1.9 BWR PENETRATIONS PROGRAM

The BWR Penetrations Program manages the effects of cracking in the various penetrations of the reactor pressure vessels at NMPNS. The BWR Penetrations Program is based on guidelines issued by the BWR Vessel and Internals Project (BWRVIP) and approved by the NRC. This program is implemented by the BWR Vessel Internals Program for managing specific aging effects. The attributes of the BWR Penetrations Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A2.1.10 BWR REACTOR WATER CLEANUP SYSTEM PROGRAM

The BWR Reactor Water Cleanup System Program manages the effects of intergranular stress corrosion cracking on the intended function of austenitic stainless steel piping in the reactor water cleanup system. This program is based on industry guidelines approved by the NRC. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The attributes of the BWR Reactor Water Cleanup System Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A2.1.11 BWR STRESS CORROSION CRACKING PROGRAM

The BWR Stress Corrosion Cracking (SCC) Program mitigates intergranular SCC in stainless steel reactor coolant pressure boundary components and piping four inches and greater nominal pipe size. This program is based on industry guidelines approved by the NRC. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda). The attributes of the BWR SCC Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A2.1.12 BWR VESSEL ID ATTACHMENT WELDS PROGRAM

The BWR Vessel ID Attachment Welds Program manages the effects of cracking in reactor pressure vessel inside diameter attachment welds. This program is based on industry guidelines issued by the BWRVIP and approved by the NRC. The BWR Vessel ID Attachment Welds Program is implemented by the BWR Vessel Internals Program for managing specific aging effects. The attributes of the BWR Vessel ID Attachment Welds Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A2.1.13 BWR VESSEL INTERNALS PROGRAM

The BWR Vessel Internals Program manages aging of materials inside the reactor vessel. Program activities include (1) inspections for the presence and effects of cracking; and (2) monitoring and control of water chemistry. This program is based on guidelines issued by the BWRVIP and approved (or pending approval[1]) by the NRC. Relevant BWRVIP-required inspections have been implemented as augmented examinations through the Inservice Inspection Program. The attributes of the BWR Vessel Internals Program related to maintaining reactor coolant water chemistry are included in the Water Chemistry Control Program.

A2.1.14 CLOSED-CYCLE COOLING WATER SYSTEM PROGRAM

The Closed-Cycle Cooling Water System (CCCWS) Program manages loss of material and fouling of components exposed to closed-cycle cooling water environments. The applicable piping systems include the Reactor Building Closed Loop Cooling System, Control Building Ventilation Chilled Water System, and the heat exchanger jacket water cooling portion of the Standby Diesel Generator Protection (Generator) System. Program activities include chemistry monitoring, surveillance testing, and component inspections. The CCCWS Program implements the guidelines for controlling system performance and aging effects described in Electric Power Research Institute (EPRI) Report TR-107396.

[1] NRC review of BWRVIP-76 is not yet complete.

The CCCWS Program takes exceptions to NUREG-1801, Section XI.M21 (Closed-Cycle Cooling Water System) evaluation elements. Specifically, corrosion inhibitors are not part of the system chemistry for the applicable NMP2 closed cycle cooling water systems. However, pure demineralized water chemistry is addressed by EPRI TR-107396, and the jacket water cooling portion of Standby Diesel Generator Protection (Generator) System utilizes nitrite water chemistry.

Enhancements to the CCCWS Program include the following revisions to existing activities that are credited for license renewal:

- Direct periodic inspections to monitor for loss of material in the Reactor Building Closed Loop Cooling system piping.
- Establish a five year (minimum) frequency to inspect for degradation of components in the Control Building Chilled Water System.
- Expand periodic checks of NMP2 CCCW Systems consistent with the guidelines of EPRI TR-107396.
- Specify chemistry control parameters for the Control Building Chilled Water System.

The enhancements are scheduled for completion prior to the period of extended operation.

A2.1.15 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification (EQ) Program manages thermal, radiation, and cyclical aging for electrical equipment important to safety and active safety-related mechanical equipment located in harsh plant environments at NMPNS. Program activities (1) identify applicable equipment and environmental requirements; (2) establish, demonstrate, and document the level of qualification (including configuration, maintenance, surveillance, and replacement requirements); and (3) maintain (or preserve) qualification. The EQ Program employs aging evaluations based on 10 CFR 50.49(f) qualification methods. Components in the EQ Program must be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

A2.1.16 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program (FMP) is an existing program that manages the fatigue life of reactor coolant pressure boundary components by tracking and evaluating key plant events. This program monitors operating transients to date, calculates fatigue usage factors to date, and permits implementation of corrective measures in order not to exceed the design limit on fatigue usage.

The FMP will be enhanced with guidance for the use of the FatiguePro software package and updated methodology for environmental fatigue factors in establishing updated fatigue life calculations for components. The enhancement is scheduled for completion prior to the period of extended operation.

A2.1.17 FIRE PROTECTION PROGRAM

The Fire Protection Program provides guidance for performance of periodic visual inspections to manage aging of the various materials comprising rated fire barriers. These include (a) sealants in rated penetration seals (subject to shrinkage due to weathering); (b) concrete and steel in fire rated walls, ceilings, and floors (subject to loss of material due to flaking and abrasion; separation and concrete damage due to relative motion, vibration, and shrinkage); and (c) steel in rated fire doors (subject to loss of material due to corrosion and wear or mechanical damage). In addition this program requires testing of the diesel-driven fire pump to verify that it is performing its intended function. This activity manages aging of the fuel oil supply line to the diesel engine, which may experience loss of material due to corrosion. Inspection and testing is performed in accordance with the guidance of applicable standards.

A2.1.18 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program manages aging of water-based fire protection systems due to loss of material and biofouling. Program activities include periodic maintenance, testing, and inspection of system piping and components containing water (e.g., sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes). Inspection and testing is performed in accordance with the guidance of applicable National Fire Protection Association (NFPA) Codes and Standards and the Nuclear Electric Insurance Limited (NEIL) Members' Manual.

Enhancements to the Fire Water System Program include the following revisions to existing activities that are credited for license renewal:

- Incorporate inspections to detect and manage loss of material due to corrosion into existing periodic test procedures.
- Specify periodic component inspections to verify that loss of material is being managed.
- Add procedural guidance for performing visual inspections to monitor internal corrosion and detect biofouling.
- Add requirements to periodically check the water-based fire protection systems for microbiological contamination.
- Measure fire protection system piping wall thickness using non-intrusive techniques (e.g., volumetric testing) to detect loss of material due to corrosion.
- Establish an appropriate means of recording, evaluating, reviewing, and trending the results of visual inspections and volumetric testing.
- Define acceptance criteria for visual inspections and volumetric testing.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.19 FLOW-ACCELERATED CORROSION PROGRAM

The Flow-Accelerated Corrosion (FAC) Program (also referred to as the Erosion/Corrosion Program) manages aging effects due to flow-accelerated corrosion in carbon steel and low alloy steel piping containing single-phase and two-phase high-energy fluids. Program activities include (1) analysis using a predictive code (CHECWORKS) to determine critical locations, (2) baseline inspections to determine the extent of thinning at the selected locations, (3) follow-up inspections to confirm the predictions, and (4) repair or replacement of components, as necessary. The program considers the recommended actions in NRC Bulletin 87-01 and Information Notice 91-18, and implements the guidelines for an effective FAC program presented in EPRI Report NSAC-202L-R2. The program also implements the recommendations provided in NRC Generic Letter 89-08, *Erosion/Corrosion Induced Pipe Wall Thinning*.

A2.1.20 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program manages loss of material due to corrosion that may result from introduction of contaminants into the plant's fuel oil storage tanks. Program activities include (1) sampling and chemical analysis of the fuel oil inventory at the plant, (2) sampling, testing, and analysis of new fuel oil as it is unloaded at the plant, and (3) cleaning and inspection of fuel oil storage tanks. This program is based on maintaining fuel oil quality in accordance with the guidelines of applicable American Society for Testing Materials (ASTM) Standards.

Enhancements to the Fuel Oil Chemistry Program include the following revisions to existing activities that are credited for license renewal:

- Add requirements to periodically check diesel fuel oil for microbiological organisms.
- Add requirements to periodically inspect fuel oil storage tanks for evidence of significant degradation, including a requirement that the tank thickness be determined.
- Incorporate periodic tests for microbiological organisms into Fuel Oil Chemistry Program procedures.
- Include evaluation of microbiological organisms in fuel oil as a measure of contamination.
- Add a requirement for multilevel sampling.
- Establish a quarterly (minimum) frequency to check fuel oil for microbiological organisms, and a requirement for quarterly trending of analysis results.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.21 FUSE HOLDER INSPECTION PROGRAM

The Fuse Holder Inspection Program is a new plant-specific program that applies to fuse holders located outside of active devices that have aging effects requiring management. This program requires testing to detect deterioration of metallic clamps that would affect the ability of in-scope fuse holders to perform their intended function. The Fuse Holder Inspection Program includes the following aging stressors: moisture, fatigue, ohmic heating, mechanical stress, vibration, thermal cycling, electrical transients, chemical contamination, oxidation, and corrosion.

Analytical trending will not be included in this activity because the parameters monitored may vary depending upon the test method selected. This is an exception to the “Monitoring and Trending” element in Appendix A.1.2.3.5 to NUREG-1800. This program will be implemented prior to the period of extended operation.

A2.1.22 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD HANDLING SYSTEMS PROGRAM

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program (referred to herein as the Crane Inspection Program) manages loss of material due to corrosion of cranes within scope of license renewal (WSLR). Program activities include (1) performance of various maintenance activities on a specified frequency; and (2) pre-operational inspections of equipment prior to lifting activities. Crane inspection activities are based on the mandatory requirements of applicable industry standards and implement the guidance of NUREG-0612.

The Crane Inspection Program will be enhanced to add specific direction for performance of pre-lift corrosion inspections of certain hoist lifting assembly components. The enhancement is scheduled for completion prior to the period of extended operation.

A2.1.23 MASONRY WALL PROGRAM

The Masonry Wall Program manages aging effects so that the evaluation basis established for each masonry wall WSLR remains valid through the period of extended operation. The Masonry Wall Program is based on the structures monitoring requirements of 10 CFR 50.65. The Masonry Wall Program is implemented by the Structures Monitoring Program for managing specific aging effects.

A2.1.24 NON-EQ ELECTRICAL CABLES AND CONNECTIONS PROGRAM

The Non-EQ Electrical Cables and Connections Program is a new program that manages aging of cables and connectors WSLR exposed to adverse localized temperature, moisture, or radiation environments. Program activities include periodic visual inspection of susceptible cables for evidence of cable and connection jacket surface anomalies. This program will be implemented prior to the period of extended operation.

A2.1.25 NON-EQ ELECTRICAL CABLES USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Non-EQ Electrical Cables Used in Instrumentation Circuits Program manages aging of cables exposed to adverse localized temperature and radiation environments that could result in loss of insulation resistance. It applies to accessible and inaccessible electrical cables that are not in the EQ Program and are used in circuits with sensitive, high-voltage, low-level signals such as radiation monitoring, nuclear instrumentation, and other non-nuclear instrumentation that are within the scope of license renewal. Activities include routine calibration tests of instrumentation loops and are implemented through the Surveillance Testing and Preventive Maintenance Programs. Testing is based on requirements of the plant technical specifications and implemented through the work control system.

Enhancements to the Non-EQ Electrical Cables Used in Instrumentation Circuits Program include the following revisions to existing activities that are credited for license renewal:

- Implement reviews of cable calibration or surveillance data for indications of aging degradation affecting instrument circuit performance. The first reviews will be completed prior to the period of extended operation and every ten years thereafter.

- In cases where a calibration or surveillance program does not include the cabling system in the testing circuit, or as an alternative to the review of calibration results described above, provide requirements and procedures to perform cable testing to detect deterioration of the insulation system, such as insulation resistance tests or other testing judged to be effective in determining cable insulation condition. The first test will be completed prior to the perod of extended operation and every 10 years thereafter.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.26 NON-EQ INACCESSIBLE MEDIUM VOLTAGE CABLES PROGRAM

The Non-EQ Inaccessible Medium Voltage Cable Inspection Program is a new program that provides reasonable assurance that the intended functions of inaccessible medium voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49, and are potentially exposed to adverse localized environments caused by moisture and significant voltage, are maintained consistent with the current licensing basis through the period of extended operation. Program activities include visual inspections of accessible areas of ducts and banks via manholes or inspection covers, and testing if warranted by an engineering evaluation. The program considers the technical information and guidance provided in applicable industry publications. This program will be implemented prior to the period of extended operation.

A2.1.27 NON-SEGREGATED BUS INSPECTION PROGRAM

The Non-Segregated Bus Inspection Program is a new plant-specific program that manages aging effects for components and materials internal to the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V buses required for the recovery of offsite power following a Station Blackout (SBO) event. These normally-energized components are not subject to the environmental qualification requirements of 10 CFR 50.49, but can be affected by elevated temperatures prior to the end of the period of extended operation. Program activities include visual inspection of normally-energized non-segregated bus duct internal components for surface anomalies, such as embrittlement, discoloration, cracking, chipping, or surface contamination, which could affect the life of the component. The program considers the technical information and guidance provided in applicable industry publications.

Analytical trending will not be included in this activity because the parameters inspected are not readily quantifiable in an appropriate form. This is an exception to the “Monitoring and Trending” element in Appendix A.1.2.3.5 to NUREG-1800. This program will be implemented prior to the period of extended operation.

A2.1.28 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program is a new program that manages aging effects with potentially long incubation periods for susceptible components WSLR. Program activities include visual, volumetric, and other established inspection techniques consistent with industry practice to provide a means of verifying that an aging effect is either (1) not occurring, or (2) progressing so slowly that it has a negligible effect on the intended function of the structure or component. The program also provides measures for verifying the effectiveness of existing aging management programs. This program is a new program that will be implemented prior to the period of extended operation.

A2.1.29 OPEN-CYCLE COOLING WATER SYSTEM PROGRAM

The Open-Cycle Cooling Water System Program manages aging of components exposed to raw, untreated (e.g., service) water. Program activities include (a) surveillance and control of biofouling (including biocide injection), (b) verification of heat transfer capabilities for components cooled by the Service Water System, (c) inspection and maintenance, (d) walkdown inspections, and (e) review of maintenance, operating and training practices and procedures. Inspections may include visual, Ultrasonic Testing (UT), and Eddy Current Testing (ECT) methods. This program is based on the recommendations of Generic Letter (GL) 89-13.

A2.1.30 PREVENTIVE MAINTENANCE PROGRAM

The scope of the Preventive Maintenance (PM) Program includes, but is not limited to, valve bodies, heat exchangers, expansion joints, tanks, ductwork, fan/blower housings, dampers, and pump casings. This program provides for performance of various maintenance activities on a specified frequency based on vendor recommendations and operating experience. These activities provide opportunities for component condition monitoring to manage the effects of aging for many SSCs WSLR.

Enhancements to the PM Program include the following revisions to existing activities that are credited for license renewal:

- Expand the PM Program to encompass activities for certain additional components identified in the license renewal aging management reviews.

- Explicitly define the aging management attributes, including the systems and the component types/commodities included in the program.

- Specifically list activities credited for aging management, parameters monitored, and the aging effects detected.
- Establish a requirement that inspection data be monitored and trended.
- Establish detailed parameter-specific acceptance criteria.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.31 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program manages cracking of and loss of material from the reactor pressure vessel closure studs. This program implements guidance provided in Regulatory Guide 1.65. Augmented examinations are performed through the Inservice Inspection Program, which is based on the 1989 edition of the ASME Boiler and Pressure Vessel Code with no Addenda, as approved by the NRC in plant-specific exemptions. This is an exception to the program described in NUREG-1801 (which cites ASME Section XI requirements covered in the 1995 edition through 1996 addenda).

A2.1.32 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is an existing program that manages loss of fracture toughness due to neutron irradiation embrittlement in the Reactor Pressure Vessel (RPV) beltline material. Program activities include (1) periodic withdrawal and testing of surveillance capsules from the RPV; (2) use of test results and allowable stress loadings for the ferritic RPV materials to determine operating limits; and (3) comparison with a large industry data set to confirm validity of test results. Analysis and testing are based on the requirements of 10 CFR 50, Appendix H, and ASTM Standard E-185. NMPNS has committed to implement the Integrated Surveillance Program (ISP) described in BWRVIP-116 (currently under review by the NRC staff). Since the proposed ISP for license renewal has no provision to store tested specimens, such storage is considered unnecessary. This is an exception to the program described in NUREG-1801.

Enhancements to the Reactor Vessel Surveillance Program include the following revisions to existing activities that are credited for license renewal:

- Incorporate the requirements and elements of the Integrated Surveillance Program, published by the BWRVIP, into the Reactor Vessel Surveillance Program.

- Project analyses of upper shelf energy and pressure temperature limits to 60 years using methods prescribed by Regulatory Guide 1.99, Revision 2, and include the applicable bounds of the data, such as operating temperature and neutron fluence.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.33 SELECTIVE LEACHING OF MATERIALS PROGRAM

The Selective Leaching of Materials Program is a new program that manages aging of components susceptible to selective leaching. The potentially susceptible components include valve bodies, valve bonnets, pump casings, and heat exchanger components in various systems. This program will be implemented through the One Time Inspection Program prior to the period of extended operation.

A2.1.34 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program manages aging of structures, structural components, and structural supports WSLR. The program provides for periodic visual inspections, surveys, and examination of all safety related buildings (including the containment buildings and substructures within the primary containment) and various other buildings WSLR. Program activities identify degradation of materials of construction, which include structural steel, concrete, masonry block, sealing materials. While not credited for mitigation of aging, protective coatings are also inspected under this program. The Structures Monitoring Program, which was initially developed to meet the regulatory requirements of 10 CFR 50.65, implements guidance provided in Regulatory Guide 1.160, NUMARC 93-01 and NEI 96-03.

Enhancements to the Structures Monitoring Program include the following revisions to existing activities that are credited for license renewal:

- Expand the program to include the following activities or components WSLR, but not within the current scope of 10 CFR 50.65: (a) NMP2 Fire Rated Assemblies & Watertight Penetration Visual Inspections and (b) NMP2 masonry walls in the Turbine Building and Service Water Tunnel serving a fire barrier function.

- Expand the parameters monitored during structural inspections to include those relevant to aging effects requiring management identified for structural bolting.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.35 SYSTEMS WALKDOWN PROGRAM

The Systems Walkdown Program manages aging effects for accessible external surfaces. The specific aging effect of concern is loss of material from external surfaces of pumps, valves, piping, bolts, heat exchangers, tanks, expansion joints, electrical penetrations, electrical enclosures and cabinets, HVAC components, and other carbon steel components. Program activities include system engineer walkdowns (i.e., field evaluations of system components to assess system performance and material condition), evaluation of inspection results, and appropriate corrective actions.

Enhancements to the Systems Walkdown Program include the following revisions to existing activities that are credited for license renewal:

- Explicitly state the aging management attributes, including the systems and component types/commodities included in the program.
- Specifically list parameters monitored, and provide guidance for assessment of identified deterioration. This will include extent of corrosion (loss of material), condition of coatings (material degradation), leakage or indications of leakage, etc.
- Components WSLR included in the Systems Walkdown Program will be visually inspected for loss of material, material degradation and leakage per the upgraded program documentation. The frequency of inspections will be at least once per refuel cycle for each structure and system.
- Define a methodology that specifies consistent criteria for program data collection.
- Specify the parameters and data that will be monitored and trended.
- Specify acceptance criteria for visual inspections and guidelines for using the criteria to assess observed material condition against intended function for inspected SSCs.
- Upgrade program documents so that they fall under the administrative controls program.

- Include previous operating experience and ensure future operating experience is properly incorporated.

Enhancements are scheduled for completion prior to the period of extended operation.

A2.1.36 WATER CHEMISTRY CONTROL PROGRAM

The Water Chemistry Control Program manages aging effects by controlling the internal environment of the reactor water, feedwater, condensate, and control rod drive systems, and related auxiliaries (such as the suppression pool, condensate storage tank, and spent fuel pool). The aging effects/mechanisms of concern include (1) stress corrosion cracking; (2) loss of material due to corrosion; and (3) fouling. Program activities include monitoring and controlling concentrations of known detrimental chemical species below the levels known to cause degradation. The Water Chemistry Control Program implements the guidelines for BWR water chemistry presented in EPRI Reports TR-103515-R1 and TR-103515-R2. This is an exception to the program described in NUREG-1801 (which identifies the February 1994 version of BWRVIP-29 as the basis for BWR water chemistry programs).

A2.2 TIME-LIMITED AGING ANALYSES SUMMARIES

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

A2.2.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. The evaluation of reactor vessel neutron embrittlement is a TLAA. The following TLAA discussions are related to the issue of neutron embrittlement:

- Upper Shelf Energy
- Pressure-Temperature (P-T) Limits
- Axial Weld Failure Probability

A2.2.1.1 UPPER-SHELF ENERGY

Ferritic Reactor Pressure Vessel (RPV) materials undergo a transition in fracture behavior from brittle to ductile as the temperature of the material is increased. Charpy V-notch tests are conducted in the nuclear industry to monitor changes in the fracture behavior during irradiation. Neutron irradiation to fluences above approximately $1x10^{17}$ n/cm^2 causes an upward shift in the ductile-to-brittle transition temperature and a drop in upper-shelf energy (USE). To satisfy the acceptance criteria for USE contained in 10 CFR 50 Appendix G, the RPV beltline materials must have a Charpy USE of no less than 50 ft-lbs throughout the life of the RPV unless it can be demonstrated that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The USE for the limiting beltline weld materials for and the limiting beltline plate materials for NMP2 is predicted to remain above 50 ft-lbs throughout the period of extended operation, based on projected fluence values. Therefore, the USE for the NMP2 RPV beltline materials has been projected (reevaluated) for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A2.2.1.2 PRESSURE - TEMPERATURE (P – T) LIMITS

10 CFR 50 Appendix G requires that the RPV be operated within established pressure-temperature (P-T) limits during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. NMP2 Technical Specifications contain P-T limit curves for heatup, cooldown, inservice leakage testing, and hydrostatic testing, and limit the maximum rate of change of reactor coolant temperature.

The P-T limit curves are periodically revised to account for changes in fracture toughness of the RPV components due to anticipated neutron embrittlement effects for higher accumulated fluences. Calculation of P-T limit curves using the projected fluence at the end of the period of extended operation would result in unnecessarily restrictive operating curves. However, projection of the Adjusted Reference Temperature (ART), which is used in development of the curves, to the end of the period of extended operation provides assurance that development of P-T limit curves will be feasible up to the maximum predicted Effective Full Power Year (EFPY).

Projections of the ART values for the beltline materials have been made for the period of extended operation, providing reasonable assurance that it will be possible to prepare P-T curves that will permit continued plant operation. The P-T curves (and the related Technical Specifications) will continue to be updated either as required by 10 CFR 50, Appendix G, to assure the operational limits remain valid at the current cumulative neutron fluence levels, or on an as-needed basis to provide appropriate operational flexibility.

A2.2.1.3 AXIAL WELD FAILURE PROBABILITY

In the safety evaluation presented in *Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report* (Reference A2.3.6), the NRC staff indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than $5x10^{-6}$ per reactor year, given the assumptions on flaw density, distribution, and location described in the SER. Projected values of mean Reference Temperature Nil Ductility Transition Temperature (RT_{NDT}) and upper bound RT_{NDT} for the limiting axial welds at NMP2 are below the bounding mean RT_{NDT} value determined by the NRC staff in the SER for BWRVIP-74-A (Reference A2.3.1). Thus, there is reasonable assurance that the RPV failure frequency due to failure of the limiting axial weld is expected to remain less than $5x10^{-6}$ per reactor year for NMP2.

Inspection of the axial welds in accordance with the ASME XI code requirements will continue at NMP2 during the period of extended operation.

Supporting analyses will be completed prior to the period of extended operation to confirm that the RPV axial weld failure probability for the limiting NMP2 axial weld remains bounded for the period of extended operation. Based on the scoping evaluation discussed above, there is reasonable assurance the failure probability will remain acceptable for the period of extended operation.

A2.2.2 METAL FATIGUE ANALYSIS

ASME Section III requires calculation of cumulative usage factors (CUFs) to demonstrate fatigue-tolerant design for reactor vessels, vessel internals, Class 1 piping and components, metal containments, and penetrations. These values are indexed to the number of transients anticipated over the design life of the component (usually 40 years).

Designated plant events have been counted and categorized to ensure that the number of actual operational transient cycles does not exceed the number of transients assumed in the plant design for fatigue. For certain events that affect fatigue usage, linear projections of the actual data to the end of the period of extended operation will exceed the analyzed number of design basis transients. For those locations where additional fatigue analysis is required to take advantage of the implicit margin (and to more accurately determine CUFs), the EPRI FatiguePro fatigue monitoring software will be implemented.

The following thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs:

- Reactor Vessel Fatigue Analysis
- ASME Section III Class 1 Piping and Components Fatigue Analysis
- Feedwater (FWS) Nozzle and Control Rod Drive Return Line (CRDRL) Nozzle Fatigue and Cracking Analyses
- Non-ASME Section III Class 1 Piping and Components Fatigue Analysis
- Reactor Vessel Internals Fatigue Analysis
- Environmentally Assisted Fatigue

A2.2.2.1 REACTOR VESSEL FATIGUE ANALYSIS

The original design of RPV pressure boundary components included analyses of fatigue resistance. Components were evaluated by calculating the alternating stresses associated with applicable design transients and determining a CUF based on the number of anticipated transients for the original 40-year life of the plant. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

For the critical RPV component locations, transients contributing to fatigue usage will be tracked by the FMP (Appendix A2.1.16) with additional usage added to the baseline CUF. The FMP provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

A2.2.2.2 ASME SECTION III CLASS 1 PIPING AND COMPONENTS FATIGUE ANALYSIS

The reactor coolant pressure boundary (RCPB) piping at NMP2 was designed to meet ASME Section III Class 1 requirements for fatigue loading. The subject piping and components were evaluated by calculating the alternating stresses associated with applicable design transients and determining a CUF based on the number of anticipated transients for the original 40-year life of the plant. Fatigue-tolerant design is demonstrated for components with CUFs less than 1.0 (or less than 0.1 for components in break exclusion zones). Additional pipe break postulation criteria are applied to high-energy ASME Class 1 piping with a CUF greater than 0.1.

For the bounding locations for ASME Class 1 systems, transients contributing to fatigue usage will be tracked by the FMP (Appendix A2.1.16) with additional usage added to the baseline CUF.

A2.2.2.3 FEEDWATER (FWS) NOZZLE AND CONTROL ROD DRIVE RETURN LINE (CRDRL) NOZZLE FATIGUE AND CRACKING ANALYSES

BWRs have experienced fatigue crack initiation and growth in Feedwater System (FWS) and Control Rod Drive Return Line (CRDRL) nozzles. Rapid thermal cycling (occurring as a result of bypass leakage past loose-fitting thermal sleeves, or in nozzles lacking thermal sleeves) initiated fatigue cracks that propagated due to larger (in terms of the magnitude of temperature and pressure change) thermal cycles resulting from plant transients. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, identifies interim and long-term procedural and design changes to minimize thermal fatigue cracking, as well as inspection requirements.

Various calculations were prepared in response to NUREG-0619 (e.g., to support enhanced inspection intervals, to incorporate updated fatigue crack growth curves, etc.), and CUFs were determined on the basis of anticipated transients for the original 40-year life of the plant. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

The NMP2 FWS nozzles require continued monitoring (including analysis using FatiguePro) to demonstrate compliance over the period of extended operation. Transients contributing to fatigue usage of the FWS nozzles will be tracked by the FMP (Appendix A2.1.16) with additional usage added to the baseline CUF.

In NUREG-0619, the NRC evaluated a number of options proposed by GE to resolve the problem of cracking in the CRDRL nozzle and identified acceptable methods for performing the modifications. NMP2 implemented the recommendation to cut and cap the CRDRL nozzle without rerouting the CRDRL. Therefore, there are no fatigue concerns associated with the CRDRL nozzle at NMP2.

A2.2.2.4 NON-ASME SECTION III CLASS 1 PIPING AND COMPONENTS FATIGUE ANALYSIS

With the exception of the RCPB piping at NMP2, piping and components WSLR were designed to codes other than ASME Section III Class 1. Applicable codes include ASA B31.1-1955 and ASME Section III Class 2 or 3. These codes do not require explicit fatigue analysis. Instead, the effects of cyclic loading are accounted for through application of stress range reduction factors based on the anticipated number of equivalent full temperature thermal expansion cycles over the original 40-year life of the plant.

No locations in the Non-ASME Class 1 piping at NMP2 are expected to require development of fatigue analyses. ASME Section III Class 2 and 3 piping generally experiences less severe thermal transients and does not include any of the locations identified in NUREG/CR-6260 (Reference A2.3.2). Therefore, the existing fatigue design basis for NMP2 is considered valid for the period of extended operation. If fatigue monitoring of ASME Class 1 piping at NMP2 indicates higher fatigue usage than expected, Non-ASME Class 1 piping will be evaluated for possible fatigue concerns.

A2.2.2.5 REACTOR VESSEL INTERNALS FATIGUE ANALYSIS

Determination of CUFs was not a design requirement for reactor vessel internals at NMP2. However, certain locations were evaluated for fatigue using ASME Section III methods to calculate alternating stresses and determine CUF values based on a number of anticipated transients (generally, for the original 40-year life of the plant). Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

While all CUFs determined for components comprising the reactor vessel internals are less than 1.0, the calculated values for certain locations exceed 0.4 (considered a general threshold of significance). Thus, the CUFs for these locations (i.e., the shroud, core support plate and studs, and jet pumps at NMP2) will be revised or reevaluated to remove conservatism and/or encompass the period of extended operation. In particular, a more extensive fatigue analysis of the NMP2 jet pumps (whose original design analyses are proprietary to GE) will be performed prior to the period of extended operation.

The potential for cracking of components comprising the reactor vessel internals, both due to fatigue and (more significantly) intergranular stress corrosion cracking (IGSCC), is managed by the BWR Vessel Internals Program (Appendix A2.1.13), which incorporates comprehensive inspection and evaluation guidelines issued by the BWRVIP and approved by the NRC. These activities provide assurance that any unexpected degradation resulting from fatigue in the reactor vessel internals for the current license period and the period of extended operation will be identified and corrected. Therefore, the effects of fatigue on the intended function(s) of the reactor vessel internals will be adequately managed for the period of extended operation.

A2.2.2.6 ENVIRONMENTALLY ASSISTED FATIGUE

Generic Safety Issue (GSI) 190, *Fatigue Evaluation of Metal Components for 60-year Plant Life*, was established to address NRC concerns regarding environmental effects on fatigue of pressure boundary components for 60 years of plant operation. The NRC staff studied the probability of fatigue failure for selected metal components based on the increased CUFs determined in NUREG/CR-6260 (Reference A2.3.2) and a 60-year plant life. The NRC closed this GSI, and concluded that environmental effects did not substantially affect core damage frequency. However, since the nature of age-related degradation indicated the potential for an increase in the frequency of pipe leaks as plants continue to operate, licensees are required to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

NMP2 will assess the impact of the reactor coolant environment on a sample of critical component locations, including locations equivalent to those identified in NUREG/CR-6260 as part of the FMP (Appendix A2.1.16). These locations will be evaluated by applying environmental correction factors (F_{en}) to existing and future fatigue analyses. Evaluation of the sample of critical components will be completed prior to the period of extended operation.

A2.2.3 ENVIRONMENTAL QUALIFICATION (EQ)

The following EQ analyses have been identified as TLAAs:

- Electrical Equipment EQ
- Mechanical Equipment EQ

A2.2.3.1 ELECTRICAL EQUIPMENT EQ

10 CFR 50.49 requires that certain safety related and non-safety related electrical equipment remain functional during and after identified Design Basis Events. To establish reasonable assurance that this equipment can function when exposed to postulated harsh environmental conditions, licensees are required to determine the equipment's qualified life and to develop a program that maintains the qualification of that equipment.

For components within the scope of the EQ Program (Appendix A2.1.15), analyses of thermal exposure, radiation exposure, and mechanical cycle aging that cannot be shown to remain valid for the period of extended operation will be projected to extend the qualification of components before reaching the aging limits established in the applicable evaluation, or the components will be refurbished or replaced.

A2.2.3.2 MECHANICAL EQUIPMENT EQ

To demonstrate compliance with General Design Criterion 4 of Appendix A to 10 CFR 50, the NRC staff required that NMP2 submit evaluations of the environmental effects on nonmetallic subcomponents comprising safety related mechanical equipment that must remain functional in harsh environments during and after identified Design Basis Events (Reference A2.3.3). Threshold radiation values and maximum service temperatures for these materials were compared with the maximum postulated environmental conditions to establish qualification. If necessary, a material replacement life limit was calculated.

For components within the scope of the EQ Program (Appendix A2.1.15), analyses of thermal exposure, radiation exposure, and mechanical cycle aging that cannot be shown to remain valid for the period of extended operation will be projected to extend the qualification of the components before reaching the aging limits established in the applicable evaluation, or the components will be refurbished or replaced.

A2.2.4 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

For NMP2, the containment liner analysis has been identified as a TLAA.

A2.2.4.1 CONTAINMENT LINER ANALYSIS

The NMP2 Mark II containment is a reinforced concrete structure consisting of a drywell chamber located above a suppression pool, with a drywell floor separating the two. Except at various penetrations and access openings through the walls, the primary containment liner is a continuous steel membrane (attached to the inside face of the wall) that functions as a leak-tight barrier to the release of fission products. The containment wall is designed to withstand anticipated loads without participation of the liner as a structural component. The portion of the liner functioning as the suppression pool floor is welded to the wall liner through a corner junction embedment. The fatigue analysis for the NMP2 containment liner, in accordance with requirements specified in ASME Section III, was conducted assuming a 40-year life. Fatigue-tolerant design is demonstrated for those locations with CUFs less than 1.0.

The CUF calculations for the NMP2 containment liner will be reevaluated based on an increased number of assumed cyclic loads prior to the period of extended operation. All cyclic loads considered in the original fatigue analyses, including hydrodynamic loads and loads resulting from Type A leak rate tests, will be reevaluated and revised as necessary. The analysis will also evaluate the potential for additional fatigue for subcomponents of the containment liner, including the heads of various hatches, airlock bulkheads, and the drywell head. The revised analysis will demonstrate that the 60-year CUF values for all controlling locations remains less than 1.0. Therefore, the containment liner analysis has been projected for the period of extended operation.

A2.2.5 OTHER PLANT-SPECIFIC TLAAS

The following Plant-Specific TLAAs have been identified for NMP2:

- RPV Biological Shield
- Main Steam Isolation Valve Corrosion Allowance
- Stress Relaxation of Core Plate Hold Down Bolts

A2.2.5.1 RPV BIOLOGICAL SHIELD

Discovery of weld defects during fabrication of the Biological Shield Wall (BSW) resulted in stress and fracture mechanics analyses to determine an acceptable flaw size. The results showed the majority of the flaws were acceptable, while some flaws required repair (Reference A2.3.4). A related calculation was prepared to estimate the amount of neutron irradiation embrittlement (in terms of the 30 ft-lb transition temperature shift) of the BSW structural steel at the end of a 40-year life.

Based on projected fluence value, the USE of the BSW material is reduced but does not invalidate the original fracture mechanics analyses. Therefore, fracture toughness of the NMP2 BSW has been projected (reevaluated) for the period of extended operation.

A2.2.5.2 MAIN STEAM ISOLATION VALVE CORROSION ALLOWANCE

The Main Steam Isolation Valve (MSIV) bodies were fabricated from low-alloy steel and are exposed to a dry steam environment during plant operation. During a refueling outage, the MSIVs are exposed to treated water and air. To provide for 40-year service in these environments, USAR Section 5.4.5 indicates a 0.120-inch corrosion allowance was added to the MSIV wall thickness in addition to the minimum required by applicable codes.

The amount of wall thinning based on maximum expected corrosion rates of the MSIV bodies remains bounded by the corrosion allowance assumed in the design of these valves. Therefore, the corrosion allowance calculation for the NMP2 MSIV bodies remains valid for the period of extended operation.

A2.2.5.3 STRESS RELAXATION OF CORE PLATE HOLD-DOWN BOLTS

Hold-down bolts located around the rim of the core plate are subcomponents of the core plate assembly that ensure the core plate safety function. Preload in these bolts could be reduced over time by the effects of IGSCC and fluence; thus, Reference A2.3.5 determined that loss of preload should be evaluated as a potential TLAA.

The potential for cracking of components comprising the reactor vessel internals due to IGSCC is managed by the BWR Vessel Internals Program (Appendix A2.1.13). Prior to the period of extended operation, NMP2 will either:

(1) Install core plate wedges (as part of a proposed core shroud tie-rod repair) to eliminate the need for the enhanced inspections of the core plate hold-down bolts recommended by BWRVIP-25; or

(2) Perform an analysis (incorporating detailed flux/fluence analyses and improved stress relaxation correlations) to demonstrate that the core plate hold-down bolts can withstand all normal, emergency, and faulted loads considering the effects of stress relaxation, until the end of the period of extended operation.

These activities provide assurance that stress relaxation of the NMP2 core plate hold-down bolts will be adequately managed for the period of extended operation.

A2.3 REFERENCES

A2.3.1 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated October 18, 2001, *Subject: Acceptance for Referencing of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74-A)" and Appendix A, "Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10CFR54.21)"*.

A2.3.2 NUREG/CR-6260, INEL-95/0045, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, February 1995.*

A2.3.3 Letter from U.S. Nuclear Regulatory Commission to Niagara Mohawk Power Corporation dated October 25, 1983, *Subject: Summary of Meeting with Niagara Mohawk Power Corporation on Deviations from the Standard Review Plan (NUREG-0800) for Nine Mile Point Nuclear Station, Unit 2.*

A2.3.4 Letter from Niagara Mohawk Power Corporation to U.S. Nuclear Regulatory Commission dated August 1, 1980, forwarding the final report concerning the Nile Mile Point Unit 2 biological shield wall in accordance with 10 CFR 50, paragraph 50.55(e)(3).

A2.3.5 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated December 7, 2000, *Subject: Safety Evaluation for Referencing of BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) Report for Compliance with the License Renewal Rule (10 CFR Part 54) and Appendix B, BWR Core Plate Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)*.

A2.3.6 Letter from U.S. Nuclear Regulatory Commission to BWRVIP Chairman dated March 7, 2000, *Subject: Supplement to Final Safety Evaluation of the BWR Vessel and Internal Project BWRVIP-05 Report (TAC No. MA3395).*