June 10, 2005

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10 CFR Part 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Response to Request for Additional Information and Submittal of Additional Information in Support of the Monticello License Renewal Application (TAC No. MC6440)

References:
1) NMC letter to NRC, "Application for Renewed Operating License," (L-MT-05-014), dated March 16, 2005.

2) NRC letter to NMC, "Request for Additional Information (RAI) for the Review of the Monticello Nuclear Generating Plant License Renewal Application," dated May 12, 2005 (ADAMS Accession No. ML051330005)

3) NRC letter to NMC, "Summary of Meeting Held on May 11, 2005, Between the U.S. Nuclear Regulatory Commission (NRC) Staff and Nuclear Management Company, LLC Representatives to Discuss the Monticello Nuclear Generating Plant License Renewal," dated May 18, 2005 (ADAMS Accession No. ML051400115)

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted a License Renewal Application (LRA) (Reference 1) to renew the operating licenses for the Monticello Nuclear Generating Plant (MNGP).

By letter dated May 12, 2005, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) regarding the LRA for MNGP (Reference 2).

During a meeting on May 11, 2005, between the NRC Staff and NMC representatives to discuss the MNGP LRA (Reference 3), NMC agreed to provide additional information in response to NRC questions during the meeting.

Enclosure 1 provides NMC's response to the NRC's question RAI 4.8-1, contained in the NRC RAI dated May 12, 2005.

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Enclosure 2 provides NMC's response to the NRC's question RAI A2.1-1, also contained in the NRC RAI dated May 12, 2005. Revised Appendix A supercedes in its entirety Appendix A of NMC's letter, dated March 16, 2005. The revised Appendix A includes the commitments in the individual programs described in Section A2.1, "Aging Management Programs," and adds a new Section A.5, "Commitments," that provides a table summarizing the actions committed to by NMC in the License Renewal Application for the MNGP.

The new Appendix A Table A.5, "Commitments," supercedes Enclosure 3, "Compilation of Commitments Related to License Renewal Aging Management for the Monticello Nuclear Generating Plant," of NMC's letter dated March 16, 2005, and includes the two new commitments (listed below) that are made in this letter.

Enclosure 3 contains the additional information that NMC agreed to provide during the meeting with the NRC staff on May 11, 2005.

In addition to the responses provided in Enclosures 1 and 2, a sampling plan for the One-Time Inspection for the Aging Management Program (AMP) audit will be available for NRC review as discussed at the May 11, 2005 meeting.

This letter contains the following new regulatory commitments:

- NMC will perform top guide grid inspections using the EVT-1 method of examination, for the high fluence locations (grid beam and beam-to-beam crevice slot locations with fluence exceeding 5.0×10^{20} n/cm^2). Ten percent (10%) of the total population will be inspected within 12 years with a minimum of 5% inspected within the first 6 years.

- NMC will retain the capsules removed from the MNGP reactor vessel as part of the Reactor Vessel Surveillance Program.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on June 10, 2004.

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
License Renewal Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 1

Response to Request for Additional Information Regarding the Monticello Nuclear Generating Plant License Renewal Application

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted an application to renew the operating licenses for the Monticello Nuclear Generating Plant (MNGP), dated March 16, 2005. As a result of the NRC's Request for Additional Information (RAI) following their review of the License Renewal Application (LRA), NMC is hereby providing a response to the NRC's RAI.

NRC RAI 4.8-1

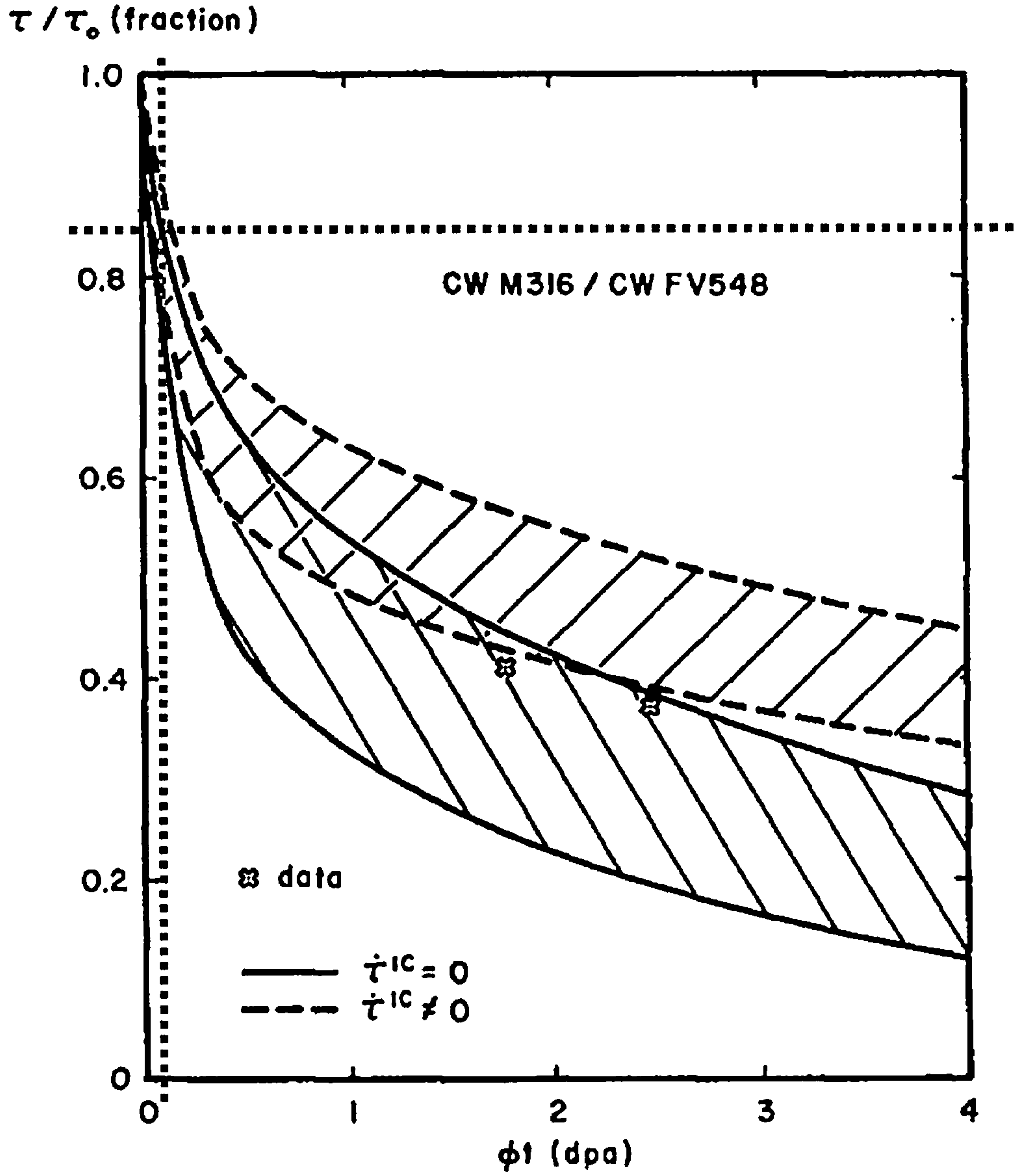
Please provide details of your analysis and technical basis for the Time-Limited Aging Analysis (TLAA) for Stress Relaxation of Rim Holddown Bolts, Section 4.8 of the LRA.

NMC Response to RAI 4.8-1

The core plate configuration for the MNGP is identified properly in Figure 2-3 of BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines." Therefore, MNGP was specifically considered in the original BWRVIP-25 evaluation, incorporating typical values of temperature and fluence. An analysis was initially performed for a 40-year plant life. The analysis was later performed for a 60-year plant life as discussed in Paragraph B.4 of BWRVIP-25; Appendix B of BWRVIP-25 addresses License Renewal. BWRVIP-25 identified a range of expected stress relaxation, for a group of plants which includes MNGP, from 5 to 19% for a 60-year plant life. The MNGP Time-Limited Aging Analysis (TLAA), conservatively noted the bounding value for stress relaxation of 19% (as documented in Section 4.8 of the LRA) for the core plate bolts over the 60-year operating period.

To more accurately address MNGP for License Renewal, a plant-specific calculation was performed that incorporated the MNGP core plate geometry, an operating temperature of 288°C (550°F) and a MNGP fluence calculation that was performed specifically for License Renewal in accordance with guidance provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, (LRA Section 4.2.1). The maximum fluence applicable to the bolts in the highest fluence region of the core plate was determined to be 2.2×10^{19} n/cm^2 at the end of the 60-year plant life. The resultant relaxation was determined to be 8% based on GE Design Documents. The analysis assumed that all of the bolts were at this fluence even though many bolts experience a lower fluence depending on their specific location. This plant-specific analysis is bounded by the original application that conservatively assumed a higher value of 19% relaxation.

Confidence in the stress relaxation value that was employed can be independently confirmed using other reports available to the NRC. Specifically, stress relaxation as a function of fluence has been presented in the Section 7 of BWRVIP-99, "Crack Growth Rates in Irradiated BWR Stainless Steel Internal Components." This section deals directly with stress relaxation as a function of fluence. Figure 7-13 from BWRVIP-99 is attached; it shows data and modeling projections of stress relaxation versus fluence in displacements per atom (dpa). One dpa is equivalent to a fluence of 6-7 x 10^{20} n/cm^2. It is noted that the data presented in the figure was measured on Type 316 stainless steel material. Comparison with Type 304 data is considered appropriate in that the two commercial alloys have the same single-phase austenitic microstructure and crystal structure, with no precipitates present in either alloy. The only compositional difference is the addition of molybdenum (Mo) to Type 316 to increase pitting resistance. The mechanical properties are essentially identical at 288°C (550°F). Therefore, the effect of irradiation on stress relaxation for both alloys is essentially the same. Since the fluence of interest for the MNGP calculations is less than 10% of the 1 dpa equivalent fluence, dotted lines have been included on the attached copy of Figure 7-13 to depict a fluence level of 0.1 dpa which substantiates that the projected stress relaxation is less than 19%. This supports the value used in the analysis and, in turn, the GE Design Curve.



BWRVIP-99: Figure 7-13. Radiation creep relaxation of shear stresses in springs of 20% cold worked 316 stainless steel along with modeling curves. The dotted line represents 0.1 dpa which is equivalent to a fluence of ~6-7 x 10^{19} n/cm.

ENCLOSURE 2

Response to Request for Additional Information Regarding the Monticello Nuclear Generating Plant License Renewal Application

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted an application to renew the operating licenses for the Monticello Nuclear Generating Plant (MNGP), dated March 16, 2005. As a result of the NRC's Request for Additional Information (RAI) following their review of the License Renewal Application (LRA), NMC is hereby providing a response to the NRC's RAI.

NRC RAI A2.1-1

Please include a description of any commitments, which were provided in your LRA transmittal letter, along with a schedule of implementation, as part of the Aging Management Program (AMP) descriptions in the USAR Supplement in Appendix A of the LRA.

NMC Response to RAI A2.1-1

Appendix A has been revised to include the commitments in the individual programs described in Section A2.1, "Aging Management Programs," and to add a new Section A.5, "Commitments," that provides a table summarizing the actions committed to by NMC in the License Renewal Application for the MNGP. This new Appendix A Table A.5, "Commitments," supercedes Enclosure 3, "Compilation of Commitments Related to License Renewal Aging Management for the Monticello Nuclear Generating Plant," of NMC's letter dated March 16, 2005. Appendix A is being reissued in its entirety to include the commitments in the text of Appendix A, (including the two new commitments made in this letter) account for repagination and to add the changes discussed above. The new Appendix A is as follows with revision bars included in the margin to highlight the revisions:

APPENDIX A

USAR SUPPLEMENT

APPENDIX A
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A1 APPENDIX A INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include an Updated Safety Analysis Report (USAR) Supplement. This appendix provides the required supplement for the Monticello Nuclear Generating Plant (MNGP) USAR.

Section A2 of this appendix contains a summary description of the programs for managing the effects of aging during the period of extended operation. Section A3 contains a summary of the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation. Section A4 contains summaries of TLAA supporting activities. Section A.5 contains the table of commitments related to License Renewal Aging Management for the Monticello Nuclear Generating Plant.

The information in Sections A2, A3, A4 and A5 will be incorporated into the MNGP USAR following receipt of the extended license in accordance with 10 CFR 50.71(e). Other changes to specific sections of the MNGP USAR based on the results of analyses performed in conjunction with the License Renewal Program and the NRC Safety Evaluation Report for MNGP License Renewal will also be made at that time.

A2 PROGRAMS THAT MANAGE THE EFFECTS OF AGING

This section provides summaries of the programs and activities, in alphabetical order, credited for managing the effects of aging. These aging management programs may not exist as discrete programs at MNGP. In many cases they exist as a compilation of various implementing documents that, when taken as a whole, satisfy the intent of NUREG-1800 and/or NUREG-1801 attributes.

The MNGP quality assurance program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, published July 2001. The elements of corrective action, confirmation process, and administrative controls in the quality assurance program are applicable to both safety related and non-safety related systems, structures, and components that are subject to an aging management review.

A2.1 Aging Management Programs

A2.1.1 10 CFR 50, Appendix J

Program Description

The MNGP 10 CFR 50, Appendix J Program specifies pneumatic pressure tests and visual examinations to verify the structural and leak tight integrity of the primary containment. An overall (Type A) pressure test assesses the capacity of the containment to retain design basis accident pressure. This test also measures total leakage through the containment pressure-retaining boundary. Local (Type B & C) tests measure leakage through individual penetration isolation barriers. These barriers are maintained as required to keep overall and local leakage under Technical Specification and plant administrative limits.

Tests are performed at intervals determined by the risk and performance factors applicable to each tested item in accordance with governing regulations and standards. This risk and performance based approach to testing provides reasonable assurance that developing leakage is detected and corrected well before it reaches a magnitude that could compromise containment function.

Visual examinations are performed prior to each Type A test. These examinations are also performed at least once during each containment in-service inspection period in which no Type A test is conducted. The examinations are performed to detect corrosion and other types of deterioration on the accessible surfaces of the containment.

A2.1.2 ASME Section XI In-Service Inspection, Subsections IWB, IWC, and IWD

Program Description

The MNGP ASME Section XI In-Service Inspection, Subsections IWB, IWC, and IWD Program is part of the MNGP ASME Section XI In-Service Inspection Program. This program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and is subject to the limitations and modifications of 10 CFR 50.55a. The program provides for condition monitoring of Class 1, 2, and 3 pressure-retaining components and their integral attachments.

Class 1 and 2 piping is being inspected in accordance with the Risk Informed In-Service Inspection (RI-ISI) Program as described in the Electric Power Research Institute (EPRI) Topical Report TR-112657, Rev. B-A, Revised Risk Informed In-Service Inspection Evaluation Procedure. The NRC has approved the use of RI-ISI in a safety evaluation documented in NRC letter dated July 24, 2002, "Monticello Nuclear Generating Plant - Risk Informed In-Service Inspection Program (TAC NO. MB3819).

The program is updated periodically as required by 10 CFR 50.55a.

The Plant Chemistry Program augments this program where applicable.

A2.1.3 ASME Section XI, Subsection IWF

Program Description

The MNGP ASME Section XI, Subsection IWF Program is part of the MNGP ASME Section XI In-Service Inspection Program. The ASME Section XI, Subsection IWF Program is performed in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and 10 CFR 50.55a and provides for condition monitoring of Class 1, 2, 3, and MC component supports. Component supports are selected for inspection in accordance with the ASME code classification. The quantity of component supports selected for examination is increased as a result of discovered support deficiencies. Visual inspection is the primary method for identifying deficiencies.

The program is updated periodically as required by 10 CFR 50.55a.

Commitments

Prior to the period of extended operation, the MNGP ASME Section XI, Subsection IWF Program will be enhanced to provide inspections of Class MC components supports consistent with NUREG-1801, Chapter III Section B1.3.

A2.1.4 Bolting Integrity

Program Description

The Bolting Integrity Program manages the aging affects associated with bolting in the scope of license renewal through periodic inspection, material selection, thread lubricant control, assembly and torque requirements, and repair and replacement requirements. These activities are based on the applicable requirements of ASME Section XI and plant operating experience and includes consideration of the guidance contained in NUREG-1339, Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants, EPRI NP-5769, Degradation and Failure of Bolting in Nuclear power Plants, EPRI TR-104213, Bolted Joint Maintenance & Application Guide, and EPRI NP-5067 Volumes 1 and 2, Good Bolting Practices. The program credits other MNGP Aging Management Programs for the inspection of installed bolts. These other programs are:

- 10 CFR 50, Appendix J,
- ASME Section XI In-Service Inspection, Subsections IWB, IWC and IWD,
- Primary Containment In-Service Inspection,
- Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems,
- ASME Section XI, Subsection IWF,
- Buried Piping and Tanks Inspection,
- Bus Duct Inspection,
- BWR Vessel Internals,
- Reactor Head Closure Studs Monitoring,
- System Condition Monitoring, and
- Structures Monitoring.

Commitments

Prior to the period of extended operation, the guidance for performing visual bolting inspections contained in EPRI TR-104213, Bolted Joint Maintenance & Application Guide, and the Good Bolting Practices Handbook (EPRI NP-5067 Volumes 1 and 2) will be included in the Bus Duct Inspection Program, Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program, Structures Monitoring Program and the System Condition Monitoring Program.

A2.1.5 Buried Piping & Tanks Inspection

Program Description

The buried piping and tanks inspection program consists of preventive and condition monitoring measures to manage the aging effects for buried piping, conduit and tanks in scope for license renewal. Buried components in scope for license renewal include carbon steel piping, bolting, conduit and tanks (loss of material due to general, crevice, galvanic, MIC and pitting corrosion) and cast iron piping (loss of material due to general, crevice, galvanic, MIC and pitting corrosion and selective leaching). Preventive measures consist of protective coatings and/or wraps on buried components. Condition monitoring consists of periodic inspections of buried components.

In addition, buried components are not routinely uncovered during maintenance activities. Therefore, other system monitoring and functional testing activities are relied upon to provide effective degradation aging management of buried piping and tanks. Some of these activities are neither preventive nor mitigative in nature, but they do provide indication of a leak. However, the potential problem is detected at an early stage, i.e. small leak, such that repairs can be made prior to loss of component intended function.

Commitments

Prior to the period of extended operation:

1) The Buried Piping and Tanks Inspection Program will update the implementing procedures to include inspections of buried components when they are uncovered.
2) The Diesel Fuel Oil Storage Tank, T-44, internal inspection will be added to the list of scheduled inspections in the Buried Piping and Tanks Inspection Program.
3) The Buried Piping and Tanks Inspection Program will be revised to include a provision that if evaluations of pipe wall thickness show a susceptibility to corrosion, further evaluation as to the extent of susceptibility will be performed.
4) The Buried Piping and Tanks Inspection Program will be revised to specify a 10-year buried pipe inspection frequency.
5) The Buried Piping and Tanks Inspection Program will be revised to specify a 10-year inspection frequency for Diesel Fuel Oil Storage Tank T-44.
6) The Buried Piping and Tanks Inspection Program will be revised to include a review of previous buried piping issues to determine possible susceptible locations.

A2.1.6 Bus Duct Inspection Program

Program Description

The purpose of this aging management program is to demonstrate, for in-scope non-segregated bus ducts, that the aging effects caused by ingress of moisture or contaminants (dust and debris), insulation degradation caused by heat or radiation in the presence of oxygen, and bolt relaxation caused by thermal cycling will be adequately managed so that there is reasonable assurance that the non-segregated bus ducts will perform their intended function in accordance with the current licensing basis during the period of extended operation. The intended function of non-segregated bus ducts is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.

Industry operating experience indicates that the failure of bus ducts is caused by the cracking of bus bar insulation (bus sleeving) combined with the accumulation of moisture or debris. Cracked insulation in the presence of moisture or debris provides phase-to-phase or phase-to-ground electrical tracking paths, which can result in catastrophic failure of the buses.

Bus ducts exposed to appreciable ohmic heating during operation may experience loosening of bolted connections because of the repeated cycling of connected loads. This phenomenon can occur in heavily loaded circuits, i.e., those exposed to appreciable ohmic heating. Sandia 96-0344 identified instances of bolted connection loosening at several plants due to thermal cycling. NRC Information Notice 2000-14 identified torque relaxation of splice plate connecting bolts as one potential cause of a bus duct fault.

The primary objective of the aging management program is to provide an inspection of bus ducts. Non-segregated bus duct insulation aging degradation from ingress of moisture or contaminants (dust and debris), or heat or radiation in the presence of oxygen causes insulation surface anomalies. In managing this aspect of the aging management program, visual inspection of interior portions of bus ducts will be performed to identify aging degradation of insulating and metallic components and water/debris intrusion. The external portions of bus ducts and structural supports will be inspected in accordance with a plant specific structural monitoring program. Additionally, bus ducts exposed to appreciable ohmic heating during operation may experience loosening of bolted connections. In managing this aspect of the aging management program, bolted connections at sample sections of the buses in the bus ducts will be checked for proper torque, or the bolted joints will be checked to ensure low resistance.

The purpose of the aging management program is to provide reasonable assurance that the intended functions of nonsegregated bus ducts that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by the ingress of moisture, contaminants (dust and debris), insulation degradation caused by heat or radiation in the presence of oxygen, and bolt relaxation caused by thermal cycling will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information provided in Information Notice No. 89-64.

Commitments

Prior to the period of extended operation, the Bus Duct Inspection Program will be implemented consistant with the appropriate ten elements described in Appendix A of NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.

A2.1.7 BWR Control Rod Drive Return Line Nozzle

Program Description

The MNGP BWR Control Rod Drive Return Line Nozzle Program is part of the MNGP ASME Section XI In-Service Inspection Program. The BWR Control Rod Drive Return Line Nozzle Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and provides for condition monitoring of the BWR Control Rod Drive Return Line (CRDRL) nozzle.

In 1977 the CRDRL nozzle safe end was removed and the CRDRL nozzle was capped. In 1986 the CRDRL nozzle was modified again by removing the portion of the existing weld butter layer susceptible to IGSCC, by re-cladding the weld prep area with corrosion resistant cladding, and by installing a new nozzle cap of non-IGSCC susceptible stainless steel. As a result of capping the CRDRL nozzle, the NUREG-0619 augmented examinations are no longer required. Not performing the NUREG-0619 augmented examinations is considered a NUREG-1801 XI.M6 program exception.

The program is updated periodically as required by 10 CFR 50.55a.

A2.1.8 BWR Feedwater Nozzle

Program Description

The MNGP BWR Feedwater Nozzle Program is part of the MNGP ASME Section XI In-Service Inspection Program. The BWR Feedwater Nozzle Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda with Appendix VIII. The program provides for condition monitoring of the BWR feedwater nozzles. The BWR feedwater nozzles were all repaired in 1977 and the safe ends were all replaced in 1981 with a tuning fork design with a welded-in thermal sleeve.

The BWR Feedwater Nozzle Program is not currently augmented by the recommendations of General Electric (GE) NE-523-A71-0594, Alternate BWR Feedwater Nozzle Inspection Requirement. The program will be enhanced by including the recommendations of the GE NE-523-A71-0594-A, Revision 1.

The Program is updated periodically as required by 10 CFR 50.55a.

Commitments

Prior to the period of extended operation, the BWR Feedwater Nozzle Program will be enhanced so:

1) The parameters monitored and inspected are consistent with the recommendations of GE NE-523-A71-0594-A, Revision 1.
2) The regions being inspected, examination techniques, personnel qualifications, and inspection schedule are consistent with the recommendations of GE NE-523-A71-0594-A, Revision 1.
3) That inspections will be scheduled per recommendations of GE NE-523-A71-0594-A, Revision 1.

A2.1.9 BWR Penetrations

Program Description

The MNGP BWR Penetrations Program is part of the MNGP ASME Section XI In-Service Inspection Program. The BWR Penetrations Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda (with approved ISI Relief Requests) and provides for condition monitoring of the BWR penetrations.

The BWR water chemistry is controlled per the EPRI guidelines of BWRVIP-130 (TR-1008192) BWR Water Chemistry Guidelines - 2004 Revision. BWRVIP-130 supersedes previous revisions of the guidelines, including BWRVIP-29 (TR-103515), BWR Water Chemistry Guidelines - 1993 Revision.

Program activities incorporate the inspection and evaluation guidelines of BWRVIP-49, BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines, for instrument penetrations and BWRVIP-27, BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate DP Inspection and Flaw Evaluation Guidelines, for the Standby Liquid Control System.

The program is updated periodically as required by 10 CFR 50.55a and the BWRVIP.

A2.1.10 BWR Stress Corrosion Cracking

Program Description

The Monticello Nuclear Generating Plant BWR Stress Corrosion Cracking Program is an existing program and is part of the MNGP ASME Section XI In-Service Inspection Program. ASME Section XI is being implemented with ultrasonic (UT) volumetric, surface, and visual inspections and the Risk-Informed ISI Program. NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, and its Supplement 1 are part of the MNGP BWR Stress Corrosion Cracking Program. All IGSCC susceptible materials have been replaced or protected with a cladding of resistant weld material.

Therefore, all piping welds are now classified as IGSCC Category A in accordance with NUREG-0313 and GL 88-01. As part of the MNGP recirculation piping replacement effort, austenitic stainless steel portions of piping systems 4" in nominal diameter or larger operating at temperatures above 200°F of the reactor coolant pressure boundary were replaced in accordance with the requirements of NUREG-0313.

In addition, a Hydrogen Water Chemistry System was placed in operation, which reduces the oxidizing environment by introducing excess hydrogen to the reactor coolant system that combines with the free oxygen produced by radiolysis.

A2.1.11 BWR Vessel ID Attachment Welds

Program Description

The MNGP BWR Vessel ID Attachment Welds Program is part of the MNGP ASME Section XI In-Service Inspection Aging Management Program. The BWR Vessel ID Attachment Weld Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and approved ISI Relief Requests. The program provides for condition monitoring of the BWR vessel ID attachment welds. The program includes inspection and flaw evaluation in accordance with BWRVIP-48, Vessel ID Attachment Weld and Inspection and Flaw Guidelines (EPRI TR-108724).

The BWR water chemistry is controlled per the EPRI guidelines of BWRVIP-130 (TR-1008192) BWR Water Chemistry Guidelines - 2004 Revision. BWRVIP-130 supersedes previous revisions of the guidelines, including BWRVIP-29 (TR-103515, 1993 Revision) for water chemistry in BWRs. This is considered an exception to the NUREG-1801 Program Description.

The Program is updated periodically as required by 10 CFR 50.55a. In addition the Program is supplemented by implementing the guidelines of Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents.

A2.1.12 BWR Vessel Internals

Program Description

The MNGP BWR Vessel Internals Program is part of the MNGP ASME Section XI In-Service Inspection Program. The BWR Vessel Internals Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and approved ISI Relief Requests. The program provides for condition monitoring of the BWR vessel internals for crack initiation and growth.

MNGP activities include the in-vessel examination procedures and the plant water chemistry procedures. The in-vessel examination procedures implement the recommendations of the BWRVIP guidelines, as well as the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below system-specific limits based on the EPRI guidelines of BWRVIP-130 (TR-1008192) BWR Water Chemistry Guidelines - 2004 Revision. BWRVIP-130 supersedes previous revisions of the guidelines, including BWRVIP-29 (TR-103515, 1993 Revision) for water chemistry in BWRs.

The Program is updated periodically as required by 10 CFR 50.55a and the BWRVIP Program.

Commitments

Prior to the period of extended operation, the repair/replacement guidelines in BWRVIP-16, 19, 44, 45, 50, 51, 52, 57, and 58 will be added, as applicable, to the MNGP BWR Vessel Internals Program.

During the period of extended operation, NMC will perform top guide grid inspections using the EVT-1 method of examination, for the high fluence locations (grid beam and beam-to-beam crevice slot locations with fluence exceeding 5.0×10^{20} n/cm^2). Ten percent (10%) of the total population will be inspected within 12 years with a minimum of 5% inspected within the first 6 years.

A2.1.13 Closed-Cycle Cooling Water System

Program Description

The MNGP Closed-Cycle Cooling Water System Program includes: (1) preventive measures to minimize corrosion, and (2) periodic system and component performance testing and inspection to monitor the effects of corrosion and confirm intended functions are met. Preventive measures include the monitoring and control of corrosion inhibitors and other chemical parameters, such as pH, in accordance with the guidelines of Electric Power Research Institute (EPRI) TR-1007820, Closed Cooling Water Chemistry Guideline, vendor recommendations, and plant operating experience. EPRI TR-1007820 is the current revision (Revision 1) of EPRI-107396. As only minor changes were made to the MNGP Closed-Cycle Cooling Water System Program to implement EPRI TR-1007820, the program is also still in accordance with the EPRI Revision 0 guidelines identified in NUREG-1801, Chapter XI Program M21, i.e., EPRI TR-107396, Closed Cooling Water Chemistry Guidelines. Periodic inspection and testing to confirm function and monitor corrosion is also performed in accordance with EPRI TR-1007820, vendor recommendations, and industry and plant operating experience. A review of plant operating experience demonstrates these measures ensure closed-cycle cooling water (CCCW) systems are performing their intended functions.

The MNGP has four systems in License Renewal Scope that meet the definition for consideration as closed-cycle cooling water systems and portions of three additional systems (heat exchangers or coolers) that are serviced directly by these cooling water systems. These systems and portions of systems are not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to a heat sink. The adequacy of chemistry control is confirmed on a routine basis by sampling and monitoring to within established limits and by equipment performance monitoring to identify aging effects.

Corrosion inhibitor concentrations are maintained within limits based on a combination of EPRI TR-1008720 guidelines, vendor recommendations, and plant experience. System and component performance test results are evaluated in accordance with the guidelines of EPRI TR-1008720 and used as a basis for evaluating the effectiveness of actions to mitigate cracking, corrosion, and heat exchanger fouling. Acceptance criteria and tolerances are also based on system design parameters and functions. For chemical parameters monitored, many are based on ranges identical to or more restrictive than noted in both EPRI TR-1008720 and EPRI TR-107396. Others are based on vendor recommendations and plant experience.

Frequency of performance and functional tests are consistent with EPRI TR-1008720 and are based on plant operating experience, trends and equipment performance. System and component operability tests are typically performed on a more frequent basis than once per cycle whereas more intrusive inspections (disassembly, eddy current testing, etc.) are performed less frequently but at sufficient intervals to detect the impact of aging effects on component function.

Commitments

Prior to the period of extended operation, a one time inspection will be performed to monitor the effects of corrosion on select portions of closed-cycle cooling water systems that perform a pressure-integrity intended function.

A2.1.14 Compressed Air Monitoring

Program Description

The MNGP Compressed Air Monitoring Program consists of inspection, monitoring, and testing of the Instrument and Service Air System to provide reasonable assurance that they will perform their intended function for the duration of extended operation.

Commitments

Prior to the period of extended operation, the Compressed Air Monitoring Program procedures will be revised to include corrective action requirements if the acceptance limits for water vapor, oil content, or particulate are not met. Also, the acceptance criteria for oil content testing will be clarified and the basis for the acceptance limits for the water vapor, oil content, and particulate tests will be provided.

Prior to the period of extended operation, the Compressed Air Monitoring Program will be revised to include inspection of air distribution piping based on the recommendations of EPRI TR-108147.

A2.1.15 Electrical Cables & Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program

Program Description

The MNGP Electrical Cables & Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program is a new program that manages the aging of conductor insulation material on cables, connectors, and other electrical insulation materials that are installed in an adverse localized environment caused by heat, radiation, or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the component. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

In most areas of the plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the plant design environment for those areas. Cable and connection insulation materials may degrade more rapidly than expected in these adverse localized environments.

As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the electrical cables and connections covered by this aging management program are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.

The scope of this program includes accessible non-EQ electrical cables and connections, including control and instrumentation circuits, within the scope of license renewal.

The program provides reasonable assurance that the intended functions of electrical cables and connections within scope of license renewal that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205-2000, SAND96-0344, and EPRI TR-109619.

The program addresses cables and connections whose configuration is such that most cables and connections installed in adverse localized environments are accessible. This program is a sampling program in which selected cables and connections from accessible areas are inspected and represent, with reasonable assurance, all cables and connections in the adverse localized environments. If an unacceptable condition or situation is identified for a cable or connection in the inspection sample, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections.

Commitments

Prior to the period of extended operation, the MNGP Electrical Cables & Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program XI.E1. The program will manage the aging of conductor insulation material on cables, connectors, and other electrical insulation materials that are installed in an adverse localized environment caused by heat, radiation, or moisture.

A2.1.16 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits

Program Description

This program applies to non-EQ electrical cables used in radiation monitoring and nuclear instrumentation circuits with sensitive, low-level signals that are within scope of license renewal and are installed in adverse localized environments caused by heat, radiation and moisture in the presence of oxygen.

In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the plant design environment for those areas. Conductor insulation materials used in electrical cables may degrade more rapidly than expected in these adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

Exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument loop.

The purpose of the aging management program is to provide reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are used in circuits with sensitive, low-level signals exposed to adverse localized environments caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.

In this aging management program, routine calibration tests performed as part of the plant surveillance test program are used to identify the potential existence of aging degradation. When an instrumentation loop is found to be out of calibration during routine surveillance testing, troubleshooting is performed on the loop, including the instrumentation cable.

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, NMC will perform cable system testing. A proven cable system test for detecting deterioration of the insulation system (such as insulation resistance tests, time domain reflectometry test, or other testing judged to be effective in determining cable insulation condition) will be performed.

As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the electrical cables covered by this aging management program are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.

Commitments

Prior to the period of extended operation, the Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will be implemented as a new program. With exceptions, it will be consistent with the recommendations of NUREG-1801 Chapter XI, Program XI.E2.

A2.1.17 Fire Protection

Program Description

For license renewal purposes the MNGP Fire Protection Program includes a fire barrier inspection program, a diesel-driven fire pump inspection program, and a halon fire suppression system inspection.

The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of associated fire rated doors to ensure that their operability is maintained.

The diesel-driven fire pump inspection program requires that the pump be periodically tested and the diesel engine inspected to ensure that the fuel supply line can perform the intended function.

The halon fire suppression system inspection included periodic inspection and testing of the cable spreading room halon fire suppression system.

Commitments

Prior to the period of extended operation:

1) The MNGP Fire Protection Program will be revised to include a visual inspection of the halon fire suppression system to detect any signs of degradation, such as corrosion and mechanical damage. This visual inspection will provide aging management for external surfaces of the halon fire suppression system; and

2) The MNGP Fire Protection Program will be revised to include qualification criteria for individuals performing visual inspections of penetration seals, fire barriers, and fire doors. The qualification criteria will be in accordance with VT-1 or equivalent and VT-3 or equivalent, as applicable.

A2.1.18 Fire Water System

Program Description

The Fire Water System aging management program relies on testing of water based fire protection system piping and components in accordance with applicable NFPA recommendations. In addition, this program will be modified to include (1) portions of the fire protection sprinkler system that are subjected to full flow tests prior to the period of extended operation and (2) portions of the fire protection system exposed to water that are internally visually inspected. To ensure that the aging mechanisms of corrosion, and biofouling/fouling are properly being managed in the fire water system, periodic full flow flush test and system performance test are conducted. The system is also normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated.

Commitments

Prior to the period of extended operation, the Fire Water System Program:

1) Implementing procedures will be revised to include the extrapolation of inspection results to below grade fire water piping with similar conditions that exist within the above grade fire water piping, and
2) Sprinkler heads will be inspected and tested per NFPA requirements or replaced before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

Will verify procedures to be used for aging management activities of the Fire Water System apply testing in accordance with applicable NFPA codes and standards. Revise the relevant procedures as appropriate.

A2.1.19 Flow-Accelerated Corrosion

Program Description

The Flow-Accelerated Corrosion Program manages aging effects (loss of material) due to flow-accelerated corrosion (FAC) on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phase). The program implements the EPRI guidelines in NSAC-202L-R2. This program also requires the use of CHECWORKS as a predictive tool. Included in the program are: (a) an analysis to determine FAC susceptible locations; (b) performance of limited baseline inspections; (c) follow-up inspections to confirm the predictions; and (d) repairing or replacing components, as necessary.

The MNGP Flow-Accelerated Corrosion Program includes the response made to GL 89-08, Erosion/Corrosion Induced Pipe Wall Thinning.

A2.1.20 Fuel Oil Chemistry

Program Description

The Monticello Nuclear Generating Plant (MNGP) Fuel Oil Chemistry Program is an existing program using existing diesel fuel oil system procedures that encompass the NUREG-1801 program recommendations. The Fuel Oil Chemistry Program mitigates and manages aging effects on the internal surfaces of diesel fuel oil storage tanks and associated components in systems that contain diesel fuel oil. The program includes (a) surveillance and monitoring procedures for maintaining diesel fuel oil quality by controlling contaminants in accordance with applicable ASTM Standards; (b) periodic draining of water from diesel fuel oil tanks, if water is present; (c) periodic or conditional visual inspection of internal surfaces or wall thickness measurements (e.g., by UT) from external surfaces of diesel fuel oil tanks; and (d) one-time inspections of a representative sample of components in systems that contain diesel fuel oil.

Commitments

Prior to the period of extended operation:

1) The MNGP procedures related to the Diesel Fuel Oil System will be revised to include requirements to check for general, pitting, crevice, galvanic, microbiological influenced corrosion (MIC), and cracking.
2) The MNGP Fuel Oil Chemistry Program procedures will be revised to require tank draining, cleaning, and inspection if deemed necessary based on the trends indicated by the results of the diesel fuel oil analysis, or as recommended by the system engineer based on equipment operating experience.

Develop or revise existing procedures in the MNGP Fuel Oil Chemistry Program to require periodic tank inspections of the diesel fuel oil tanks.

A2.1.21 Inaccessible Medium Voltage (2kV to 34.5kV) Cables Not Subject to 10 CFR 50.49 EQ Requirements

Program Description

The purpose of this aging management program is to demonstrate that inaccessible, non-EQ medium-voltage cables susceptible to aging effects caused by moisture and voltage stress will be adequately managed so that there is reasonable assurance that the cables will perform their intended function in accordance with the current licensing basis during the period of extended operation. The intended function of insulated cables and connections is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.

Most electrical cables at MNGP are located in dry environments. However, some cables may be exposed to condensation and wetting in inaccessible locations, such as conduits, cable trenches, cable troughs, duct banks, underground vaults or direct buried installations. When an energized medium-voltage cable is exposed to wet conditions for which it is not designed, water treeing or a decrease in the dielectric strength of the conductor insulation can occur. This can potentially lead to electrical failure.

In this aging management program, periodic actions are taken to prevent cables from being exposed to significant moisture, such as inspecting for water collection in cable manholes and conduit, and draining water, as needed. In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed.

Commitments

Prior to the period of extended operation, the MNGP Inaccessible Medium Voltage (2 kV to 34.5 kV) Cables Not Subject to 10 CFR 50.49 EQ Requirements Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program XI.E3.

A2.1.22 Inspection of Overhead Heavy Load & Light Load (Related to Refueling) Handling Systems

Program Description

The Inspection Of Overhead Heavy Load & Light Load (Related To Refueling) Handling Systems program, which is implemented through plant procedures and preventive maintenance, manages loss of material of structural components for heavy load and fuel handling components within the scope of license renewal. The Inspection Of Overhead Heavy Load & Light Load (Related To Refueling) Handling Systems program provides for visual and NDE inspections of in-scope load handling components. Functional tests are also performed to assure their integrity. The cranes also comply with the maintenance rule requirements provided in 10 CFR 50.65.

Commitments

Prior to the period of extended operation, the Inspection of Overhead Heavy Load & Light Load (Related to Refueling) Handling Systems Program will be enhanced to specify a five-year inspection frequency for the fuel preparation machines.

A2.1.23 One-Time Inspection

Program Description

The MNGP One-Time Inspection Program is a new program that is being developed consistent with NUREG-1801 Chapter XI Program M32, "One-Time Inspection." Any exceptions or enhancements to NUREG-1801 will be described in the relevant element descriptions. This program includes measures to verify the effectiveness of the following aging management programs:

- Plant Chemistry Program
- Fuel Oil Chemistry Program

This program also confirms the absence of age degradation in selected components (e.g., flow restrictors, venturis, and small bore piping) within License Renewal scope.

The MNGP One-Time Inspection Program addresses concerns and confirmation for the potential long incubation period for certain aging effects on structures and components.

There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly.

The activities of the One-Time Inspection Program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; (b) identification of the inspection locations in the system or component based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and (d) evaluation of the need for follow-up examinations to monitor the progression of any identified aging degradation.

The program will manage the aging effects due to corrosion, cracking, erosion, fouling, fretting, or thermal exposure. The program will also verify the absence of reduction of neutron absorption capacity of boral in the spent fuel pool.

Commitments

Prior to the period of extended operation, the MNGP One-Time Inspection Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program XI.M32, "One-Time Inspection." This program will include measures to verify the effectiveness of the following aging management programs: Plant Chemistry Program and Fuel Oil Chemistry Program. This program will also confirm the absence of age related degradation in selected components (e.g., flow restrictors, venturis) within License Renewal scope.

A2.1.24 Open-Cycle Cooling Water System

Program Description

The MNGP Open-Cycle Cooling Water System Program relies on the implementation of the recommendations of NRC Generic Letter (GL) 89-13 to ensure that the effects of aging on the raw water service water systems will be managed for the period of extended operation. This program manages the aging effects of metallic components in water systems (e.g., piping and heat exchangers) exposed to raw, untreated (e.g., service) water. These aging effects are due to corrosion, erosion, and biofouling in systems, structures and components serviced by the OCCW system. The program includes (a) surveillance and control of biofouling; (b) tests to verify heat transfer; and (c) routine inspection and maintenance.

The MNGP Open-Cycle Cooling Water System Program complies with MNGP's response to NRC GL 89-13. Resultant commitments made to comply with GL 89-13 have been incorporated into plant procedures and programs.

A2.1.25 Plant Chemistry Program

Program Description

The MNGP Plant Chemistry Program mitigates the aging effects on component surfaces that are exposed to water as the process fluid; chemistry programs are used to control water chemistry for impurities (e.g., chloride and sulfate) that accelerate corrosion or crack initiation and growth and that cause heat transfer degradation due to fouling in select heat exchangers. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below system-specific limits based on BWRVIP-130 (EPRI TR-1008192): BWR Water Chemistry Guidelines - 2004 Revision.

BWRVIP-130 supersedes previous revisions of the BWR Water Chemistry Guidelines including BWRVIP-29 (TR-103515, 1993 Revision).

For low-flow or stagnant portions of a system, a one-time inspection of selected components at susceptible locations provides verification of the effectiveness of the Plant Chemistry Program.

A2.1.26 Primary Containment In-Service Inspection Program

Program Description

The MNGP Primary Containment In-Service Inspection Program requires visual examinations of the accessible surfaces (base metal and welds) of the drywell, torus, vent lines, internal vent system, penetration assemblies and associated integral attachments. The program also requires examination of pressure retaining bolting and the drywell interior slab moisture barrier.

The program conforms to the applicable requirements of 10CFR50.55a and the 1992 Edition with 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Subsection IWE.

A detailed VT-3 and VT-1 examination is performed once during each 10-year in-service inspection interval. This examination is performed either at the end of the interval or spread across the three periods that comprise the interval. General visual examinations that assess overall structural condition are performed once during each period.

Surface and / or volumetric examination augments visual examination as required to define the extent of observed conditions or to identify deterioration at inaccessible locations.

Limited scope examinations are performed as required to evaluate disassembled bolting and the condition of the normally submerged torus surface when the suppression pool is drained.

The program is updated periodically as required by 10 CFR50.55a.

A2.1.27 Protective Coating Monitoring & Maintenance Program

Program Description

The MNGP Protective Coating Monitoring and Maintenance Program applies to Service Level 1 protective coatings inside containment to address the concerns of NRC GL 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Cooling Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment. The Protective Coating Monitoring and Maintenance Program prevents the degradation of coatings that could lead to the clogging of ECCS suppression pool suction strainers. MNGP does not credit the Protective Coating Monitoring and Maintenance Program for the prevention of corrosion of carbon steel components.

As outlined in MNGP's response to GL 98-04, the Protective Coating Monitoring and Maintenance Program is a comparable program for monitoring and maintaining protective coatings inside the primary containment and subject to the requirements of ANSI N101.4-1972, to the extent specified in ANSI N18.7-1976 and as modified by Regulatory Guide 1.54, June 1973.

Commitments

Prior to the period of extended operation, the MNGP Protective Coating Maintenance and Monitoring Program:

1) Procedures will be updated to include Inspection of all accessible painted surfaces inside containment;
2) Will be revised to include a pre-inspection review of the previous two inspection reports so that trends can be identified; and
3) Implementation procedures will be revised to include provisions for analysis of suspected reasons for coating failure.

A2.1.28 Reactor Head Closure Studs

Program Description

The MNGP Reactor Head Closure Studs Program is part of the MNGP ASME Section XI In-Service Inspection Program. The Reactor Head Closure Studs Program is in accordance with ASME Section XI 1995 Edition through the 1996 Addenda and provides for condition monitoring of the reactor head closure stud bolting. Replacement reactor head studs available for use at Monticello include preventive measures described in RG 1.65, Material and Inspection for Reactor Vessel Closure Studs.The Program is updated periodically as required by 10 CFR 50.55.a.

A2.1.29 Reactor Vessel Surveillance

Program Description

The MNGP Reactor Vessel Surveillance Program is part of the Boiling Water Reactor's Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP) that uses data from BWR member surveillance programs to select the "best" representative material to monitor radiation embrittlement for a particular plant. The BWRVIP ISP monitors capsule test results from various member plants. This is consistent with the methodology allowed by NUREG-1801.

The MNGP Reactor Vessel Surveillance Program is required by 10 CFR 50, Appendix H. The scope of the Reactor Vessel Surveillance Program is described by the BWRVIP ISP guidance. The ISP capsule removal schedule is included in BWRVIP-86-A and its technical basis is described in BWRVIP-78.

The NRC in a Safety Evaluation (SE) to the BWRVIP, dated February 1, 2002, approved the ISP. This Safety Evaluation concluded that the ISP, if implemented in accordance with the conditions in the SE, is an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10 CFR Part 50 through the end of current facility 40 year operating licenses.

Commitments

NMC intends to use the Integrated Surveillance Program for MNGP during the period of extended operation by implementing the requirements of BWRVIP-116, which is currently being reviewed by the NRC.

Prior to and during the period of extended operation, NMC will retain the capsules removed from the MNGP reactor vessel as part of the Reactor Vessel Surveillance Program.

A2.1.30 Selective Leaching of Materials

Program Description

The MNGP Selective Leaching of Materials Program will be a new program, developed and implemented before the start of the period of extended operation. The program includes a one-time visual inspection and hardness measurement of selected components that are susceptible to selective leaching. In situations where hardness testing is not practical, a qualitative method by other NDE or metallurgical methods will be used to determine the presence and extent of selective leaching. The program will determine if selective leaching is occurring for selected components.

Any required instructions or procedures will be written during development of the program. Existing MNGP procedures or work instructions may be used.

Commitments

Prior to the period of extended operation, the MNGP Selective Leaching of Materials Program will be implemented as a new program consistent, with exceptions, to the recommendations of NUREG-1801 Chapter XI Program XI.M33, "Selective Leaching of Materials." The program will be developed and implemented before the start of the period of extended operation. The program includes a one-time visual inspection and hardness measurement of selected components that are susceptible to selective leaching. In situations where hardness testing is not practical, a qualitative method by other NDE or metallurgical methods will be used to determine the presence and extent of selective leaching. The program will determine if selective leaching is occurring for selected components.

A2.1.31 Structures Monitoring Program

Program Description

The MNGP Structures Monitoring Program provides for aging management of structures and structural components within the scope of license renewal and implements the NUREG-1801, XI.S6, Structures Monitoring Program. The Structures Monitoring Program is based on the guidance provided in RG 1.160 and NUMARC 93-01. The Structures Monitoring Program is implemented as part of the structures monitoring done under the MNGP Maintenance Rule Program and with additional inspections of the intake structure and diesel fuel oil transfer house.

The Structures Monitoring Program also implements the NUREG-1801, XI.S5, Masonry Wall Program. Masonry block wall inspections are performed as part of the maintenance rule inspections and are based on IEB 80-11 with administrative controls per IN 87-67.

As permitted by NUREG-1801, XI.S7, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, the inspection of water control structures is included in the Structures Monitoring Program. The only water control structure in scope for license renewal is the intake structure. Maintenance rule inspections are performed on the portions of the intake structure above the water line. The Structures Monitoring
Program includes separate inspections of the underwater portions of the intake structure.

In addition, special settlement checks of the diesel fuel oil transfer house are performed outside the maintenance rule inspections.

The Structures Monitoring Program does not rely upon protective coatings to manage the effects of aging.

Commitments

Prior to the period of extended operation, the MNGP Structures Monitoring Program:

1) Will be expanded, as necessary, to include inspections of structures and structural elements in scope for License Renewal that are not inspected as part of another aging management program.
2) Implementing procedures will be enhanced to ensure that structural inspections are performed on submerged portions of the intake structure from the service water bays to the wing walls;
3) Implementing procedures will be revised to include the monitoring/inspection parameters for structural components within the scope of License Renewal;
4) Will be enhanced to include a requirement to sample ground water for pH, chloride concentration and sulfate concentration;
5) Will be enhanced to include concrete evaluations of inaccessible areas if degradation of accessible areas is detected; and
6) Implementing procedures will be enhanced to include acceptance criteria for structural inspections of submerged portions of the Intake Structure.

A2.1.32 System Condition Monitoring Program

Program Description

The System Condition Monitoring Program is an existing plant-specific program that is based on system engineer monitoring. Although many monitoring activities are being performed at MNGP, this AMP brings aging management into the scope of the monitoring activities. Other groups augment this program by identifying and reporting adverse material conditions via the corrective action process or work control process. This monitoring consists of system-level performance monitoring, inspections and walkdowns, health and status reporting, and preventive maintenance. This program will be enhanced to include specific activities and criteria for managing age related degradation for SSCs within License Renewal scope. This program manages aging effects for normally accessible external surfaces of piping, tanks, hangers/supports, racks, panels, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation.

Commitments

Prior to the period of extended operation, the implementing instructions and procedures for the MNGP System Condition Monitoring Program will be revised to describe specific age degradation parameters to be monitored and inspected. Acceptance criteria will also be included.

A2.1.33 Thermal Aging & Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

Program Description

The MNGP Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program monitors the aging effects of loss of fracture toughness on the intended function of the component by performing examinations on CASS reactor vessel internal components as part of the MNGP ASME Section XI In-Service Inspection Program. The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is in accordance with ASME Section XI, Subsection IWB, Category B-N-1 and B-N-2 requirements and provides for condition monitoring of the CASS components. Additional enhanced visual inspections that incorporate the requirements of the BWRVIP are performed to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals.

The program is updated periodically as required by 10 CFR 50.55a.

A3 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 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.

Where used herein, the term "rerate" refers to the MNGP Power Rerate Program, which resulted in an increase in rated thermal power from 1670 MWt to 1775 MWt (approximately 6.3 percent). The increase in rated thermal power was implemented at MNGP in 1998. To demonstrate margin, most analyses performed for the power rerate conservatively used a power level of 1880 MWt. The continued use of this conservatism is described, where appropriate, in the following TLAA evaluations.

A3.1 Neutron Embrittlement of the Reactor Pressure Vessel and Internals

The materials of the reactor pressure vessel (RPV) and internals are subject to embrittlement due to high energy (E > 1 MeV) neutron exposure. Embrittlement means the material has lower toughness (i.e., will absorb less strain energy during a crack or rupture), thus allowing a crack to propagate more easily under thermal and/or pressure loading.

Toughness (indirectly measured in foot-pounds of absorbed energy in a Charpy impact test) is temperature-dependent in ferritic materials. An initial nil-ductility reference temperature (RTNDT), the temperature associated with the transition from ductile to brittle behavior, is determined for vessel materials through a combination of Charpy and drop weight testing. Toughness increases with temperature up to a maximum value called the "upper-shelf energy" (USE). Neutron embrittlement causes an increase in the RTNDT and a decrease in the USE of RPV steels. The increase or shift in the initial nil-ductility reference temperature (RTNDT) means higher temperatures are required for the material to continue to act in a ductile manner.

To reduce the potential for brittle fracture during RPV operation by accounting for the changes in material toughness as a function of neutron radiation exposure (fluence), operating pressure-temperature (P-T) limit curves are included in plant Technical Specifications. The P-T curves account for the decrease in material toughness associated with a given fluence, which is used to predict the loss in toughness of the RPV materials. Based on the projected drop in toughness for a given fluence, the P-T curves are generated to provide a minimum temperature limit associated with the vessel pressure. The P-T curves are determined by the RTNDT and ΔRTNDT RTNDT values for the licensed operating period along with appropriate margins.

The RPV ΔRT_{NDT} and USE, calculated on the basis of neutron fluence, are part of the licensing basis and support safety determinations. Therefore, these calculations are

Time-Limited Aging Analyses (TLAAs). The increases in RTNDT (ΔRT_{NDT}) affect the bases for relief from circumferential weld inspection and their associated supporting calculation of limiting axial weld conditional failure probability. As such, circumferential weld examination relief and axial weld failure probability are also TLAAs. Section A3.1 includes the following TLAA discussions related to the issue of neutron embrittlement:

- RPV Materials USE Reduction Due to Neutron Embrittlement
- Adjusted Reference Temperature (ART) for RPV Materials Due to Neutron Embrittlement
- Reflood Thermal Shock Analysis of the RPV
- Reflood Thermal Shock Analysis of the RPV Core Shroud
- RPV Thermal Limit Analysis: Operating P-T Limits
- RPV Circumferential Weld Examination Relief
- RPV Axial Weld Failure Probability

RPV Materials USE Reduction Due to Neutron Embrittlement

Summary Description

USE is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. 10 CFR 50 Appendix G requires the predicted end-of-life Charpy impact test USE for RPV materials to be at least 50 ft/lb (absorbed energy), unless an approved analysis supports a lower value. Initial unirradiated test data are available for only one plate heat for the MNGP RPV to demonstrate a minimum 50 ft-lb USE by standard methods. End-of-life fracture energy was evaluated by using an equivalent margin analysis (EMA) methodology approved by the NRC for all other materials (Reference 1). This analysis confirmed that an adequate margin of safety against fracture, equivalent to 10 CFR 50 Appendix G requirements, does exist. The end-of-life USE calculations satisfy the criteria of 10 CFR 54.3(a) (Reference 2). As such, these calculations are a TLAA.

Analysis

The MNGP RPV was designed for a 40-year life with an assumed neutron exposure of less than 10^{19} n/cm^2 from energies exceeding 1 MeV. The current licensing basis calculations use realistic calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year term.

The tests performed on RPV materials under the Code of Record provided limited Charpy impact data. It was possible to develop original Charpy impact test USE values for only one plate material using the methods of 10 CFR 50 Appendix H and American Society For Testing and Materials (ASTM) E185 invoked by 10 CFR 50 Appendix G. Therefore, alternative methods approved by the NRC in NEDO-32205-A, have been used to demonstrate compliance with the 40-year 50 ft-lb USE requirement.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Fluence was calculated for the MNGP RPV for the extended 60-year (54 EFPY) licensed operating periods, using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," which was approved by the NRC in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GE) (Reference 3). The NRC found that, in general, this methodology adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. For MNGP, 54 EFPY is equivalent to 3.90 x 10^8 MWh through the end of Cycle 22 at 1775 MWt plus 4.76E8 MWh at 1880 MWt. Peak fluence was calculated at the RPV inner surface (inner diameter), for purposes of evaluating USE. The value of neutron fluence was also calculated for the 1/4T location into the RPV wall measured radially from the inside diameter (ID), using Equation 3 from Paragraph 1.1 of Regulatory Guide (RG) 1.99, Revision 2. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G Subarticle G-2120 as the maximum postulated defect depth.

The End of License (EOL) USE was evaluated by an EMA using the 54 EFPY calculated fluence, and MNGP surveillance capsule results. As described in the Safety Evaluation Report (SER) to BWRVIP-74 (Reference 4), the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR/2-6 welds are 23.5% and 39% respectively. Table A3.1-1 and Table A3.1-2 provide results of the EMA for limiting welds and plates on the RPV. The results show that the limiting USE EMA percent is less than the BWRVIP-74 EMA percent acceptance criterion in all cases, and is therefore acceptable. The 54 EFPY USE values are managed in conjunction with surveillance capsule results as part of the BWRVIP Integrated Surveillance Program, BWRVIP-86-A (Reference 5) and BWRVIP-116 (Reference 15).

Table A3.1-1 Equivalent Margin Analysis for MNGP Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.17 1st Capsule Fluence = 2.93×10^{17} n/cm^2 2nd Capsule Fluence = N/A 1st Capsule Measured % Decrease = N/A (Charpy Curves) 2nd Capsule Measured % Decrease = N/A (Charpy Curves) 1st Capsule RG 1.99 Predicted % Decrease = 11.5 (RG 1.99, Figure 2) 2nd Capsule RG 1.99 Predicted % Decrease = N/A (RG 1.99, Figure 2)
Limiting Beltline Plate USE: %Cu = 0.17 54 EFPY 1/4T Fluence = 3.82×10^{18} n/cm^2 RG 1.99 Predicted % Decrease = 21 (RG 1.99, Figure 2) Adjusted % Decrease = N/A (RG 1.99, Position 2.2)
21 < 23.5%, so vessel plates are bounded by EMA

Table A3.1-2 Equivalent Margin Analysis for MNGP Weld Material

BWR/3-6 WELD
Surveillance Weld USE: %Cu = 0.04 1st Capsule Fluence = 2.93×10^{17} n/cm^2 2nd Capsule Fluence = N/A 1st Capsule Measured % Decrease = N/A (Charpy Curves) 2nd Capsule Measured % Decrease = N/A (Charpy Curves) 1st Capsule RG 1.99 Predicted % Decrease = 8 (RG 1.99, Figure 2) 2nd Capsule RG 1.99 Predicted % Decrease = N/A (RG 1.99, Figure 2)
Limiting Beltline Weld USE %Cu = 0.10 54 EFPY 1/4T Fluence = 3.82×10^{18} n/cm^2 RG 1.99 Predicted % Decrease = 19.5 (RG 1.99, Figure 2) Adjusted % Decrease = N/A (RG 1.99, Position 2.2)
19.5 < 39%, so vessel welds are bounded by EMA.

Adjusted Reference Temperature (ART) for RPV Materials Due to Neutron Embrittlement

Summary Description

The initial RT_{NDT}, nil-ductility reference temperature, is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics going from ductile to brittle behavior. RTNDT was evaluated according to the procedures in the ASME Code, Paragraph NB-2331. Neutron embrittlement raises the initial nil-ductility reference temperature. 10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the vessel. The shift to the initial nil-ductility reference temperature (RT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as $RT_{NDT} + \Delta RT_{NDT}$ + margin. The margin is defined in RG 1.99. The P-T curves are developed from the ART for the RPV materials. These are determined by the unirradiated RT_{NDT} and by the ΔRT_{NDT} calculations for the licensed operating period. RG 1.99 defines the calculation methods for ΔRT_{NDT}, ART, and end-of-life USE.

The ΔRT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a). As such, they are TLAAs.

Analysis

The MNGP RPV was designed for a 40-year life with an assumed neutron exposure of less than 10^{19} n/cm^2 from energies exceeding 1 MeV (Reference 6). The current licensing basis calculations use realistic calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year term. The ΔRT_{NDT} values were determined using the embrittlement correlations defined in RG 1.99.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Fluence was calculated for the MNGP RPV for the extended 60-year (54 EFPY) licensed operating period, using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," which was approved by the NRC in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GE) (Reference 3). The NRC found that, in general, this methodology adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. For MNGP, 54 EFPY is equivalent to 3.90 x 10^8 MWh through the end of Cycle 22 at 1775 MWt plus 4.76 x 10^8 MWh at 1880 MWt. Peak fluence was calculated at the vessel inner surface (inner diameter), for purposes of evaluating USE and ART. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter (ID), using Equation 3 from Paragraph 1.1 of RG 1.99. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G Sub-article G-2120 as the maximum postulated defect depth.

The 54 EFPY ΔRT_{NDT} for all beltline materials was calculated based on the embrittlement correlation found in RG 1.99. The peak fluence, ΔRT_{NDT}, and ART values for the 60-year (54 EFPY) license operating period are presented in Table A3.1-3. This table shows that the limiting ARTs allow P-T limits that will provide reasonable operational flexibility.

The beltline region is defined as that portion of the RPV adjacent to the active fuel that attains a fluence = 1.0 x 10^{17} n/cm^2 during plant license. This extends the beltline 18" below and 168" above the bottom of active fuel (approximately 23" above the top of active fuel). As a result, the N2 Recirculation Inlet Nozzle falls within this extended beltline region, and is included in the calculation for ART in Table A3.1-3. The nozzle fluence has been adjusted by a peak/location factor of 0.137. In the absence of copper data for the N2 nozzle, this value is based upon heats of materials used for beltline nozzles at other plants (see Table A3.1-3). The nickel content has been determined as the average from all material test reports for the MNGP N2 nozzles. Additionally, the girth weld between Shell Rings 2 and 3 falls into the extended beltline region. The limiting weld values presented in Table A3.1-3 represent this girth weld in addition to the other vertical and girth welds in the beltline region.

The MNGP ΔRT_{NDT} and ART values are managed in conjunction with surveillance capsule results from the BWRVIP Integrated Surveillance Program, BWRVIP-86-A (Reference 5) and BWRVIP-116 (Reference 15).

Table A3.1-3 60 Year Analysis Results for MNGP

Lower Shell

Thickness in inches = 5.06 | Ratio Peak/Location = 0.659 | 54 EFPY Peak I.D. fluence = 3.41 x10^{18} n/cm^2

EFPY Peak 1/4 T Fluence = 2.51 x10^{18} n/cm^2

Lower-Intermediate Shell and All Welds

Thickness in inches = 5.06 | Ratio Peak/Location = 1.00 | 54 EFPY Peak I.D. fluence = 5.17 x10^{18} n/cm^2

EFPY Peak 1/4 T Fluence = 3.82 x 10^{18} n/cm^2

N2 Nozzle

Thickness in inches = 5.06 | Ratio Peak/Location = 0.137 | 54 EFPY Peak I.D. fluence = 7.08 x10^{17} n/cm^2

EFPY Peak 1/4 T Fluence = 5.23 x10^{17} n/cm^2

COMPONENT	HEAT	%Cu	%N	CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower-Intermediate												
1-14 1-15	C2220-1 C2220-2	0.17 0.17	0.65 0.65	131 131	27 27	3.82x10^{18} 3.82x10^{18}	96 96	0 0	17 17	34 34	130 130	157 157
Lower												
1-16 1-17	A0946-1 C2193-1	0.14 0.17	0.56 0.50	100 121	27 0	2.51 x10^{18} 2.51 x10^{18}	63 76	0 0	17 17	34 34	97 110	124 110
WELDS:												
Limiting	SMAW	0.10	0.99	138.5	-65.6	3.82x10^{18}	102	12.7	28	61	163	97
NOZZLES:												
N2*	E21VW	0.18	0.86	141.9	40	5.23x10^{17}	43	0	17	34	77	117

* In the absence of Cu data for this nozzle, 0.18% is based upon heats of materials used for beltline nozzles at other plants. The mean from nine nozzles (0.119) plus one standard deviation (0.0617) was used to determine the value of 0.18%. CMTR data for the ten (10) MNGP N2 nozzles was averaged to determine the Ni content. CMTR data for the ten (10) MNGP N2 nozzles was used to determine the initial RTNDT.

Reflood Thermal Shock Analysis of the RPV

Summary Description

The MNGP USAR includes an end-of-life thermal shock analysis performed on the RPV for a design basis LOCA followed by a low-pressure coolant injection. The effects of neutron embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. This analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

Analysis

For the current operating period, a thermal shock analysis was originally performed on the RPV components. The analysis assumed a design basis LOCA followed by a low-pressure coolant injection accounting for the full effects of neutron embrittlement at the end of life (40 years). The analysis showed that the total maximum vessel irradiation (1 MeV) at the mid-core inside of the vessel to be 2.4×10^{17} n/cm^2 which was below the threshold level of any nil-ductility temperature shift for the vessel material. As a result, it was concluded that the irradiation effects on all locations of the RPV could be ignored. However, this analysis only bounded 40 years of operation.

The peak fluence at the RPV wall for the MNGP RPV is 5.17×10^{18} n/cm^2 for 54 EFPY of operation (3.90×10^8 MWh through the end of Cycle 22 at 1775 MWt plus 4.76×10^8 MWh at 1880 MWt). Based on this fluence value, the previous analysis is not bounding for the period of extended operation. The original analysis has been superseded by an analysis for BWR-6 RPVs (Reference 7) that is applicable to the MNGP BWR3 RPV. The revised analysis is applicable to MNGP as it uses a bounding main steam line break event, and an RPV thickness similar to the MNGP RPV. This analysis assumes end-of-life material toughness, which in turn depends on end-of-life ART. The critical location for fracture mechanics analysis is at ¼ of the RPV thickness (from the inside, 1/4T). For this event, the peak stress intensity occurs at approximately 300 seconds after the LOCA.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The current analysis (Reference 7) assumes end-of-life material toughness, which in turn depends on end-of-life ART. The critical location for fracture mechanics analysis is at ¼ of the vessel thickness (from the inside, 1/4T). For this event, the peak stress intensity occurs at approximately 300 seconds after the LOCA.

The analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is 1/4T) is approximately 400°F. For the MNGP vessel, the 1/4T is 1.26 inches. The current analysis is bounding for MNGP for two reasons: (1) the pressure stress (higher for a thinner vessel) is near zero in a thermal shock event, and therefore can be neglected; and (2) the thermal shock event thermal stresses in a 6-inch vessel are greater than those in a 5.06-inch vessel. Figure 3 of (Reference 7) was used to determine the appropriate parameters for the thinner vessel. Figure 3 demonstrates that 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.26 inches deep is approximately 370°F. The ART values tabulated in Table A3.1-3 list the ARTs for the limiting weld metal of the MNGP RPV. The highest calculated RPV beltline material ART value is 157°F. Using the equation for K_{IC} presented in Appendix A of ASME Section XI (Reference 8) and the maximum ART value, the material reaches upper shelf (a K_{IC} value of 200 ksi$\sqrt{\text{in}}$) at 261°F, which is well below the 370°F 1/4T temperature predicted for the thermal shock event at the time of peak stress intensity. Therefore, the revised analysis is valid for the period of extended operation.

Reflood Thermal Shock Analysis of the RPV Core Shroud

Summary Description

Radiation embrittlement may affect the ability of RPV internals, particularly the core shroud to withstand a low-pressure coolant injection thermal shock transient. The analysis of core shroud strain due to reflood thermal shock is a TLAA because it is part of the current licensing basis, supports a safety determination, and is based on the calculated lifetime neutron fluence.

Analysis

The RPV core shroud was evaluated for a low-pressure coolant injection reflood thermal shock transient considering the embrittlement effects of 40-year radiation exposure (32 EFPY). The core shroud receives the maximum irradiation on the inside surface opposite the midpoint of the fuel centerline. The total integrated neutron fluence at end of life at the inside surface of the shroud is anticipated to be 2.7 $x10^{20}$ n/cm^2 (greater than 1 MeV). The maximum thermal shock stress in this region will be 155,700 psi equivalent to 0.57% strain. This strain range of 0.57% was calculated at the midpoint of the shroud, the zone of highest neutron irradiation. The calculated strain range of 0.57% represents a considerable margin of safety relative to measured values of percent elongation for annealed Type 304 stainless steel irradiated to 8 x 10^{21} n/cm^2 (greater than 1 MeV). The measured value of percent elongation for stainless steel weld metal is 4% for a temperature of 297°C (567°F) with a neutron fluence of 8 $x10^{21}$ n/cm^2 (greater than 1 MeV), while the average value for base metal at 290°C (554°F) is 20% (Reference 9). Therefore, thermal shock effects on the shroud at the point of highest irradiation level will not jeopardize the proper functioning of the shroud following the design basis accident (DBA) during the current licensed operating period (40 years).

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

As discussed above, core shroud components were evaluated for a reflood thermal shock event, considering the embrittlement effects of lifetime radiation exposure. The analysis includes the most irradiated point on the inner surface of the shroud where the calculated value of fluence for 40-year operating period as below the threshold (3.0 x 10^{20} n/cm^2) for material property changes due to irradiation. However, using the approved fluence methodology discussed above in the section entitled "Adjusted Reference Temperature (ART) for RPV Materials Due to Neutron Embrittlement," the 54 EFPY fluence at the most irradiated point on the core shroud was calculated to be 3.84 x 10^{21} n/cm^2.

Because the measured value of elongation bounds the calculated thermal shock strain amplitude of 0.57%, the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for a 60-year operating period.

RPV Thermal Limit Analyses: Operating Pressure - Temperature Limits

Summary Description

The ART is the value of (Initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at a specific location. Neutron embrittlement increases the ART. Thus, the minimum metal temperature at which an RPV is allowed to be pressurized increases. The ART of the limiting beltline material is used to correct the beltline P-T limits to account for irradiation effects.

10 CFR Part 50, Appendix G requires RPV thermal limit analyses to determine operating pressure-temperature (P-T) limits for boltup, hydrotest, pressure tests and normal operating and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, referred to as Curve A; 2) non-nuclear heat-up / cooldown and low-level physics tests, referred to as Curve B; and 3) core critical operation, referred to as Curve C. Pressure/temperature limits are developed for three vessel regions: the upper vessel region, the core beltline region, and the lower vessel bottom head region. The calculations associated with generation of the P-T curves satisfy the criteria of 10 CFR 54.3(a). As such, this topic is a TLAA.

<u>Analysis</u>

The MNGP Technical Specifications contain P-T limit curves for heat up cooldown, and in-service leakage and hydrostatic testing and also limits the maximum rate of change of reactor coolant temperature. The criticality curves provide limits for both heat up and criticality calculated for a 32 EFPY operating period. The current technical specifications contain P-T curves developed using the 1989 Edition of the ASME Boiler and Pressure Vessel Code, incorporating the effects of the 1998 power rerate and Code Case N-640. The ART remains essentially unchanged (from 156.5°F to 157°F) for the period of extended operation.

<u>Disposition: Revision, 10 CFR 54.21(c)(1)(ii)</u>

NMC manages the P-T curves in conjunction with surveillance capsule results as part of the BWRVIP Integrated Surveillance Program (BWRVIP-86-A (Reference 5) and BWRVIP-116 (Reference 15), respectively).

<u>RPV Circumferential Weld Examination Relief</u>

<u>Summary Description</u>

Relief from RPV circumferential weld examination requirements under Generic Letter (GL) 98-05 is based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on RPV metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

MNGP has received this relief for the remaining 40 year licensed operating period. The circumferential weld examination relief analysis meets the requirements of 10 CFR 54.3(a) (Reference 2). As such, they are a TLAA.

<u>Analysis</u>

MNGP received NRC approval for a technical alternative that eliminated the RPV circumferential shell weld inspections for the current license term. The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 54 EFPY and approval by the NRC to extend this relief request.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

The USNRC evaluation of BWRVIP-05 used the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities (Reference 10). Three key assumptions of the PFM analysis are: 1) the neutron fluence was the estimated end-of-life mean fluence; 2) the chemistry values are mean values based on vessel types; and 3) the potential for beyond-design-basis events is considered. Table A3.1-4 provides a comparison of the MNGP RPV limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. Data provided in Table A3.1-4 was supplied from Tables 2.6-4 and 2.6-5 of the Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report.

For MNGP, the chemistry values are the same as those used in the NRC analysis, however, the chemistry factor is higher due to an adjustment to reflect the results from two surveillance capsules. The value of fluence is lower than that used in the NRC analysis. As a result, the shift in reference temperature is lower than the 64 EFPY shift from the NRC analysis. In addition, the unirradiated reference temperature is essentially the same. The combination of unirradiated reference temperature ($RT_{NDT}(U)$) and shift (ΔRT_{NDT} w/o margin) yields an ART that is lower than the NRC mean analysis value.

Therefore, the RPV shell weld embrittlement due to fluence has a negligible effect on the probabilities of RPV shell weld failure. The Mean RTNDT value at 54 EFPY is bounded by the 64 EFPY Mean RTNDT provided by the NRC. Although a conditional failure probability has not been calculated, the fact that the MNGP values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the MNGP RPV conditional failure probability is bounded by the NRC analysis.

The procedures and training used to limit reactor pressure vessel cold over-pressure events will be the same as those approved by the NRC when MNGP requested approval of the BWRVIP-05 technical alternative for the term of the current operating license. A request for extension for the 60-year extended operating period will be submitted to the NRC prior to the period of extended operation.

Table A3.1-4 Effects of Irradiation on RPV Circumferential Weld Properties for MNGP

Group	CB&I 64 EFPY (Reference 10)	MNGP 54 EFPY
Cu (%)	0.10	0.10
Ni (%)	0.99	0.99
CF	134.9	138.5
Fluence at clad/weld interface (10^{19} n/cm^2)	1.02	0.52
ΔRT_{NDT} w/o margin (°F)	135.6	113
$RT_{NDT(U)}$ (°F)	-65	-65.6
Mean RT_{NDT} (°F)	70.6	47.4
P (F/E) NRC[1]	1.78×10^{-5}	[2]
P (F/E) BWRVIP	-	-

1 P (F/E) stands for "probability of a failure event."

2 Although a conditional failure probability has not been calculated, the fact that the MNGP values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the MNGP RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements of Reference 10.

RPV Axial Weld Failure Probability

Summary Description

The BWRVIP recommendations for inspection of RPV shell welds (Reference 11) contain generic analyses supporting an NRC SER (Reference 10) conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate of 5×10^{-6} per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above in the section entitled "RPV Circumferential Weld Examination Relief."

MNGP received relief from the circumferential weld inspections for the remaining 40 year licensed operating period. The axial weld failure probability analysis meets the requirements of 10 CFR 54.3(a) (Reference 2). As such, it is a TLAA.

Analysis

As stated above in the section entitled "RPV Circumferential Weld Examination Relief," MNGP received NRC approval for a technical alternative that eliminated the RPV circumferential shell weld inspections for the current license term. The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The NRC SER associated with BWRVIP-05 (Reference 10) concluded 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 5×10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the RPV axial welds will be inspected. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 54 EFPY and approval by the NRC to extend the RPV circumferential weld inspection relief request.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Table A3.1-5 compares the limiting axial weld 54 EFPY properties for MNGP against the values taken from Table 2.6-5 found in the NRC SER for BWRVIP-05 and associated supplement to the SER (Reference 12). The SER supplement required the limiting axial weld to be compared with data found in Table 3 of the document. For MNGP, the comparison was made to the 'Mod 2' plant information. The supplemental SER stated that the 'Mod 2' calculations most closely match the 5×10^{-6} RPV failure frequency.

For MNGP, the fluence value is greater than that used in the NRC analysis. However, the weld material has a significantly lower copper value (0.10 vs. 0.219 used in the NRC analysis); the nickel values are essentially the same as those used in the NRC analysis. As a result, the value of ΔRT_{NDT} is lower than the NRC analysis. In addition, the unirradiated RT_{NDT} was significantly lower (-65.6°F vs. -2°F used in the NRC analysis). The MNGP limiting weld 54 EFPY Mean RT_{NDT} value is within the limits of the values assumed in the analysis performed by the NRC staff in the March 7, 2000, BWRVIP-05 SER supplement and the 64 EFPY limits and values obtained from Table 2.6.5 of the SER. Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

Table A3.1-5 Effects for Irradiation on RPV Axial Weld Properties for MNGP

Value	Mod 2 (Reference 4)	MNGP 54 EFPY
Cu (%)	0.219	0.10
Ni (%)	0.996	0.99
CF		138.5
Fluence x 10^{19} (n/cm^2)	0.148[a]	0.52
ΔRT_{NDT} (°F)	116	113
$RT_{NDT(U)}$ (°F)	-2	-65.6
Mean RT_{NDT} (°F)	114	47.4
P (F/E) NRC	5.02×10^{-6}	[b]

a Peak Axial Fluence

b Although a conditional failure probability has not been calculated, the fact that the MNGP values at the end of license are less than the Mod 2 value provided by the NRC leads to the conclusion that the MNGP RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements of Reference 10.

A3.2 Metal Fatigue of the RPV and Internals, and Reactor Coolant Pressure Boundary Piping and Components

A cyclically loaded metal component may fail because of fatigue even though the cyclic stresses are considerably less than the static design limit. Some design codes such as the ASME Boiler and Pressure Vessel Code and the ANSI piping codes contain explicit metal fatigue calculations or design limits. Cyclic or fatigue design of other components may not be to these codes, but may use similar methods. These analyses, calculations and designs to cycle count limits or to fatigue usage factor limits may be TLAAs.

Fatigue analyses are presented in the following groupings:

- RPV Fatigue Analyses
- RPV Internals Fatigue Analysis

NUREG-1801 identifies numerous fatigue related aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800.

RPV Fatigue Analyses

Summary Description

RPV fatigue analyses were performed for the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs. The end-of-40-year license fatigue usage was determined for the normal and upset pressure and thermal cycle events. Subsequent to the original stress analyses, several hardware changes, operational changes (such as the 1998 power rerate), and/or stress analysis revisions have affected the usage factors.

Calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations. The RPV fatigue analyses are TLAAs.

Analysis

The original RPV stress report included a fatigue analysis for the RPV components based on a set of design basis duty cycles. The original 40-year analyses demonstrated that the CUFs for the critical components would remain below the ASME Code Section III allowable value of 1.0.

A reanalysis was performed for RPV CUF values as a part of the 1998 power rerate implementation at MNGP. For power rerate implementation, only components in which the original and power rerate modification stress report CUF values are greater than 0.5 required reanalysis. Subsequent to the original and modification analyses, a fatigue monitoring program was developed and revised fatigue usage values were determined. These fatigue usage values consider actual thermal cycle experience through September 30, 2004. The resulting fatigue CUF values determined for the monitoring program and power rerate supersede the values determined in the original and modification RPV analyses. The current (as of September 30, 2004) and 60-year fatigue usage values are listed in Table A3.2-1.

Table A3.2-1 Fatigue Evaluation Results for Limiting Components

Component	Computed Fatigue Usage Factor (through 9/30/2004	Computed Fatigue Usage Factor 60-Year License	Monitoring Recommended by NUREG/ CR-6260
Recirculation Outlet Nozzle	0.010	0.015	Yes
Recirculation Inlet Nozzle	0.145	0.220	Yes
Steam Outlet Nozzle	0.124	0.187	No
Feedwater Nozzle	0.328	0.597	Yes
Core Spray Nozzle	0.233	0.645	Yes
Core Support Structure	0.039	0.058	No
Bottom Head and Support Skirt	0.206	0.293	Yes
Control Rod Drive Penetrations	0.179	0.288	No
Vessel Closure Bolts	0.340	0.554	No
Refueling Bellows Skirt	0.502	0.829	No

These results incorporate current fatigue monitoring program cycles accumulated through September 30, 2004. Cycle counting includes those cycles identified in MNGP USAR Table 4.2-1, which identifies the following transient cycles:

MNGP Transient Cycles

Transient Type	No. of Design Cycles (USAR Table 4.2-1)	Projected to 2030
Bolt Up/Unbolt	120	44
Startup/Shutdown @ 100°F/hr	289	207
Scrams	270	165
Design Hydrostatic Test @ 1250 psig	130	67
Reactor Overpressure @ 1375 psig	1	0
Hydrostatic Test to 1560 psig	3	2
Rapid Blowdown	1	0
Liquid Poison Flow @ 80°F	10	0
Feedwater Heater Bypass	70	0
Loss of Feedwater Heater	10	0
Loss of Feedwater Pumps	30	0
Improper Start of Shutdown Recirc Loop	10	8

It should be noted that not all cycles apply to all locations evaluated, and that the number of design cycles identified above represent design values, not the maximum allowable number of transients.

The original code analysis of the reactor vessel included fatigue analysis of the control rod drive hydraulic system return line nozzles. After several years of operation, it was discovered that the control rod drive hydraulic system return line nozzles were subject to cracking caused by a number of factors including rapid thermal cycling (Reference 13). Consequently, the control rod drive hydraulic system return line nozzles were capped and removed from service. As such, they are no longer subject to rapid thermal cycling.

Disposition: Revision and Aging Management, 10 CFR 54.21(c)(1) (ii) and (iii)

For the period of extended operation, the fatigue usage factors for the limiting components have been re-evaluated. No MNGP component exceeded the ASME Code allowable for the 60-year license. The results of the evaluation are shown in Table A3.2-1.

As stated in Chapter IV.A1 of the Generic Aging Lessons Learned (GALL) (NUREG-1801), environmental fatigue issues must be considered for Class 1 components. Chapter 4.3 (Metal Fatigue) of NUREG-1800 states that an aging management program consistent with Chapter X.M1 of the GALL is an acceptable method for management of metal fatigue for the period of extended operation. The current fatigue monitoring program tracks CUFs through cycle-based fatigue (CBF) monitoring.

CBF monitoring consists of a two-step process: (a) cycle counting, and (b) CUF computation based on the counted cycles. The cycle counting counts each transient that is defined in the plant-licensing basis based upon the mechanistic process or sequence of events experienced by the plant as determined from monitored plant instruments. The approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. Transients are identified and implemented into the aging management program. CUF computation calculates fatigue directly from counted transients and parameters for the monitored components. CUF is computed via a design-basis fatigue calculation where the numbers of cycles are substituted for assumed design basis number of cycles.

The current fatigue monitoring program includes 10 components listed in Table A3.2-1. With environmental fatigue considered (see Section A3.7), this program meets the recommendations of Chapter X.M1 of the GALL for the period of extended operation. This is consistent with the components listed in NUREG/CR-6260 (Reference 14), and the recommendations of Chapter X.M1 of the GALL.

Fatigue Analysis of RPV Internals

Summary Description

Fatigue analysis of the RPV internals was performed using the ASME Boiler and Pressure Vessel Code, Section III, as a guide. The most significant fatigue loading occurs at the jet pump diffuser to baffle plate weld location. The original 40-year calculation showed a CUF of ~0.33, less than the ASME allowable of 1.0 (Reference 6). Because this analysis used a number of cycles for a 40-year life, it is a TLAA.

Analysis

The events analyzed included: (1) Normal startup and shutdown; (2) Improper start of a recirculation loop; and (3) DBA. The fatigue evaluation determined that peak strains occurred as a result of the improper recirculation loop startup transient and the point in the time of the DBA flooding (Low Pressure Coolant Injection (LPCI)) when the shroud and shroud support plate through-wall gradients are at a maximum. None of the other events analyzed contributed significantly to fatigue usage. The 40-year CUF for this location was determined to be ~0.33, i.e., less than the ASME allowable of 1.0.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Because the original fatigue analysis used a number of cycles for a 40-year design life, the calculation was revised for a 60-year life by scaling up the number of cycles by 1.5, except for the DBA transient. The resultant fatigue usage was calculated to be ~0.5, which is less than the ASME Code allowable of 1.0. Therefore, the fatigue usage of the RPV internal components is acceptable for the period of extended operation.

A3.3 ASME Section III Class 1 Reactor Coolant Pressure Boundary (RCPB) Piping Fatigue Analysis

Summary Description

MNGP piping systems were originally designed in accordance with ASA B31.1 and USAS B31.1.0, which did not require that an explicit fatigue analysis be performed.

Reconciliation for the use of later editions of construction codes for modification to or replacement of piping and components has been performed in accordance with Section IWA-7210(c), Section XI of the ASME Code. The governing code for design, materials, fabrication and erection of piping, piping components, and pipe support modifications or replacements is ANSI B31.1, 1977 Edition including Addenda up to and including the Winter of 1978.

Portions of Class 1 systems such as the Reactor Recirculation, Core Spray and RHR inside drywell were required to be analyzed for fatigue in accordance with the ASME Code Section III for Nuclear Class I piping. The implementation of these requirements at MNGP were for the purpose of attaining a higher quality level and provide more detailed analysis to confirm protection of the reactor coolant system integrity.

The analyses demonstrate that the 40 year cumulative usage factors (CUF) for the limiting components in all effected systems are below the ASME Code Section III allowable value of 1.0. Because these analyses are based on cycles postulated to occur in the current 40 year design life, they are TLAAs.

Analysis

With the exception of the torus attached piping and safety relief discharge line piping which were evaluated as part of the Mark I program "New Loads" program, the only piping that has been explicitly analyzed for fatigue are portions of the recirculation system piping, RHR piping, and core spray piping systems. These systems were all modified under the Generic Letter (GL) 88-01 IGSCC inspection and mitigation program.

This piping was originally designed in accordance with USAS B31.1 and modifications were analyzed to ASME Section III Class 1 rules. The ASME Code limit for fatigue is 1.0. The limiting fatigue usages for these systems are shown in Table A3.3-1.

Table A3.3-1 MNGP Fatigue Monitoring Locations for RCPB Class 1 Piping

Location	40 Year Cumulative Fatigue Usage Factor
Recirculation Equalizer Line Branch Connection	0.8514
RHR Return Loop B Tapered Transition	0.8875
Core Spray Valve Joint	0.6466

For fatigue analyses, the change in stress produced by transients are compared to allowable limits. For a given stress range, the ASME code allows a maximum number of cycles. In a fatigue analysis the actual or design assumed number of cycles is compared to the allowed maximum, and this ratio is summed for all significant transients experienced by the component. The summation, or usage factor, must be less than or equal to 1.0 to be acceptable.

The fatigue analyses for these systems were evaluated using a bounding set of assumed thermal cycles that may occur over the life of the plant (40 years). These conservative evaluations resulted in fatigue usage values that are acceptable (i.e. less than 1.0) however, with the exception of the core spray piping there is not sufficient margin to extrapolate by a ratio of 1.5 with acceptable results.

Cycle based counting consists of periodically counting the relevant cycles and calculating the cumulative usage factor (CUF). This process is also conservative due to the fact that all transients within a group are assumed to be equal in severity and correspond to the maximum cycle thermal limits specified in the design. Based on the number of cycles experienced at MNGP through September 2004, the maximum fatigue usages identified in Table A3.3-1 for the Recirculation and RHR piping systems are expected not to exceed 0.90 at the end of sixty (60) years of plant operation.

<u>Disposition: Validation, 10 CFR 54.21(c)(1)(i) and Aging Management 10 CFR 54.21(c)(1)(iii)</u>

The limiting location for RCPB core spray piping is less than 0.65. Consequently, current analyses are validated for the period of extended operation by:

$$U_{max,40} < 0.65, \times 60/40 = U_{max,60} = 0.975 < 1.0$$

The limiting locations for the recirculation and RHR piping are less than 0.90 taking into account actual cycles accumulated through 2002 and projecting those cycles to 60 years. The MNGP cycle based fatigue monitoring system manages this aging mechanism to ensure that fatigue does not exceed the allowable limit of 1.0.

A3.4 RCPB Section III Class 2 AND 3, ASA B31.1 AND, USAS B31.1 Piping and Components

Summary Description

MNGP piping systems were originally designed in accordance with ASA B31.1 and USAS B31.1.0, which did not require that an explicit fatigue analysis be performed.

Reconciliation for the use of later editions of construction codes for modification to or replacement of piping and components has been performed in accordance with Section IWA-7210(c), Section XI of the ASME Code. The governing code for design, materials, fabrication and erection of piping, piping components, and pipe support modifications or replacements is ANSI B31.1, 1977 Edition including Addenda up to and including the Winter of 1978.

The codes and standards which MNGP was designed and constructed to did not include fatigue analyses for piping, component supports or component connections and anchors. The only exceptions are some ASME Class MC containment piping support and penetration analyses for "New Loads" (Section A3.8), and RCPB piping discussed in the preceding section.

Analysis

Although the code of construction did not invoke fatigue analyses, a stress range reduction factor which is applied to the allowable stress range for expansion stresses is required to account for cyclic thermal conditions. The allowable secondary stress range is 1.0 S_A for 7,000 equivalent full temperature thermal cycles or less and is incrementally reduced to 0.5 SA for greater than 100,000 cycles. With the exception of piping described in Section A3.3 and Section A3.8, MNGP piping analyses incorporated stress range reduction factors for a finite number of thermal cycles in lieu of fatigue analyses.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

An estimate of the number of thermal cycles experienced by these piping systems can be conservatively approximated by the maximum number of thermal cycles used in reactor nozzle fatigue analyses. For MNGP the bounding number of cycles used for the qualification of a vessel nozzle is 1,500 for the feedwater nozzle. The maximum number of cycles projected through the extended period of operation is therefore, 1.5 times 1,500 (2,250). This conservative amount of full range cycles is significantly less than the 7000 cycle limit, consequently existing analyses are valid through the extended term of operation.

A3.5 Irradiation Assisted Stress Corrosion Cracking

Summary Description

Austenitic stainless steel RPV internal components exposed to a neutron fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. As described in the SER to BWRVIP-26, IASCC of RPV internals is a TLAA.

Analysis

Fluence calculations have been performed for the RPV and internals, including the effects of a potential power uprate (1880 MWt). Three components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide, (2) Shroud, and (3) Incore Instrumentation Dry Tubes and Guide Tubes.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The top guide, shroud, and incore instrumentation dry tubes and guide tubes are susceptible to IASCC. The aging effect associated with IASCC (crack initiation and growth) will require aging management. All three components (top guide, shroud, and incore instrumentation dry tubes and guide tubes) have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (incore instrumentation dry tubes and guide tubes). BWRVIP recommendations are implemented at MNGP by the Water Chemistry and the In-Service Inspection Programs.

A3.6 Stress Relaxation of Rim Holddown Bolts

Summary Description

As described in the SER to BWRVIP-25, plants must consider relaxation of the rim-hold-down bolts as a TLAA issue. Because MNGP has not installed core plate wedges, the loss of preload must be considered in the TLAA evaluation.

Analysis

The core plate hold-down bolts connect the core plate to the core shroud. These bolts are subject to stress relaxation due to thermal and irradiation effects. For the 40-year lifetime, the BWRVIP concluded that all rim hold-down bolts would maintain some preload throughout the life of the plant.

Disposition: Revision 10 CFR 54.21(c)(1)(ii)

For the period of extended operation, the expected loss of preload was assumed to be 19%, which bounds the original BWRVIP analysis. With a loss of 19% in preload, the core plate will maintain sufficient preload to prevent sliding under both normal and accident conditions. Therefore, the loss of preload is acceptable for the period of extended operation.

A3.7 Effects of Reactor Coolant Environment

Summary Description

Generic Safety Issue (GSI)-190 was identified by the NRC because of concerns about the effects of reactor water environments on the fatigue life of components and piping during the period of extended operation. GSI-190 was closed in December of 1999, and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Fatigue calculations that include consideration of environmental effects to establish cumulative usage factors can be treated as TLAAs under 10 CFR Part 54 or they could be used to establish the need for an aging management program.

To qualify as a TLAA, the analysis must satisfy all (6) criteria defined in 10 CFR 54.3. Failure to satisfy any one of these criteria eliminates the analysis from further consideration as a TLAA.

Fatigue design for MNGP has been determined to be a TLAA, even though the design limits are based on cycles rather than an explicit time period. Reactor water environmental effects, however, are not included in the MNGP current licensing basis (CLB). Consequently, the criterion of 10 CFR 54.3(a)(6) is not satisfied. Nevertheless, environmental effects on Class 1 component fatigue have been evaluated to determine if any additional actions are required for the extended period of operation.

Analysis

The NRC staff assessed the impact of reactor water environment on fatigue life at high fatigue locations and presented the results in NUREG/CR-6260, "Application of NUREG/CR-5999, Interim Fatigue Curves for Selected Nuclear Power Plant Components," in March of 1995. Methodology for the determination of environmental correction factors to be applied to the fatigue analyses for carbon and low-alloy steels is contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels." Methodology for environmental fatigue factors for austenitic stainless steels is contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels."

As a part of the NRC's Fatigue Action Plan, incorporation of environmental fatigue effects originally involved a reduced set of fatigue design curves, such as those proposed by Argonne National Laboratory (ANL) in NUREG/CR-5999. As a part of the effort to close GSI-166 (later GSI-190) for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratory (INEL) evaluated fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system (NSSS) vendors. The ANL fatigue curves were used by INEL to recalculate the cumulative usage factors (CUFs) for fatigue-sensitive component locations in early and late vintage Combustion Engineering (CE) pressurized water reactors (PWRs), early and late vintage Westinghouse PWRs, early and late vintage General Electric (GE) boiling water reactors (BWRs), and Babcock & Wilcox Company (B&W) PWRs. The results of the INEL calculations were published in NUREG/CR-6260 (Reference 14). The INEL calculations took advantage of conservatisms present in governing ASME Code fatigue calculations, including the numbers of actual plant transients relative to the numbers of design-basis transients, but did not recalculate stress ranges based on actual plant transient profiles. The BWR calculations, especially the early-vintage GE BWR calculations, are directly relevant to MNGP.

In order to comply with the requirements, MNGP has evaluated the locations specified in NUREG/CR-6260 for the older vintage BWR plants. These locations consist of:

Reactor Vessel (Lower Head to Shell Transition)

Feedwater Nozzle

Recirculation System (Vessel Nozzles and RHR Return Line Tee)

Core Spray System (Nozzle/Safe End)

Residual Heat Removal Piping (Tapered Transition)

Limiting Feedwater Piping Location

For each location, detailed environmental fatigue calculations have been performed using F_{en} relationships for carbon and low-alloy steel locations (NUREG/CR-6583) and stainless steel locations (NUREG/CR-5704). The calculations incorporate F_{en} methodology to determine a multiplier on the cumulative usage factor (CUF) so that environmental effects can be assessed. As can be seen in Table A3.7-1, all locations are acceptable through the extended term of operation due to the fact that all CUFs remain below the acceptance criteria of 1.0.

Table A3.7-1 Summary of Environmental Fatigue Usage Factors for MNGP

Location	Component	Material	Usage Factor (U_{env})
Reactor Vessel	Shell	Carbon Steel	0.569
Feedwater Nozzle	Safe End	Carbon Steel	0.938
Recirculation Inlet Nozzle	Safe End	Stainless Steel	0.749
Core Spray Nozzle	Safe End	Carbon Steel	0.194
Recirculation Piping	RHR Tee	Stainless Steel	0.864
Feedwater Piping	FWTR/RCIC Tee	Carbon Steel	0.513

Disposition: Revision 10 CFR 54.21(c)(1)(ii)

The cumulative usage factors for all locations, when conservatively re-evaluated to include environmental effects, remains below 1.0. Although, based on a projection of experienced cycles, these locations have been shown to be acceptable through the period of extended operation, the MNGP thermal fatigue monitoring program periodically reviews and updates fatigue analyses to ensure continued compliance with fatigue acceptance criteria.

A3.8 Fatigue Analyses of the Primary Containment, Attached Piping, and Components

The MNGP primary containment was designed in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including Winter of 1965. Subsequently, during large scale testing for the Mark III containment system and the in-plant testing for Mark I primary containment systems, new suppression chamber hydrodynamic loads were identified. These new loads are related to the loss-of-coolant-accident (LOCA) scenario and safety relief valve (SRV) operation.

Containment fatigue analyses are provided for the following groups:

Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

Fatigue Analysis of the SRV Discharge Piping Inside the Suppression Chamber and Internal Structures

Fatigue Analysis of Suppression Chamber External Piping and Penetrations

Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analysis

Primary Containment Process Penetration Bellows Fatigue Analyses

Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

Summary Description

New hydrodynamic loads were identified subsequent to the original design for the containment suppression chamber vents. These loads result from blowdown into the suppression chamber during a postulated LOCA and during SRV operation for plant transients. The results of analyses of these effects are presented in the MNGP USAR. Consequently, these analyses are TLAAs.

Analysis

Analysis of the suppression chamber, vent system and downcomers (Reference 17) identified that the vent header-downcomer intersection and the torus shell were limiting in terms of fatigue usage. Fatigue usages for all other locations were found to be less than 0.015. The calculated values for the vent header-downcomer intersection and the torus shell were 0.684 and 0.66 respectively. Subsequent to that evaluation, all locations were re-evaluated for the effects of power rerate implemented in 1998. It was estimated that power rerate conditions could result in an increase in the number of SRV cycles experienced due to higher steaming rates at increased power levels. The number of cycles was estimated to increase by 14 percent coincident with the increase to 1775 MWt (from 1670 MWt) and by 26 percent due to an increase to 1880 MWt.

The revised fatigue evaluation conservatively estimated the fatigue usage of the vent header-downcomer intersection as 0.862 (1.26 x 0.684). The revised maximum fatigue for the torus shell was similarly calculated to be 0.98, using increased SRV actuations postulated for rerate conditions and applicable event combinations.

Disposition: Validation, 10 CFR 54.21(c)(1)(i) and Aging Management, 10 CFR 54.21(c)(1)(iii)

All locations with the exception of the vent header-downcomer intersection and the torus shell have reported 40 year fatigue usage factors of less than 0.2. Consequently, those locations are validated by review of the current analyses (e.g. Umax,40 < 0.20, x 60/40 = Umax,60 = 0.30<<1.0).

Since only the SRV load cases contribute to fatigue during normal operation, operation may continue until the contribution from SRV discharges has not exceeded the conservative design values used in the evaluation.

The MNGP cycle based fatigue monitoring program includes periodic counting of the SRV cycles, comparing the total number of experienced SRV cycles to the design basis number of cycles and, confirming that the fatigue usage will remain below the acceptance criteria of 1.0 or identifying when the limit is likely to be exceeded such that adequate corrective measures can be implemented. As of December 31, 2003 the total number of normal operation SRV lifts experienced at the MNGP was 506 and the design basis is 934. Extrapolation of current SRV lifts results in an conservative estimate due to the fact that counted lifts do not differentiate the operating condition at which the lift was experienced (e.g., power level), the design value of 934 postulates that all SRVs lifts occur in the same suppression chamber bay and, the rate of SRV challenges experienced in the first 7 years of operation is significantly higher than subsequently experienced. Without consideration for these conservatisms, 414 additional challenges can be expected throughout the 60 year extended operating period. This would result in a 60 year SRV total of 920.

All applicable plant cycles are currently monitored to ensure that the cumulative usage factors remains below 1.0 for the limiting components. In the unlikely event that fatigue usage is predicted to exceed 1.0 prior to 60 years of operation, appropriate corrective action will be taken in accordance with the MNGP Corrective Action Program.

Fatigue Analysis of the SRV Piping Inside the Suppression Chamber and Internal Structures

Summary Description

The Reactor Pressure Relief System includes safety/relief valves (SRVs) located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. The SRVs, which discharge to the suppression pool, provide two main protective functions:

Overpressure relief - The valves open to limit the pressure rise in the reactor.

Depressurization - The valves are opened to depressurize the reactor.

The Plant Unique Analysis Report (Reference 18) describes the fatigue analysis of the SRV discharge lines. These analyses assume a limited number of SRV actuations throughout the 40 year life of MNGP and are therefore TLAAs.

Torus internal structures (i.e., catwalk and monorail) are Service Level E structures. Consequently, no fatigue evaluation is required to demonstrate acceptability of these structures.

Analysis

The criteria presented in Volume 5 of the MNGP PUAR (Reference 18) describes the evaluation of the SRVDL piping system. The evaluation included the effects of LOCA related loads and SRV discharge related loads. LOCA and SRV discharge loads were formulated using procedures and test results which included the effects of plant unique geometry and operating parameters contained in the Plant Unique Load Definition (PULD) report (Reference 19). The analysis also considered the interaction effects of the vent system and the suppression chamber.

Per (Reference 18), the critical location for fatigue usage is the SRV piping at the elbow adjacent to the elbow support beam junction. The fatigue usage for this location was calculated to be 0.309.

Subsequent to that evaluation, this location was reevaluated for the effects of power rerate implemented in 1998. It was estimated that power rerate conditions could result in an increase in the number of SRV cycles experienced due to higher steaming rates at increased power levels. The number of cycles was estimated to increase by 14 percent coincident with the increase to 1775 MWt (from 1670 MWt) and by 26 percent due to an increase to 1880 MWt. Conservatively using the 1880 MWt SRV factor, an increase to 0.389 was calculated.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The limiting location for the SRV piping is less than 0.40. Current analyses are validated by:

$U_{max,40} < 0.40, \times 60/40 = U_{max,60} < 0.60 < 1.0$

This increase in service life does not significantly effect SRV discharge piping fatigue usage. Consequently, the current calculation is validated for the period of extended operation.

Fatigue Analysis of Suppression Chamber External Piping and Penetrations

Summary Description

These analyses include the large and small bore torus attached piping (TAP), suppression chamber penetrations and the ECCS suction header. Fatigue analyses were completed that were based on cycles postulated to occur within the 40 year operating life of the plant. Therefore these calculations are TLAAs.

Analysis

Rigorous analytical techniques were used to evaluate the effects of LOCA related and SRV discharge loads as defined in the NRC's Safety Evaluation Report NUREG-0661 and in the Mark I Containment Load Definition Report (Reference 20). These techniques included detailed analytical models and refined methods for computing the dynamic response of the TAP systems which included consideration of the interaction effects of each piping system and the suppression chamber.

The results of the TAP structural analysis for each load type were used to evaluate load combinations for the piping and penetrations in accordance with NUREG-0661 and the Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide (PUAGG). The analysis results were compared with the acceptance limits specified in the PUAAG and the applicable sections of the ASME Code for Class 2 piping and for Class MC components.

Fatigue effects were specifically addressed for the suppression chamber penetrations and the suction header, whereas the evaluation for the piping was generically addressed for all Mark I plants by the Mark I Owners' Group. Analyses documented in this report identify cumulative usage factors for the Mark I plants of less than 0.5. The generic fatigue evaluation included 36 piping systems from 15 plants. Stress results for the most limiting piping systems and locations were selected for each plant. Thus, the reported usage factors are representative of the most limiting location within the data for the plant group. For MNGP, the SRV discharge piping was identified as the limiting location. The SRV discharge piping was re-evaluated for the effects of power rerate, which was implemented in 1998. It was estimated that power rerate conditions could result in an increase in the number of SRV cycles experienced due to higher steaming rates at increased power levels. The number of cycles was estimated to increase by 14 percent coincident with the increase to 1775 MWt (from 1670 MWt) and by 26 percent due to an increase to 1880 MWt. Conservatively using the 1880 MWt SRV factor, an increase to 0.389 was calculated.

The TAP penetration fatigue usage was conservatively evaluated for the effects of power rerate by increasing the SRV cycles by a factor of 1.26 to correspond to a power level of 1880 MWt (the actual rerate power level was 1775 MWt, which corresponds to a 1.14 SRV factor). This conservative application, in addition to the bounding analysis, confirmed that fatigue usage for the TAP penetrations would remain below 1.0 (0.985) based on cycles anticipated to occur during the 40 year operating life of MNGP.

Disposition: Validation, 10 CFR 54.21(c)(1)(i) and Aging Management, 10 CFR 54.21(c)(1)(iii)

The limiting location for TAP is less than 0.40. Current analyses are validated by:

$U_{max,40} < 0.40$, x $60/40 = U_{max,60} < 0.60 < 1.0$

This increase in service life does not significantly effect TAP fatigue usage. Consequently, the current calculation is validated for the period of extended operation.

Conversely, although TAP penetration fatigue usage has been conservatively validated for 40 years of operation there is not sufficient margin to project additional cycles for a 60 year extended term of operation and remain below the acceptance criteria of 1.0.

Since SRV load cases are the primary contributor to fatigue during normal operation, operation may continue until the contribution from SRV discharges has not exceeded the conservative design values used in the evaluation.

The MNGP cycle based fatigue monitoring includes periodic counting of the SRV cycles. The SRV cycles are compared to the design basis number of cycles to confirm that the fatigue usage will remain below the acceptance criteria of 1.0 and to provide timely identification of when the limit may be exceeded such that adequate corrective measures can be enacted. As of December 31, 2003 the total number of SRV lifts experienced at the MNGP was 506. Projecting this rate of SRV lifts throughout 60 years of operation indicates that the fatigue usage will remain below 1.0 for the period of extended operation.

All applicable plant cycles are currently monitored to ensure that the cumulative usage factors remains below 1.0 for the limiting components. In the unlikely event that fatigue usage is predicted to exceed 1.0 prior to 60 years of operation, appropriate corrective action will be taken in accordance with the MNGP Corrective Action Program.

Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analysis

Summary Description

The drywell-to-suppression chamber vent line bellows are included in the Mark I Containment Long Term Program plant-unique analysis. A fatigue analysis of the vent line bellows demonstrates their adequacy to accommodate thermal and internal pressure load cycles for the life of the plant. As such this analysis is a TLAA.

Analysis

The suppression chamber is in the general form of a torus, which is below and encircles the drywell. The suppression chamber is connected to the drywell by eight vent lines which are connected to a common header. A vent line bellows assembly connects each vent line to the suppression chamber allowing for differential movement between the drywell and the suppression chamber.

Vent line bellows stresses are due primarily to differential thermal expansion of the reactor pressure vessel and the drywell during normal startup and shutdown evolutions and, due to accident conditions. The original vent line bellows was designed and analyzed in accordance with ASME Section III, 1965 Edition including the Summer 1966 Addenda. The current evaluation was performed in accordance with ASME Section III, Subsection NC, using the 1995 Edition including the 1996 Addenda. The current analysis for the vent line bellows conservatively used as the basis for the expected number of cycles to be experienced during the forty (40) year design life 300 startup/shutdown cycles and 1 cycle due to postulated accident conditions.

The result of this analysis was confirmation that cumulative usage factor (CUF) is significantly below the acceptance criteria of 1.0 for the 40 year design life.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

By inspection of the current analysis, which predicts a maximum 40 year design CUF of 0.10, the fatigue adequacy of the vent line bellows at MNGP is validated. The capacity of the vent line bellows is adequate for the number of transient cycles expected during the extended 60 year operating period.

$U_{max,40} = 0.10, \times 60/40 = U_{max,60} = 0.15 < 1.0$

Primary Containment Process Penetration Bellows Fatigue Analysis

Summary Description

Containment pipe penetrations that are required to accommodate thermal movement have expansion bellows. The bellows are designed for a minimum number of operating cycles over the design life of the plant. Consequently, the primary containment process penetrations bellows cycle basis is a TLAA.

Analysis

At MNGP, the only containment process piping that is subject to significant thermal expansion and contraction are those that penetrate the drywell shell. Typically these penetrations, which were designed to the ASME Code, Section III, Class B requirements, are a triple flued head design which has a guard pipe between the process piping and the penetration nozzle. This permits the penetration to be vented to the drywell should a rupture of the hot line occur within the penetration.

These containment penetration process bellows have been designed for a minimum of 7,000 operating cycles.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Transient cycles on the bellows are composed primarily of thermal cycles experienced by the attached piping. The cycle requirements can be conservatively approximated by the maximum number of thermal cycles specified for any reactor pressure vessel nozzle. For MNGP the limiting nozzle from a total cycle standpoint is the feedwater nozzle, which has as its design basis 1,500 applied cycles for a 40 year operating period. For the 60 year extended operating period, the number of cycles can be estimated by multiplying the 40 year value times 1.5 which results in an estimated design cycle expectation of less than 2,250 or less than one-third of the original design requirement. Consequently, the current containment penetration bellows fatigue design criteria remain valid with significant margin for the 60 year extended operating period.

A3.9 Environmental Qualification of Electrical Equipment (EQ)

Summary Description

10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants, specifically requires that an environmental qualification program be established to demonstrate that certain electrical components located in "harsh" plant environments are qualified to perform their safety function in those harsh environments after the effects of in-service aging.

The MNGP Environmental Qualification Program meets the requirements of 10 CFR 50.49 for the applicable components important to safety.

10 CFR 50.49(e)(5) contains provisions for aging that include consideration of all significant types of aging degradation that can affect component functional capability.

10 CFR 50.49(e) also requires replacement or refurbishment of components qualified for less than the current license term prior to the end of designated life unless additional life is established through ongoing qualification.

Supplementary EQ regulatory guidance for compliance with these different qualification criteria is provided in the Division of Operating Reactors (DOR) Guidelines (Reference 16), NUREG-0588, Regulatory Guide 1.89, and in Generic Letter 82-09.

The MNGP EQ Program manages component thermal, radiation and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Aging evaluations for EQ components that specify a qualification of at least 40 years are TLAAs for license renewal. The EQ Program will manage the aging effects of applicable components in the EQ program. Section 4.4.2.1.3 of NUREG-1800 states that the staff has evaluated the EQ Program (10 CFR 50.49) and determined that it is an acceptable aging management program to address EQ according to 10 CFR 54.21(c)(1)(iii), Aging Management. This evaluation is documented in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report, Section X.E1, "Environmental Qualification of Electric Components."

The MNGP EQ Program is an existing program, established to meet commitments for 10 CFR 50.49, that are consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section X.E1, "Environmental Qualification of Electric Components." In accordance with 10 CFR 54.21(c)(1)(iii), the EQ Program, which implements the requirements of 10 CFR 50.49, is viewed as an aging management program for license renewal. Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) is performed as part of the EQ Program at MNGP.

Analysis

Aging evaluations of electrical components in the EQ program at MNGP that specify a qualified life of at least forty years are TLAAs.

Aging evaluations are normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation or by including new aging data. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters such as the assumed ambient temperature of the component, the activation energy, or in the application of a component (e.g. de-energized vs. energized). Important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). These attributes are discussed in more detail below.

- Analytical Methods - The MNGP EQ Program generally uses the same analytical models in the reanalysis of an aging evaluation as those previously applied for the current evaluation. The Arrhenius methodology is an acceptable model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, acceptable methods for establishing the 60 year normal radiation dose includes multiplying the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years) or using the actual calculated value for 60 years. The result is added to the accident radiation dose to obtain the total integrated dose for the component. In some cases, the normal radiation dose is insignificant when compared to the accident dose. In such cases the accident dose may be valid for both the 40 year and 60 year dose. For cyclical aging a similar approach may be used. Other models may be justified on a case-by-case basis.

- Data Collection and Reduction Methods - Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the primary method used for a reanalysis per the EQ Program. Temperature data used in an aging evaluation should be conservative and based on plant design temperature or on actual plant temperature data. When used, plant temperature data can be obtained in several ways including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperature used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to the material activation energy values as part of a reanalysis are to be justified on a plant specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.
- Underlying Assumptions - EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.
- Acceptance Criteria and Corrective Action - The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid. A reanalysis is performed in a timely manner (that is, sufficient time is available to maintain, replace, or re-qualify the component if the reanalysis is unsuccessful).

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

Based on a review of the MNGP EQ Program and operating experience, the continued effective implementation of the program provides reasonable assurance that (a) the aging effects will be managed, and (b) EQ components will continue to perform their intended function(s) consistent with the current licensing basis for the period of extended operation. Therefore, the MNGP EQ Program is an acceptable aging management program for license renewal under 10 CFR 54.21(c)(1)(iii) during the period of extended operation.

A3.10 Reactor Building Crane Load Cycles

Summary Description

The MNGP Reactor Building Crane System consists of an 85 ton bridge crane. The crane is capable of handling the drywell head, reactor vessel head, pool plugs and spent fuel pool shipping cask. A refueling service platform, with necessary handling and grappling fixtures, services the refueling area and the spent fuel pool.

The Reactor Building Crane System has been modified to incorporate redundant safety features which were not a part of the original design. The modification consists of a new trolley with redundant design features and a capacity of 85 tons on the main hook with redundancy features and an auxiliary 5 ton capacity hook. This modification was implemented for handling heavy loads both during refueling operations and during operations involving the off site shipment of spent fuel. Such off site shipments of fuel can take place either when the plant is operating or shut down. The redundant crane has been installed to reduce the probability of a heavy load drop to the category of an incredible event.

NUREG-0612 suggests that cranes should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, Overhead and Gantry Cranes, and of CMAA-70, Specifications for Electric Overhead Traveling Cranes. The Reactor Building Crane, manufactured prior to the issuance of CMAA-70 and ANSI B30.2, was designed to meet EOCI 61.

Since the evaluation used as a basis, an expected number of load cycles over the 40 year life of the plant Reactor Building Crane load cycles are a TLAA.

Analysis

Reactor Building Crane System design conservatively considers that the following heavy load cycles will be required during the 40 year plant life: 20 lifts per year of Reactor Building shield blocks and plugs, 2 lifts per year of the reactor vessel head, 2 lifts per year of the drywell vessel head, 2 lifts per year of the steam separator assembly and, 2 lifts per year of the steam dryer assembly.

Without consideration for the fact that the modified Reactor Building Crane System was installed after several years of operation the total amount of heavy lifts expected during a 40 year life is 1,120 cycles.

Disposition: Validation, 10 CFR54.21(c)(1)(i)

The Reactor Building Crane was conservatively designed to handle up to 70,000 heavy loads over the 40 year operating life of the plant. By inspection, the crane is expected to be subjected to less than 2,000 heavy lifts during the 60 year extended operating period, which is significantly less than the design value. Therefore, fatigue life is not significant for the operation of the Reactor Building Crane System and the current analysis remains valid for the period of extended operation.

A3.11 Fatigue Analyses of HPCI & RCIC Turbine Exhaust Penetrations

Summary Description

To evaluate the effects of testing the operability and performance of the turbine-pump units on a periodic basis MNGP conducted a detailed evaluation of the thermal cycles experienced during testing. Since the number of cycles used in the evaluation is based on a 40 year plant life, this is a TLAA.

Analysis

The existing evaluation of the High Pressure Coolant Injection turbine exhaust nozzle used test conditions of 292°F and 50 psig in conjunction with Mark I loads to calculate a cumulative fatigue usage factor. The main conclusion of this evaluation was that the maximum number of High Pressure Coolant Injection turbine tests allowed was only 260, or approximately one test every other month assuming a 40 year plant life.

The major factor was the design temperature of 292°F, the saturated steam temperature associated with the torus at a design pressure of 50 psig. Since the normal operating pressure of the torus is close to atmospheric, it was believed that the actual test temperature was closer to 212°F. To confirm this, the High Pressure Coolant Injection and Reactor Core Isolation Cooling torus nozzles were instrumented to obtain the actual temperature responses during operational testing. The conclusion of these tests was that the maximum temperature that either of these nozzles will experience is expected not to exceed 225°F. A thermal stress analysis was subsequently completed for both nozzles. Finite element models were developed for both nozzles which included explicit modeling of the nozzle to insert plate welds and nozzle to sleeve welds. The evaluation was performed for the following thermal load cases:

A through wall temperature of the nozzle wall at 225°F with the torus insert plate at 70°F. This corresponds to the initial heatup of the nozzle that occurs immediately after turbine start.

A through wall temperature of 118°F to simulate a rapid cooldown which occurs during reflood. This corresponds to the average temperature of the Reactor Core Isolation Cooling nozzle immediately after turbine shutdown.

These two cases were separately evaluated for each penetration. Based on the results, usage factors were calculated in accordance with Section III, Subsection NE of the ASME code. The maximum peak stress ranges for the heatup and cooldown cycles are 77.4 ksi and 83.5 ksi for the High Pressure Coolant Injection and Reactor Core Isolation Cooling penetrations, respectively. Based on an assumption that 676 single safety relief valve (SRV) actuations and 258 multiple valve actuations will occur during the 40 year plant life, the SRV usage factors for the High Pressure Coolant Injection and Reactor Core Isolation Cooling nozzles are 0.009 and 0.043, respectively. The worst case fatigue loading for both nozzles that could be caused by Mark I LOCA loads is a DBA CO acting simultaneously with OBE. One turbine actuation cycle was also postulated for this case. From the Mark I program stress results, the maximum LOCA usage factors for the High Pressure Coolant Injection and Reactor Core Isolation Cooling nozzles are 0.044 and 0.228, respectively. By summing the usage factors for the SRV actuations and Mark I LOCA loads plus OBE, cumulative usage factors of 0.053 and 0.271 were obtained for the High Pressure Coolant Injection and Reactor Core Isolation Cooling nozzles, respectively.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Considering that the effects of power rerate implemented in 1998 may increase the design cycles for SRV actuations by as much as 26 percent due to higher steaming rates (conservative percent increase corresponding to 1880 MWt), the maximum contribution due to SRV cycles is 1.26 times 0.043, or 0.054. Consequently, the maximum cumulative fatigue usage for 40 years is 0.282. This updated, current analysis, therefore, is validated for 60 years of operation by:

$U_{max,40} = 0.282$, x $60/40 = U_{max,60} = 0.423 < 1.0$

This results in a minimum of 0.577 available fatigue usage due to operational testing of the Reactor Core Isolation Cooling turbine which corresponds to 3,826 operational tests (an average of more than 5 tests per month over the 60 year extended life).

A4 TLAA SUPPORTING ACTIVITIES

A4.1 Environmental Qualification (EQ) of Electrical Components

The purpose of the MNGP EQ Program is to ensure that safety-related electrical equipment is capable of performing its function in a harsh environment (effects of a loss of coolant accident (LOCA), high energy line break (HELB), or post LOCA radiation) and is qualified in accordance with the Equipment Qualification Final Rule, 10 CFR 50.49, dated February 22, 1983. The MNGP program will continue through the end of the 20-year period of extended operation.

A4.2 Metal Fatigue of Reactor Coolant Pressure Boundary

The MNGP Metal Fatigue of the Reactor Coolant Pressure Boundary aging management program is part of the MNGP Thermal Fatigue Monitoring Program. The MNGP Thermal Fatigue Monitoring Program provides for the periodic review of plant transients for impact on selected components. In addition, MNGP has evaluated environmental effects in accordance with NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components." Selected components were evaluated using material specific guidance presented in NUREG/CR-6583 for carbon and low alloy steels and in NUREG/CR-5704 for austenitic stainless steels. The MNGP program ensures that limiting components remain within the acceptance criteria for cumulative fatigue usage throughout the licensed term and, that if trends indicate otherwise, appropriate corrective action can be implemented.

A4.3 Exemptions

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

A.5 COMMITMENTS

ITEM	COMMITMENT	SOURCE	SCHEDULE
1.	Each year, following the submittal of the MNGP License Renewal Application and at least three months before the scheduled completion of the NRC review, NMC will submit amendments to the MNGP application pursuant to 10 CFR 54.21 (b). These revisions will identify any changes to the current licensing basis that materially affect the contents of the License Renewal Application, including the USAR supplements and any other aspects of the application.	LRA Section 1.4	Annually
2.	In accordance with the guidance of Appendix A.3.2.1.2 of NUREG-1800, Appendix B of the latest issued supplement to NUREG-0933 will be reviewed for new GSIs designated as USI-, HIGH-, or MEDIUM- priority. Any identified that involve TLAAs or aging effects for structures and components subject to an aging management review will be included in the annual update of the LRA.	LRA Section 3.0.7	Annually
3.	Inspection of the steam dryer is to be accomplished using the guidelines in the approved BWRVIP topical report(s) for steam dryer inspection. In the event a new steam dryer is installed, NMC will reevaluate the inspection requirements.	LRA Table 3.1.2-3, Note 136	As Required
4.	The interior of the Diesel Fire Pump House masonry block walls is covered with insulation. The Structures Monitoring Program will require that the interior surfaces of the walls will be examined if exterior wall surfaces show evidence of significant aging effects.	LRA Table 3.5.2-8, Note 516	As Required
5.	The procedures and training used to limit reactor pressure vessel cold over-pressure events will be the same as those approved by the NRC when MNGP requested approval of the BWRVIP-05 technical alternative for the term of the current operating license. A request for extension for the 60-year extended operating period will be submitted to the NRC prior to the period of extended operation.	LRA Section 4.2.6	As Required
6.	MNGP site-specific administrative work instructions will be applicable to both safety and non-safety related systems, structures and components that are subject to an aging management review consistent with the current licensing basis during the period of extended operation.	LRA Section B1.3	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
7.	Site documents that implement aging management activities for license renewal will be enhanced to ensure that an AR is prepared in accordance with plant procedures whenever non-conforming conditions are found (i.e., the acceptance criteria is not met).	LRA Section B1.3	Prior to Period of Extended Operation
8.	Revisions will be made to procedures and instructions that implement or administer aging management programs and/or activities for the purpose of managing the associated aging effects for the duration of extended operation.	LRA Section B1.3	Prior to Period of Extended Operation
9.	The MNGP ASME Section XI, Subsection IWF Program will be enhanced to provide inspections of Class MC components supports consistent with NUREG-1801, Chapter III, Section B1.3.	LRA Section B2.1.3	Prior to Period of Extended Operation
10.	The guidance for performing visual bolting inspections contained in EPRI TR-104213, Bolted Joint Maintenance & Application Guide, and the Good Bolting Practices Handbook (EPRI NP-5067 Volumes I and 2) will be included in the Bus Duct Inspection Program, Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program, Structures Monitoring Program and the System Condition Monitoring Program.	LRA Section B2.1.4	Prior to Period of Extended Operation
11.	The Buried Piping and Tanks Inspection Program will update the implementing procedures to include inspections of buried components when they are uncovered.	LRA Section B2.1.5	Prior to Period of Extended Operation
12.	The Diesel Fuel Oil Storage Tank, T-44, internal inspection will be added to the list of scheduled inspections in the Buried Piping and Tanks Inspection Program.	LRA Section B2.1.5	Prior to Period of Extended Operation
13.	The Buried Piping and Tanks Inspection Program will be revised to include a provision that if evaluations of pipe wall thickness show a susceptibility to corrosion, further evaluation as to the extent of susceptibility will be performed.	LRA Section B2.1.5	Prior to Period of Extended Operation
14.	The Buried Piping and Tanks Inspection Program will be revised to specify a 10-year buried pipe inspection frequency.	LRA Section B2.1.5	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
15.	The Buried Piping and Tanks Inspection Program will be revised to specify a 10-year inspection frequency for Diesel Fuel Oil Storage Tank T-44.	LRA Section B2.1.5	Prior to Period of Extended Operation
16.	The Buried Piping and Tanks Inspection Program will be revised to include a review of previous buried piping issues to determine possible susceptible locations.	LRA Section B2.1.5	Prior to Period of Extended Operation
17.	The Bus Duct Inspection Program will be implemented consistant with the appropriate ten elements described in Appendix A of NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.	LRA Section B2.1.6	Prior to Period of Extended Operation
18.	The BWR Feedwater Nozzle Program will be enhanced so the parameters monitored and inspected are consistent with the recommendations of GE NE-523-A71-0594-A, Revision 1.	LRA Section B2.1.8	Prior to Period of Extended Operation
19.	The BWR Feedwater Nozzle Program will be enhanced so the regions being inspected, examination techniques, personnel qualifications, and inspection schedule are consistent with the recommendations of GE NE-523-A71-0594-A, Revision 1.	LRA Section B2.1.8	Prior to Period of Extended Operation
20.	The BWR Feedwater Nozzle Program will be enhanced so that inspections will be scheduled per recommendations of GE NE-523-A71-0594-A, Revision 1.	LRA Section B2.1.8	Prior to Period of Extended Operation
21.	The repair/replacement guidelines in BWRVIP-16, 19, 44, 45, 50, 51, 52, 57, and 58 will be added, as applicable, to the MNGP BWR Vessel Internals Program.	LRA Section B2.1.12	Prior to Period of Extended Operation
22.	NMC will perform top guide grid inspections using the EVT-1 method of examination, for the high fluence locations (grid beam and beam-to-beam crevice slot locations with fluence exceeding 5.0×10^{20} n/cm^2). Ten percent (10%) of the total population will be inspected within 12 years with a minimum of 5% inspected within the first 6 years.	LRA Section B2.1.12	During the Period of Extended Operation
23.	A one time inspection will be performed to monitor the effects of corrosion on select portions of closed-cycle cooling water systems that perform a pressure-integrity intended function.	LRA Section B2.1.13	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
24.	The Compressed Air Monitoring Program procedures will be revised to include corrective action requirements if the acceptance limits for water vapor, oil content, or particulate are not met. Also, the acceptance criteria for oil content testing will be clarified and the basis for the acceptance limits for the water vapor, oil content, and particulate tests will be provided.	LRA Section B2.1.14	Prior to Period of Extended Operation
25.	The Compressed Air Monitoring Program will be revised to include inspection of air distribution piping based on the recommendations of EPRI TR-108147.	LRA Section B2.1.14	Prior to Period of Extended Operation
26.	The MNGP Electrical Cables & Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program XI.E1. The program will manage the aging of conductor insulation material on cables, connectors, and other electrical insulation materials that are installed in an adverse localized environment caused by heat, radiation, or moisture.	LRA Section B2.1.15	Prior to Period of Extended Operation
27.	The Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will be implemented as a new program. With exceptions, it will be consistent with the recommendations of NUREG-1801 Chapter XI Program XI.E2.	LRA Section B2.1.16	Prior to Period of Extended Operation
28.	The MNGP Fire Protection Program will be revised to include a visual inspection of the halon fire suppression system to detect any signs of degradation, such as corrosion and mechanical damage. This visual inspection will provide aging management for external surfaces of the halon fire suppression system.	LRA Section B2.1.17	Prior to Period of Extended Operation
29.	The Fire Protection Program plan document will be revised to include qualification criteria for individuals performing visual inspections of penetration seals, fire barriers, and fire doors. The qualification criteria will be in accordance with VT-1 or equivalent and VT-3 or equivalent, as applicable.	LRA Section B2.1.17	Prior to Period of Extended Operation
30.	The Fire Water System Program implementing procedures will be revised to include the extrapolation of inspection results to below grade fire water piping with similar conditions that exist within the above grade fire water piping.	LRA Section B2.1.18	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
31.	The Fire Water System Program sprinkler heads will be inspected and tested per NFPA requirements or replaced before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.	LRA Section B2.1.18	Prior to Period of Extended Operation
32.	The Fire Water System Program will verify the procedures to be used for aging management activities of the Fire Water System apply testing in accordance with applicable NFPA codes and standards. Revise the relevant procedures as appropriate.	LRA Section B2.1.18	Prior to Period of Extended Operation
33.	The MNGP procedures related to the Diesel Fuel Oil System will be revised to include requirements to check for general, pitting, crevice, galvanic, microbiological influenced corrosion (MIC), and cracking.	LRA Section B2.1.20	Prior to Period of Extended Operation
34.	The MNGP Fuel Oil Chemistry Program procedures will be revised to require tank draining, cleaning, and inspection if deemed necessary based on the trends indicated by the results of the diesel fuel oil analysis, or as recommended by the system engineer based on equipment operating experience.	LRA Section B2.1.20	Prior to Period of Extended Operation
35.	Develop or revise existing procedures in the MNGP Fuel Oil Chemistry Program to require periodic tank inspections of the diesel fuel oil tanks.	LRA Section B2.1.20	Prior to Period of Extended Operation
36.	The MNGP Inaccessible Medium Voltage (2 kV to 34.5 kV) Cables Not Subject to 10 CFR 50.49 EQ Requirements Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program XI.E3.	LRA Section B2.1.21	Prior to Period of Extended Operation
37.	The Inspection of Overhead Heavy Load & Light Load (Related to Refueling) Handling Systems Program will be enhanced to specify a five-year inspection frequency for the fuel preparation machines.	LRA Section B2.1.22	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
38.	The MNGP One-Time Inspection Program will be implemented as a new program consistent with the recommendations of NUREG-1801 Chapter XI Program M32, "One-Time Inspection." This program will include measures to verify the effectiveness of the following aging management programs: Plant Chemistry Program and Fuel Oil Chemistry Program. This program will also confirm the absence of age degradation in selected components (e.g., flow restrictors, venturis) within License Renewal scope.	LRA Section B2.1.23	Prior to Period of Extended Operation
39.	The MNGP Protective Coating Maintenance and Monitoring Program procedures will be updated to include inspection of all accessible painted surfaces inside containment.	LRA Section B2.1.27	Prior to Period of Extended Operation
40.	The MNGP Protective Coating Maintenance and Monitoring Program will be revised to include a pre-inspection review of the previous two inspection reports so that trends can be identified.	LRA Section B2.1.27	Prior to Period of Extended Operation
41.	The MNGP Protective Coating Maintenance and Monitoring Program implementation procedures will be revised to include provisions for analysis of suspected reasons for coating failure.	LRA Section B2.1.27	Prior to Period of Extended Operation
42.	NMC intends to use the Integrated Surveillance Program for MNGP during the period of extended operation by implementing the requirements of BWRVIP-116, which is currently being reviewed by the NRC.	LRA Section B2.1.29	Prior to Period of Extended Operation
43.	NMC will retain the capsules removed from the MNGP reactor vessel as part of the Reactor Vessel Surveillance Program.	LRA Section B2.1.29	Prior to and during the Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
44.	The MNGP Selective Leaching of Materials Program will be implemented as a new program consistent, with exceptions, to the recommendations of NUREG-1801 Chapter XI Program M33, "Selective Leaching of Materials." The program will be developed and implemented before the start of the period of extended operation. The program includes a one-time visual inspection and hardness measurement of selected components that are susceptible to selective leaching. In situations where hardness testing is not practical, a qualitative method by other NDE or metallurgical methods will be used to determine the presence and extent of selective leaching. The program will determine if selective leaching is occurring for selected components.	LRA Section B2.1.30	Prior to Period of Extended Operation
45.	The MNGP Structures Monitoring Program will be expanded, as necessary, to include inspections of structures and structural elements in scope for License Renewal that are not inspected as part of another aging management program.	LRA Section B2.1.31	Prior to Period of Extended Operation
46.	The MNGP Structures Monitoring Program implementing procedures will be enhanced to ensure that structural inspections are~performed on submerged portions of the intake structure from the service water bays to the wing walls.	LRA Section B2.1.31	Prior to Period of Extended Operation
47.	The MNGP Structures Monitoring Program implementing procedures will be revised to include the monitoring/inspection parameters for structural components within the scope of License Renewal.	LRA Section B2.1.31	Prior to Period of Extended Operation
48.	The MNGP Structures Monitoring Program will be enhanced to include a requirement to sample ground water for pH, chloride concentration and sulfate concentration.	LRA Section B2.1.31	Prior to Period of Extended Operation
49.	The MNGP Structures Monitoring Program will be enhanced to include concrete evaluations of inaccessible areas if degradation of accessible areas is detected.	LRA Section B2.1.31	Prior to Period of Extended Operation
50.	The MNGP Structures Monitoring Program implementing procedures will be enhanced to include acceptance criteria for structural inspections of submerged portions of the Intake Structure.	LRA Section B2.1.31	Prior to Period of Extended Operation

ITEM	COMMITMENT	SOURCE	SCHEDULE
51.	Implementing instructions and procedures for the System Condition Monitoring Program will be revised to describe specific age degradation parameters to be monitored and inspected. Acceptance criteria will also be included.	LRA Section B2.1.32	Prior to Period of Extended Operation
52.	Incorporate requirements for inclusion of NUREG/CR-6260 locations in implementing procedures for the MNGP Thermal Fatigue Monitoring Program.	LRA Section B3.2	Prior to Period of Extended Operation

Appendix A References:

1. NEDO-32205A, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels, Revision 1, February 1994 as transmitted by BWROG-94037, L.A. England (BWROG) to Daniel G. McDonald (NRC), "BWR Owners' Group Topical Report on Upper Shelf Energy Equivalent Margin Analysis - Approved Version," March 21, 1994.
2. 10 CFR 54.3, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants-Definitions," May 8, 1995.
3. Letter, S.A. Richard (NRC) to J.F. Klapproth (GE), "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," (TAC No. MA9891) MFN 01-050, September 14, 2001.
4. C.I. Grimes (NRC) to Carl Terry (Niagara Mohawk Power Company), "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)' and Appendix A, 'Demonstration Of Compliance With The Technical Information Requirements Of The License Renewal Rule (10 CFR 54.21)'," October 18, 2001.
5. EPRI report TR-1003346, "Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," October 2002 (BWRVIP-86-A) (EPRI Proprietary).
6. MNGP Updated Safety Analysis Report, Revision 21, December 15, 2004.
7. Ranganath, S., "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper G1/5.
8. Appendix A to Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition.
9. EPRI TR-108279, "Fracture Toughness and Tensile Properties of Irradiated Austenitic Stainless Steel Components Removed from Service," (BWRVIP-35), EPRI, Palo Alto CA, June 1997 (EPRI Proprietary Information).
10. Gus C. Lainas (NRC) to Carl Terry (Niagara Mohawk Power Company), BWRVIP Chairman, "Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report," (TAC No. M93925), July 28, 1998.
11. EPRI Report TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), September 28, 1995, with supplementing letters of June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998. (EPRI Proprietary Information)
12. Jack R. Strosnider (NRC), to Carl Terry (BWRVIP Chairman), "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," (TAC No. MA3395), March 7, 2000.
13. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," November 1980.

14. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

15. EPRI Report 1007824, "Integrated Surveillance Program (ISP) Implementation for License Renewal," July 2003 (BWRVIP-116)(EPRI Proprietary).

16. DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors, U.S. Nuclear Regulatory Commission, June 1979.

17. Monticello PUAR, Volume 3, Vent System Analysis, NSP-74-103, Revision 1, November 1982.

18. Monticello PUAR, Volume 5, Safety Relief Valve Discharge Piping Analysis, NSP-74-105, Revision 1, November 1982.

19. Mark I Plant Unique Load Definition, Monticello Nuclear Generating Plant, NEDO-24576, Rev. 1, October 1981.

20. Mark I Containment Program Load Definition Report, NEDO-21888, Revision 2 December 1981.

ENCLOSURE 3

Response to NRC Questions During Meeting to Discuss the Monticello Nuclear Generating Plant License Renewal Application

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted an application to renew the operating licenses for the Monticello Nuclear Generating Plant (MNGP), dated March 16, 2005. As a result of the NRC's questions during the NMC meeting with the NRC to discuss the MNGP License Renewal Application (LRA), NMC is hereby providing responses to the following NRC staff questions:

NRC Question

- Determine the need for an inspection of the Top Guide in addition to the requirements of Boiling Water Reactor Vessel Internals Program (BWRVIP) - 26.

NMC Response

In response to the NRC Staff's request concerning the need to augment the Top Guide inspections NMC has determined that top guide grid inspections, of the high fluence locations, are prudent for the license renewal period of extended operation. This determination was reached in consideration of the NRC Staff's current position with respect to the conclusions contained in BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines." It is worthy to note, however, that MNGP has performed an inspection of the high fluence areas. These inspections were performed on approximately a twenty-five percent sampling of the high fluence locations of the top guide grid. Fifteen (15) cells were inspected and no evidence of cracking was noted. NMC agrees, however, to reinstate an inspection of the high fluence areas of the top guide grid, concurrent with the implementation of the license renewal period of extended operation for MNGP. The examination procedures implementing the MNGP Reactor Vessel Internals Program (LRA, Appendix B, Section B2.1.12) will be revised to include this inspection of areas of high fluence (grid beam and beam-to-beam crevice slot locations with fluence exceeding 5.0×10^{20} n/cm^2). The inspection will be similar to the inspections required for the Control Rod Guide Tube, which is performed in accordance with BWRVIP-47 (BWR Lower Plenum Inspection and Flaw Evaluation Guidelines). Ten percent (10%) of the total population will be inspected within 12 years with a minimum of 5% inspected within the first 6 years. NMC will conduct these inspections using the EVT-1 method of examination.

This aging management approach was accepted by the NRC in NUREG-1796, Safety Evaluation Report Related to the License Renewal of Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, in paragraph 3.1.2.3.6.

NRC Question

- Commit to retaining the capsules removed from the reactor vessel as part of the Reactor Vessel Surveillance Program.

NMC Response

NMC will retain the capsules removed from the MNGP reactor vessel as part of the Reactor Vessel Surveillance Program. As noted in the MNGP LRA, the MNGP Reactor Surveillance Program is required by 10 CFR 50, Appendix H, and is implemented in accordance with BWRVIP-86-A, "BWRVIP Integrated Surveillance Program Plan," for the current licensed operating period and BWRVIP-116, "BWRVIP Integrated Surveillance Program (ISP) Implementation for License Renewal," for the license renewal period of extended operation. Since MNGP is one of the host plants for the ISP, both (2) remaining capsules will be retained in-vessel until removed for testing in accordance with the ISP. Testing will be conducted in accordance with ASTM E-185. All capsule specimens will be retained in the event that additional analyses are required. Any changes to the capsule withdrawal schedule, or storage requirements will be submitted to the NRC for approval, as required by 10 CFR 50, Appendix H.

MNGP LRA Appendix B, Section B2.1.29, Reactor Vessel Surveillance Program, identifies that no exceptions to the GALL are taken for this program. This includes conformance with GALL program XI.M31 item 4, which states, "All pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program is re-established.)"

NRC Question

- Confirm that the 54 Effective Full Power Years (EFPY) used for Time-Limited Aging Analyses bounds the plant-specific EFPY based on historic and predicted capacity factors.

NMC Response

NMC has confirmed that the 54 Effective Full Power Years (EFPY) used for Time-Limited Aging Analyses bounds the plant-specific EFPY for MNGP based on a conservative evaluation of plant history and projected capacity factors. This evaluation results in an expectation of less than 49.5 EFPY at the end of the license renewal period of extended operation. Assuming a 100% capacity factor over the same operating period also results in a projected less than 54 EFPY for MNGP.