4.0 TIME-LIMITED AGING ANALYSES

Two areas of plant technical assessment are required to support an application for a renewed operating license. The first area of technical review is the Integrated Plant Assessment, which is described in Section 2.0 and Section 3.0 of this License Renewal Application. The second area of technical review is the identification and evaluation of time-limited aging analyses (TLAAs) and exemptions. The identifications and evaluations included in this section meet the requirements contained in 10 CFR 54.21(c) and provide the information necessary for the NRC to make the finding contained in 10 CFR 54.29(a)(2).

4.1 Time-Limited Aging Analysis Process

Title 10 of the Code of Federal Regulations, Part 54 (10 CFR 54) sets forth the requirements for License Renewal of Operating Nuclear Power Plants. 10 CFR 54.21(c)(1) requires a listing and an evaluation of TLAAs. 10 CFR 54.21(c)(2) requires a listing and evaluation of active plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs as defined in 10 CFR 54.3(a). The overall TLAA methodology is provided in Figure 4.1-1.

4.1.1 Identification Process for Time-Limited Aging Analyses

This section documents the identification and disposition of TLAAs, including TLAA related exemptions granted in accordance with 10 CFR 50.12, which are applicable to the Monticello Nuclear Generating Plant (MNGP) for the period of extended operation.

TLAAs are defined in 10 CFR 54.3 as those licensee calculations and analyses that:

1. Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);
2. Consider the effects of aging;
3. Involve time-limited assumptions defined by the current operating term, for example, 40 years;
4. Were determined to be relevant by the licensee in making a safety determination;
5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CFR 54.4(b); and
6. Are contained or incorporated by reference in the current licensing basis (CLB).

Potential TLAAs, which could meet the 6 criteria, can be identified in two ways:

- Reviewing lists of previously identified TLAAs and choosing those generically applicable to MNGP for further evaluation.
- Searching the MNGP CLB for calculations/analyses that contain a time-sensitive element.

4.1.1.1 TLAA Industry Related Search

Industry License Renewal related documents, including previous applications by other plants, have already identified several TLAAs. These TLAAs tend to be generically applicable to other similar type plants (e.g. BWR plants). These documents were searched to identify a list of known TLAAs, which could be potentially applicable to MNGP.

Methodology

The following documents were searched for TLAAs which could potentially be applicable to MNGP:

- Generic Aging Lessons Learned Report, NUREG-1801
- Standard Review Plan for License Renewal, NUREG-1800, Chapter 4
- NEI 95-10, Industry Guidance for Implementing the Requirements of 10 CFR 54 - The License Renewal Rule
- Boiling Water Reactor Owners Group (BWROG) generic technical reports
- Previously submitted License Renewal Applications for other plants

4.1.1.2 TLAA Current Licensing Basis (CLB) Document Search

The Current Licensing Basis (CLB) documents were searched to determine if any potential TLAAs not previously identified by the industry may exist for MNGP.

Methodology

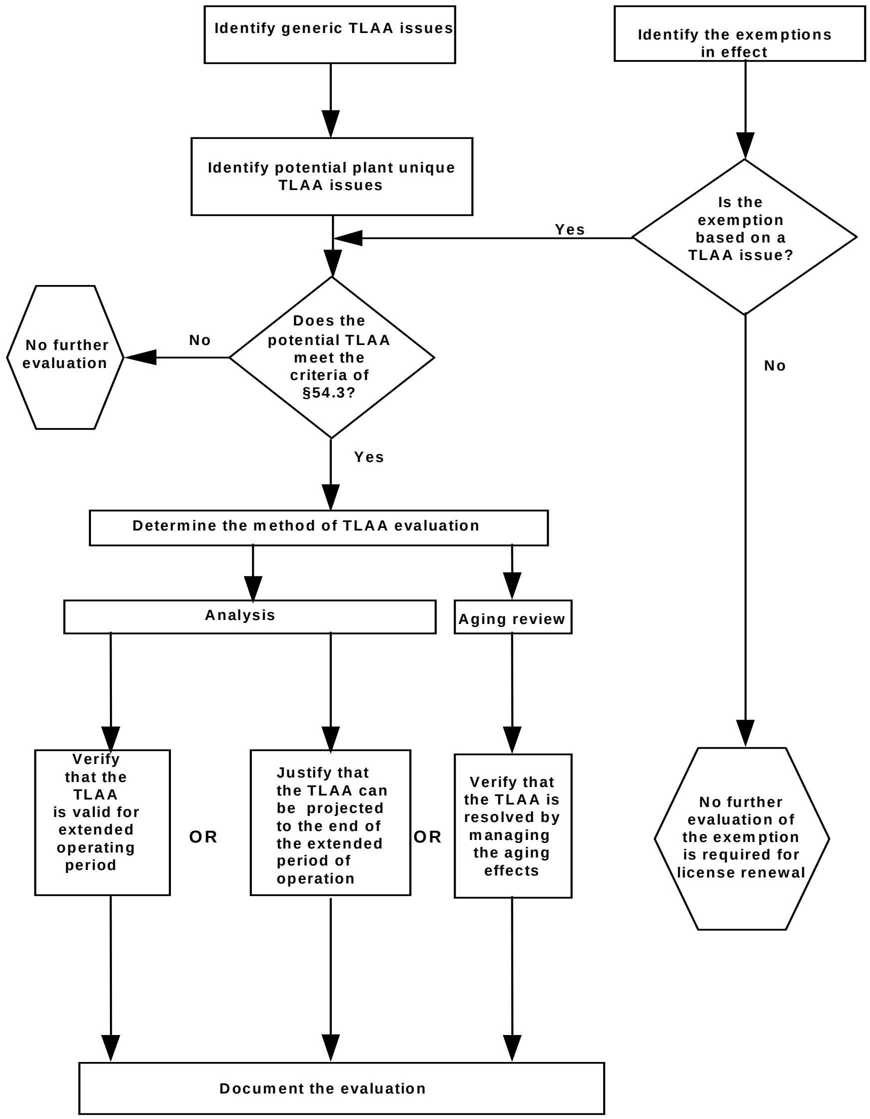
The following documents were searched electronically for keywords which are indicative of a time-limited element:

- MNGP Updated Safety Analysis Report (USAR)
- Technical Specifications
- Docketed Correspondence
- MNGP Calculations, Analyses, and Reports

In addition, Aging Management Review (AMR) evaluations also included identification of potential TLAAs.

Each document identified in the above activities as containing a potential TLAA was reviewed to determine if it was applicable to MNGP. The plant-specific TLAAs were then compared to those potential TLAAs identified through the Industry search to identify a comprehensive list of TLAAs (see Section 4.1.2 for a discussion of the impact of Exemptions on TLAAs).

Figure 4.1-1 TLAA Methodology



4.1.2 Identification of 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 in effect based on time-limited aging analyses as defined in 10 CFR 54.3. As shown in Figure 4.1-1, the TLAA identification process includes exemptions granted by the NRC.

Active 10 CFR 50.12 exemptions were reviewed to determine if the exemption was based on an assumption not previously identified as applicable to a TLAA. This review included keyword searches (e.g. 50.12, exemption etc.) of the MNGP CLB documentation.

Most of the identified items were related to 10 CFR 50, Appendix R; 10 CFR 50, Appendix J; ASME Code and inspection requirements; and schedule exemptions.

No additional plant specific exemption TLAAs were identified by these searches.

4.1.3 Evaluation Process of Time-Limited Aging Analyses

Each potential TLAA identified in Section 4.1.1 was screened against the six criteria of 10 CFR 54.3(a). Once a TLAA was identified as applicable to MNGP, an evaluation was performed, as required by 10 CFR 54.21(c)(1), to demonstrate that at least one of the following criteria was applicable:

i. The analyses remain valid for the period of extended operation.

ii. The analyses have been projected to the end of the period of operation.

iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The results of “generic” industry information and MNGP specific documentation resulted in the TLAAs identified in Table 4.1-1 and are discussed in Section 4.2 through Section 4.10.

4.1.4 1998 MNGP Power Rerate

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.

Table 4.1-1 List of MNGP Time-Limited Aging Analyses (TLAAs)

TLAA Category	Number	TLAA	Section	Disposition 10 CFR 54.21(c)(1)	Comments
Neutron Embrittlement of the Reactor Vessel and Internals	1	RPV Materials USE Reduction Due to Neutron Embrittlement	4.2.1	Analyses have been projected to the end of the period of extended operation	
	2	Adjusted Reference Temperature (ART) for RPV Materials Due to Neutron Embrittlement	4.2.2	Analyses have been projected to the end of the period of extended operation	
	3	Reflood Thermal Shock Analysis of the RPV	4.2.3	The analyses remain valid for the period of extended operation	
	4	Reflood Thermal Shock Analysis of the RPV Core Shroud	4.2.4	The analyses remain valid for the period of extended operation	
	5	RPV Thermal Limit Analysis: Operating P-T Limits	4.2.5	Analyses have been projected to the end of the period of extended operation	
	6	RPV Circumferential Weld Examination Relief	4.2.6	Analyses have been projected to the end of the period of extended operation	
	7	RPV Axial Weld Failure Probability	4.2.7	Analyses have been projected to the end of the period of extended operation	

Table 4.1-1 List of MNGP Time-Limited Aging Analyses (TLAAs)

TLAA Category	Number	TLAA	Section	Disposition 10 CFR 54.21(c)(1)	Comments
Metal Fatigue - RPV, Internals and Pressure Boundary	8	RPV Fatigue Analyses	4.3.1	Analyses have been projected to the end of the period of extended operation and effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
	9	Fatigue Analysis of RPV Internals	4.3.2	Analyses have been projected to the end of the period of extended operation	
	10	ASME Section III Class 1 Reactor Coolant Pressure Boundary (RCPB) Piping and Fatigue Analysis	4.3.3	The analyses remain valid for the period of extended operation and effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
	11	RCPB Section III Class 2 and 3, USAS B31.1 Piping and Components	4.3.4	The analyses remain valid for the period of extended operation	
Neutron Embrittlement	12	Irradiation Assisted Stress Corrosion Cracking	4.4	Effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
Environmental Fatigue	13	Effects of Reactor Coolant Environment	4.5	Analyses have been projected to the end of the period of extended operation	

Table 4.1-1 List of MNGP Time-Limited Aging Analyses (TLAAs)

TLAA Category	Number	TLAA	Section	Disposition 10 CFR 54.21(c)(1)	Comments
Fatigue of Primary Containment, Piping, and Components	14	Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers	4.6.1	The analyses remain valid for the period of extended operation and effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
	15	Fatigue Analysis of the SRV Piping Inside the Suppression Chamber and Internal Structures	4.6.2	The analyses remain valid for the period of extended operation	
	16	Fatigue Analysis of Suppression Chamber External Piping and Penetrations	4.6.3	The analyses remain valid for the period of extended operation and effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
	17	Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analysis	4.6.4	The analyses remain valid for the period of extended operation	
	18	Primary Containment Process Penetration Bellows Fatigue Analysis	4.6.5	The analyses remain valid for the period of extended operation	

Table 4.1-1 List of MNGP Time-Limited Aging Analyses (TLAAs)

TLAA Category	Number	TLAA	Section	Disposition 10 CFR 54.21(c)(1)	Comments
Environmental Qualification	19	Environmental Qualification of Electrical Equipment (EQ)	4.7	Effects of aging on the intended function(s) will be adequately managed for the period of extended operation	
Loss of Preload	20	Stress Relaxation of Core Plate Rim Holddown Bolts	4.8	Analyses have been projected to the end of the period of extended operation	
Plant Specific TLAAs	21	Concrete Containment Tendon Prestress	NA	NA	MNGP containment design does not include prestress tendons. Consequently, this NUREG-1800 potential TLAA is not applicable to MNGP.
	22	Reactor Building Crane Load Cycles	4.9	The analyses remain valid for the period of extended operation.	
	23	Fatigue Analyses of High Pressure Coolant Injection and Reactor Core Cooling Turbine Exhaust Penetrations	4.10	The analyses remain valid for the period of extended operation.	

Table 4.1-1 List of MNGP Time-Limited Aging Analyses (TLAAs)

TLAA Category	Number	TLAA	Section	Disposition 10 CFR 54.21(c)(1)	Comments
Plant Specific TLAAs	24	High-Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor	NA	NA	Break locations postulated on pipe size and time of operation, not fatigue criteria. Consequently, this NUREG-1800 potential TLAA is not applicable to MNGP.
	25	Inservice Flaw Growth Analysis that Demonstrates Structure Stability for 40 Years	NA	NA	Flaw evaluations have been completed to determine inspection intervals, not acceptability through end of licensed term. Consequently, this NUREG-1800 potential TLAA is not applicable to MNGP.

4.2 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 (RT_{NDT}), 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 RT_{NDT} and a decrease in the USE of RPV steels. The increase or shift in the initial nil-ductility reference temperature (ΔRT_{NDT}) 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 RT_{NDT} and ΔRT_{NDT} 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 RT_{NDT} (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 4.2 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

4.2.1 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 in NEDO-32205-A (Reference 1) for all other materials. 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 3.90×10^8 MWh through the end of Cycle 22 at 1775 MWt plus 4.76×10^8 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 4.2.1-1 and Table 4.2.1-2 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 and BWRVIP-116, Reference 5 and Reference 15, respectively).

Table 4.2.1-1 Equivalent Margin Analysis for MNGP Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.17 1st Capsule Fluence = 2.93 x 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 x 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 4.2.1-2 Equivalent Margin Analysis for MNGP Weld Material

BWR/3-6 WELD
Surveillance Weld USE: %Cu = 0.04 1st Capsule Fluence = 2.93 x 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 x 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.

4.2.2 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. RT_{NDT} 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 (ΔRT_{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} + ΔRT_{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 4.2.2-1. 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 the 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 4.2.2-1. 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 4.2.2-1). 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 4.2.2-1 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 4.2.2-1 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 X 10^{18} n/cm^2					
							EFPY Peak 1/4 T Fluence = 2.51 X 10^{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 X 10^{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 X 10^{17} n/cm^2					
							EFPY Peak 1/4 T Fluence = 5.23 X 10^{17} n/cm^2					
COMPONENT	HEAT	%Cu	%Ni	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	3.82 X10^{18} 3.82 X10^{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	3.82 X10^{18}	102	12.7	28	61	163	97
NOZZLES:												
N2*	E21VW	0.18	0.86	141.9	40	5.23 X10^{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 MNGP N2 nozzles was averaged to determine the Ni content. CMTR data for the ten MNGP N2 nozzles was used to determine the initial RT_{NDT}.

4.2.3 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 x 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 x 10^{18} n/cm^2 for 54 EFPY of operation (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). 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 1/4 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 1/4 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 described in Section 4.2.2 and tabulated in Table 4.2.2-1 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$\sqrt{in}$) 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.

4.2.4 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 flux at end of life at the inside surface of the shroud is anticipated to be 2.7 x 10^{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 x 10^{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 is 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 in Section 4.2.2, 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.

4.2.5 RPV Thermal Limit Analysis: 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 heatup/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.

Analysis

The MNGP Technical Specifications contain P-T limit curves for heatup/cooldown, in-service leakage and hydrostatic testing. They also limit the maximum rate of change of reactor coolant temperature. The criticality curves provide limits for both heatup 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.

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

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

4.2.6 RPV Circumferential Weld Examination Relief

Summary Description

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.

Analysis

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 4.2.6-1 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

4.2.6-1 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.

The Mean RT_{NDT} value at 54 EFPY is bounded by the 64 EFPY Mean RT_{NDT} 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 4.2.6-1 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[i]	1.78×10^{-5}	[ii]
P (F/E) BWRVIP	-	-

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

4.2.7 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 in Section 4.2.6.

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 in Section 4.2.6, 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 4.2.7-1 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 x 10^{-6} RPV failure frequency.

For MNGP, the fluence value is significantly 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 4.2.7-1 Effects for Irradiation on RPV Axial Weld Properties for MNGP

Value	Mod 2	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[i]	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 x 10^{-6}	[ii]

i. Peak Axial Fluence
ii. 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.

4.3 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.

4.3.1 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 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 (Reference 16). These revised 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 4.3.1-1.

Table 4.3.1-1 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 the MNGP USAR (Reference 6), Table 4.2-1, which identifies the following transient cycles:

MNGP Transient Cycles

Transient Type	No. of Design Cycles (Reference 6)	Projected to 2030
Bolt Up/Unbolt	120	44
Startup/Shutdown @ 100^{o}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^{o}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 in Reference 6 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 4.3.1-1.

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 4.3.1-1. With environmental fatigue considered, 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.

4.3.2 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.

4.3.3 ASME Section III Class 1 Reactor Coolant Pressure Boundary (RCPB) Piping and 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 (see Section 4.6) 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 4.3.3-1.

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. Therefore, a cycle based counting approach was used to evaluate these systems.

Cycle based counting consists of periodically counting the relevant cycles and calculating the cumulative usage factor (CUF). This process is 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 4.3.3-1 for the Recirculation and RHR piping systems are expected not to exceed 0.90 at the end of sixty years of plant operation.

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

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, x 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.

Table 4.3.3-1 MNGP Fatigue 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

4.3.4 RCPB Section III Class 2 and 3, 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 4.6), 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 SA for 7,000 equivalent full temperature thermal cycles or less and is incrementally reduced to 0.5 S_A for greater than 100,000 cycles. With the exception of piping described in Section 4.3.3 and Section 4.6 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)

A conservative estimate of the number of thermal cycles experienced by these piping systems can be approximated by using the maximum number of thermal cycles 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 (Reference 26). 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.

4.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

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 power rerate. 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.

4.5 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 six 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 separately 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 (Reference 14), “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.”

In order to satisfy 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 the following table, 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.

Location	Component	Material	Usage Factor (60 year 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 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.

4.6 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 Analysis

4.6.1 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 19) 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, MNGP re-evaluated all locations 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 decay 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 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. $U_{max,40} < 0.20 \times 60/40 = U_{max,60} = 0.30 << 1.0$).

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 lift coincidentally 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.

4.6.2 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 (PUAR) (Reference 22) 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 22) describes the evaluation of the SRVDL piping system evaluation. 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 24). The analysis also considered the interaction effects of the vent system and the suppression chamber.

Per Reference 22, 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, MNGP re-evaluated this location 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 decay 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:

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

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.

4.6.3 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 described in Reference 23, and as defined in the NRC's Safety Evaluation Report NUREG-0661 and in the Mark I Containment Load Definition Report (LDR) (Reference 25). 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 (PUAAG). 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 (Reference 27). 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. MNGP re-evaluated the SRV discharge piping analysis 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 decay 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 analysis was conservatively evaluated for the effects of rerate by increasing the SRV cycles by a factor of 1.26 to correspond to a power level of 1880 MWt (the actual rerate level is 1775 MWt which corresponds to a 1.14 SRV factor). This conservative application 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.

4.6.4 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 suppression chamber 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-year design life, 300 startup/shutdown cycles and one 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.

4.6.5 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.

4.7 Environmental Qualification of Electrical Equipment (EQ)

Summary Description

10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants (Reference 17) 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 18), NUREG-0588 (Reference 20) Regulatory Guide 1.89 (Reference 21) and in Generic Letter 82-09 (Reference 28).

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 manages 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 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 total integrated 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.

4.8 Stress Relaxation of Rim Holddown Bolts

Summary Description

As described in the SER to BWRVIP-25, plants must consider relaxation of the rim holddown 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 holddown 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 holddown 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.

4.9 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 Crane Manufactures Association of America (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.

4.10 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 Design Basis Accident Condensation Oscillation (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, 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).

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