CHAPTER 18

AGING MANAGEMENT PROGRAMS AND ACTIVITIES

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18.0 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

18.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment and the time-limited aging analyses for license renewal identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the programs and their implementation activities. RNP will employ the Corrective Action and Document Control Programs to address the program elements of corrective action, confirmation process, and administrative (document) controls for both safety related and non-safety related structures and components that perform an intended function for license renewal.

18.1.1 ASME SECTION XI, SUBSECTION IWB, IWC, AND IWD PROGRAMS

The ASME Section XI, Subsection IWB, IWC, and IWD Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. The RNP Fourth Ten-Year Interval Program was developed and prepared to meet the ASME Code, Section XI, 1995 Edition through 1996 Addenda.

RNP will continue to perform Inservice Inspection (ISI) examinations for the reactor coolant loop piping, valve, and pump casings as required by Table IWB-2500-1 of Section XI of the ASME Boiler and Pressure Vessel Code, unless relief has been granted by the NRC under applicable provisions in 10 CFR 50.55a from meeting the staff's ISI requirements of 10 CFR 50.55a(g)(4).

This program is consistent with the corresponding program described in the GALL Report (Reference 18.1.1).

18.1.2 WATER CHEMISTRY PROGRAM

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion and contaminants that may cause loss of heat transfer effectiveness due to fouling of heat transfer surfaces. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. Alternatively, chemical agents, such as corrosion inhibitors, oxygen scavengers, and biocides, may be introduced to prevent certain aging mechanisms. The RNP Water Chemistry Program is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines as prescribed by NEI 97-06.

This program is consistent with the corresponding program described in the GALL Report.

18.1.3 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is credited for aging management of the Reactor Head Closure Studs and Stud Components for the aging effects/mechanisms of concern: (1) Loss of Pre-load Due to Stress Relaxation, and (2) Loss of Material Due to Wear. The closure studs, nuts, and washers are included within the scope of the ASME Section XI, Subsections IWB, IWC, and IWD Program.

This program is consistent with the corresponding program described in the GALL Report.

18.1.4 STEAM GENERATOR TUBE INTEGRITY PROGRAM

The Steam Generator (SG) Tube Integrity Program specifies inspection scope, frequency, and acceptance criteria for the plugging and repair of flawed SG tubes in accordance with the plant Technical Specifications and the guidance of NEI 97-06. Other components, in addition to SG tubes, are inspected under this program. As part of the existing program, RNP will evaluate the details of new revisions to NEI 97-06 as they are released to determine if exceptions are needed. The process of evaluating changes to the Steam Generator Tube Inspection Program will continue during the period of extended operation.

This program is consistent with the corresponding program described in the GALL Report.

18.1.5 CLOSED-CYCLE COOLING WATER SYSTEMS PROGRAM

The program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing monitoring consisting of inspection and performance evaluations. Concentrations of corrosion inhibitors are maintained in accordance with the guidelines of EPRI-TR-1007820. Performance monitoring of diesel generator jacket water cooling systems is accomplished as part of regularly scheduled operation and testing of that equipment. Inspections and performance monitoring associated with the CCW heat exchangers are addressed in the Open Cycle Cooling Water System Program.

This program is consistent with the corresponding program described in the GALL Report.

18.1.6 ASME SECTION IX, SUBSECTION IWF PROGRAM

This program consists of periodic visual examination of component supports for signs of degradation. The examination requirements for the IWF portion of the RNP Inservice Inspection (ISI) program are taken from paragraph IWF-2500 (1995 Edition), which are essentially the same as specified in Table IWF-2500-1 (1989 Edition). The RNP Fourth Ten-Year Interval ISI Program was developed and prepared to meet the ASME Code, Section XI, 1995 Edition through 1996 Addenda, and is subject to the limitations and modifications of 10 CFR 50.55a(b)(2), with the exception of design and access provisions and preservice examination requirements.

This program is consistent with the corresponding program described in the GALL Report.

18.1.7 10 CFR PART 50, APPENDIX J PROGRAM

This program consists of inspections of accessible surfaces of containment and monitoring of leakage rates through containment liner/welds, penetrations, fittings, and access openings for detecting degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. This program is implemented in accordance with 10 CFR Part 50, Appendix J,
Regulatory Guide 1.163, and NEI 94-01, Rev. 0.

This program is consistent with the corresponding program described in the GALL Report.

18.1.8 FLUX THIMBLE EDDY CURRENT INSPECTION PROGRAM

The Flux Thimble Eddy Current Inspection Program is a plant-specific program that determines the amount of wear on the flux thimbles and whether the amount of wear expected to occur during the next inspection interval will cause the total amount of wear to exceed the ASME standards specified for the examination. The Flux Thimble Eddy Current Inspection Program was implemented to satisfy NRC Bulletin 88-09 requirements that a thimble tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation.

Additional details regarding flaw acceptance criteria and inspection methodology are provided in a letter from G. Vaughn (CP&L) to NRC, Serial NLS-91-024: “Response to NRC Bulletin No. 88-09,” dated February 8, 1991. Examination frequency details are provided in letter, Serial RNP-RA/09-0025, “Status Update for NRC Bulletin No. 88-09, ‘Thimble Tube Thinning in Westinghouse Reactors,’” dated April 13, 2009.

18.1.9 FIRE PROTECTION PROGRAM

The Fire Protection Program manages the aging effects as applicable for fire barriers (fire barrier walls, ceilings, and floors, penetration seals and fire rated doors) and non-water-based fire suppression systems (Halon and carbon dioxide). It includes pump testing in the aging management strategy for the diesel-driven fire water pump fuel supply line. The Fire Protection Program requires periodic visual inspection of fire barrier penetration seals and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection requirements require that the pump be periodically tested to ensure that the fuel supply line can perform the intended function. The program also includes periodic inspection and test of halon and carbon dioxide fire suppression systems.

The Fire Protection Program has been enhanced to note that concrete surface inspections performed under structures monitoring procedures are credited for inspection of fire barrier walls, ceilings, and floors.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.10 BORIC ACID CORROSION CONTROL PROGRAM

The Boric Acid Corrosion Control Program manages the aging effects for susceptible materials of structures and components that perform a license renewal intended function and that are exposed to the effects of borated water leaks. The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy of corrective actions. This program is implemented in response to NRC Generic Letter 88-05.

The scope of the Boric Acid Corrosion Control Program has been expanded to (1) ensure the mechanical, structural, and electrical components in scope for license renewal are addressed, and (2) identify additional areas in which components may be susceptible to exposure from boric acid (e.g., containment, auxiliary, and spent fuel buildings).

This program is consistent with the corresponding program described in the GALL Report.

18.1.11 FLOW-ACCELERATED CORROSION PROGRAM

The program consists of the following actions: (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning, (3) replace/repair components, and (4) perform follow up inspections to confirm or quantify and take longer-term corrective actions as necessary. Originally, this program was prepared in response to NRC Generic Letter 89-08. The program relies on implementation of the EPRI guidelines provided by NSAC-202L (Latest Revision).

The Flow-Accelerated Corrosion Program has been modified to (1) include additional components potentially susceptible to FAC and/or erosion, and (2) specify corrective actions be taken in accordance with the Corrective Action Program when certain acceptance criteria are not met.

This program is consistent with the corresponding program described in the GALL Report.

18.1.12 BOLTING INTEGRITY PROGRAM

This program consists of guidelines on materials selection, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and inspection techniques. The program relies on other aging management programs for management of specific aging effects for specific components. The Section XI, Subsection IWB, IWC and IWD Program is credited with inspections of bolting within Section XI boundaries. In addition, the Preventive Maintenance Program performs regular inspections of Reactor Coolant Pump bolting. Loss of Mechanical Closure Integrity Due to Loss of Material Due to Aggressive Chemical Attack (Boric Acid Wastage) has also been identified as a potential aging mechanism for mechanical system bolted closures at RNP. Aging management reviews have directly credited the Boric Acid Corrosion Control Program for management of this aging mechanism. Otherwise, from the standpoint of loss of material due to corrosion, bolting on mechanical components is treated as a subcomponent (i.e., a part of the parent component), and the Systems Monitoring Program is utilized to manage loss of material.

The Bolting Integrity Program is not utilized for aging management of structural bolting. The ASME Section XI, Subsection IWF Program is credited for aging management of structural bolting associated with Class 1, 2, and 3 component supports, and the Structures Monitoring Program is credited for aging management of all structural bolting other than those associated with Class 1, 2, and 3 components.

Program administrative controls have been modified as necessary to specifically prohibit the use of MoS_2 compounds in high strength bolting applications. Also, a program enhancement was implemented to inspect and evaluate bolting on CVC-381 prior to the end of the current operating period to address susceptibility to cracking.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.13 OPEN CYCLE COOLING WATER SYSTEM PROGRAM

The program includes (1) surveillance and control of biofouling, (2) monitoring, inspecting, and cleaning to verify heat transfer, (3) routine inspection and maintenance, (4) system walk-down inspection, and (5) review of maintenance, operating, and training practices and procedures. The program provides assurance that aging effects for the open-cycle cooling water system can be managed for an extended period of operation. This program was originally developed in response to NRC Generic Letter 89-13.

The ECCS room coolers have been replaced with stainless steel coils, which eliminates potential aging effects. The program includes periodic eddy current testing of each ECCS room cooler.

This program is consistent with the corresponding program described in the GALL Report.

18.1.14 INSPECTION OF OVERHEAD HEAVEY LOAD AND LIGHT LOAD HANDLING

The program provides guidelines and inspection attributes for monitoring the physical condition of crane structures within the scope of license renewal. Rails and girders are visually inspected on a routine basis for degradation. Functional testing requirements are specified. These cranes must also comply with the Maintenance Rule requirements provided in 10 CFR 50.65.

Administrative controls for Inspection of Overhead Heavy Load and Light Load Handling equipment have been enhanced to (1) include requirements for inspecting the Turbine Gantry Crane in addition to the other cranes that require inspection, (2) note that cranes are to be inspected using the attribute inspection checklist for structures, and (3) revise the attribute inspection checklist for structures to include GALL terminology, such as wear.

This program is consistent with the corresponding program described in the GALL Report.

18.1.15 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program manages the aging effects of Loss of Material and Flow Blockage due to fouling of fire protection system water flow paths. To ensure no significant corrosion, MIC, or biofouling has occurred in the water-based fire protection system, periodic full flow flush testing, system performance testing, and inspections during maintenance are conducted. Also, the system is normally maintained at required operating pressure and is monitored such that loss of system pressure would be detected and corrective actions initiated. The program relies on testing of water-based fire protection system piping and components in accordance with applicable National Fire Protection Association (NFPA) requirements.

This program has been modified to include:

Fire Protection Sprinkler Systems

(1) For sprinkler heads in service for 50 years, either sprinkler head replacement or sampling/field service testing of heads in accordance with NFPA 25 requirements based on the in-service date of the affected systems, and (2) full flow testing of portions of fire protection wet pipe sprinkler systems through the system cross mains, which are not routinely subject to flow, at the greatest flow and pressure allowed by the design of the systems or, alternatively, inspections or ultrasonic (UT) testing of a representative sample of these systems has been performed. Results from initial tests or inspections, reflecting 40 years of service, were used to determine the scope and subsequent test/inspection intervals. The intervals are not expected to exceed 10 years.

Fire Protection Suppression Piping

UT examination on a representative sampling of the above ground fire protection piping normally containing water was performed. Each sampling included different sections of piping. Alternatively, internal inspections were conducted on a representative sampling of these piping systems. Results from initial tests or inspections, reflecting 40 years of service, were used to determine the scope and subsequent test/inspection intervals. The intervals are not expected to exceed 10 years.

Halon/Carbon Dioxide Fire Suppression Systems

The NRC staff guidance with respect to halon/carbon dioxide fire suppression systems as documented in a letter from C. Grimes (NRC) to A. Nelson (Nuclear Energy Institute) and D. Lochbaum (Union of Concerned Scientists), Proposed Staff Guidance on Aging Management of Fire Protection Systems for License Renewal, dated January 28, 2002 determined that activities related to halon/carbon dioxide fire suppression systems are operational activities and not aging management programs. Therefore, no additional actions were required.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.16 BURIED PIPING AND TANKS SURVEILLANCE PROGRAM

The Buried Piping and Tanks Surveillance Program manages the aging effect of Loss of Material for buried portions of the Fuel Oil System and bottoms of above ground fuel oil tanks. There are no buried tanks within this program. The program includes an impressed current, cathodic protection system. Preventive measures to mitigate corrosion by protecting the external surface of buried piping and components are performed under a different AMP, which is the Buried Piping and Tanks Inspection Program. At the time of the License Renewal application, the Buried Piping and Tanks Surveillance Program included surveillance and monitoring of the cathodic protection system based on the guidance of NACE Standard RP-0169-76.

A review has been performed to ascertain the need to update, as necessary, administrative controls to ensure consistency with NACE Standard RP-0169-2007 (the commitment referenced a 1996 version of this standard, which has been superseded by the 2007 version) regarding acceptance criteria for the cathodic protection system. Additional leak testing provisions for underground piping have been incorporated.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.17 ABOVE GROUND CARBON STEEL TANKS PROGRAM

The Above Ground Carbon Steel Tanks Program manages the aging effects of Loss of Material for external surfaces of Fuel Oil System tanks and appurtenances. The program includes preventive measures to mitigate corrosion by protecting the external surface of carbon steel components, per standard industry practice, with protective paint or coating and with sealant or caulking at the interface with soil or concrete. Visual inspections during periodic system walk-downs are performed to monitor degradation of the protective paint, coating, caulking, or sealant. For tanks in contact with the ground, the tank sits on a layer of oily sand (no longer credited) and a cathodic protection system is provided. These measures assure that degradation is not occurring and that the component intended function will be maintained during the extended period of operation.

The administrative controls for the Program have been revised to indicate that the external surfaces of the fuel oil tanks are to be inspected periodically and to incorporate corrective action requirements.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.18 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program relies on a combination of surveillance and maintenance procedures. Monitoring and controlling fuel oil contamination in accordance with the guidelines of ASTM Standards, and other activities in accordance with the current licensing basis, maintains the fuel oil quality. Corrosion resulting from exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by periodic inspection and cleaning of tanks.

Administrative controls for the program have been enhanced to (1) improve sampling and de-watering of selected storage tanks, (2) formalize existing practices for draining and filling the Diesel Fuel Oil Storage Tank periodically, (3) formalize bacteria testing for fuel oil samples from various tanks, and (4) incorporate quarterly trending of fuel oil chemistry parameters.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.19 REACTOR VESSEL SURVEILLANCE PROGRAM

Periodic testing of metallurgical surveillance samples is used to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with 10 CFR 50, Appendix H.

The administrative controls for the program have been revised to require surveillance test samples to be stored in lieu of optional disposal.

This program is consistent with the corresponding program described in the GALL Report.

18.1.20 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program manages the aging effect of Loss of Material for buried components in RNP systems. The program includes preventive measures to mitigate corrosion by protecting the external surface of buried piping, tanks, and components by use of, for example, coating or wrapping. The program includes visual examinations of buried components when they are made accessible by excavation for maintenance or other reasons.

The program has been enhanced to (1) incorporate a requirement to ensure an appropriate as-found coating and material condition inspection is performed whenever buried piping or tanks within the scope of this program are exposed, (2) add precautions to ensure backfill with material that is free of gravel or other sharp or hard material that can damage the coating, (3) add a requirement that coating inspection shall be performed by qualified personnel to assess its condition and (4) add a requirement that a coating engineer should assist in evaluation of any coating degradation noted during the inspection.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.21 ASME SECTION XI, SUBSECTION IWE PROGRAM

The ASME Section XI, Subsection IWE Program consists of periodic visual, surface, and volumetric inspection of steel containment components for signs of degradation, assessment of damage, and corrective actions. This program is in accordance with ASME Section XI, Subsection IWE, and in accordance with 10 CFR 50.55a(g), with modifications and approved relief requests.

The administrative controls for the program have been enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.22 ASME SECTION XI, SUBSECTION IWL PROGRAM

The ASME Section XI, Subsection IWL Program consists of periodic visual inspection of concrete surfaces of reinforced and prestressed concrete containments for signs of degradation, assessment of damage, and corrective actions. This program is in accordance with the ASME Section XI, Subsection IWL, and addenda, and in accordance with 10 CFR 50.55a(g), with modifications and approved relief requests. The RNP prestressing tendons are grouted in place. Therefore, ASME Section XI Subsection IWL rules regarding unbonded post-tensioning systems are not applicable.

Enhancements have been made to administrative controls to require supervisors to notify Civil/Structural Design Engineering of the location and extent of proposed excavations of foundation concrete and to require Civil/Structural Design Engineering to examine representative sample areas of below-grade concrete when excavated for any reason.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.23 STRUCTURES MONITORING PROGRAM

The program consists of periodic inspection and monitoring of the condition of structures and structure component supports to ensure that aging degradation leading to a loss of intended function will be detected and that the extent of degradation can be determined. The inspection criteria are based on ACI 349.3R-96 and ASCE 11-90, as well as INPO Good Practice document 85-033, “Use of System Engineers,” NEI 96-03, “Guidelines for Monitoring the Condition of Structures at Nuclear Plants,” and NUMARC 93-01, “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

The Structures Monitoring Program administrative controls have been enhanced to (1) include buildings and structures and associated acceptance criteria in scope for license renewal but outside the scope of the Maintenance Rule, (2) identify interfaces between structures monitoring inspections of concrete surfaces and the Fire Protection Program requirements for barriers, (3) state clearly the boundary definition between systems and structures, (4) provide inspection criteria for portions of systems covered by structures monitoring and require corrective action(s) to be initiated for unacceptable inspection attributes, (5) expand system walk-down inspection criteria to include observation of adjacent components, (6) inspect above grade accessible concrete, (7) revise personnel responsibilities to include providing assistance in evaluating structural deficiencies when requested by the responsible engineer, inspecting excavated concrete to monitor for potential aging effects, and notifying Civil/Structural Design Engineering of the location and extent of proposed excavations, and (8) include trending requirements for structures based on aggressive ground water and lake water.

In accordance with 10 CFR 54.37(b), additional portions of the Nitrogen Supply System that were not included in the initial scope of License Renewal constitute “newly identified SSCs”. The structural elements include the liquid nitrogen storage tank foundation, high pressure bottle storage shelter, high pressure bottle racks, structural steel and concrete, and piping / tubing supports. The aging management review identified that the effects of aging for these sections of this system will be controlled through the Structures Monitoring Program. Aging effects associated with these components have been added to the administrative controls for Structures Monitoring Program.

This program is consistent with the corresponding program described in the GALL Report.

18.1.24 DAM INSPECTION PROGRAM

The Dam Inspection Program manages the following aging effects for the Lake Robinson Dam and associated concrete and steel structures: (1) Loss of Material for steel structures, (2) Loss of Form for earthen structures, and (3) Loss of Material and Change in Material Properties for concrete structures. Detection of aging effects is accomplished by an independent inspection using the FERC/U.S. Army Corps of Engineers “Recommended Guidelines for Safety Inspection of Dams.”

The system monitoring administrative controls have been revised to (1) identify the “Recommended Guidelines for Safety Inspection of Dams” as the required management program document for the dam, (2) require the responsible system engineer to review the inspection report and initiate corrective actions for any unacceptable attributes identified during the inspection process, (3) include “Recommended Guidelines for Safety Inspections of Dams” as the applicable inspection guidance in the dam inspection procedure for RNP,

(4) inspect above grade accessible concrete, (5) inspect submerged spillway concrete on a frequency not to exceed ten years, and (6) include trending requirements for structures based on aggressive ground water and lake water.

18.1.25 SYSTEMS MONITORING PROGRAM

The Systems Monitoring Program is based upon current plant activities delineated in administrative controls for performing system walkdowns.

Administrative controls for the program have been enhanced to: (1) include aging effects identified in the aging management reviews, (2) identify inspection criteria in checklist form, (3) include guidance for inspecting connected piping/components, (4) require that the extent of degradation to be recorded in the System Walkdown Report and that appropriate corrective action(s) are taken, (5) add a section specifically addressing corrective actions, and (6) ensure ‘loss of material due to wear’ is specifically included as an aging effect/mechanism identified in the system walkdown checklist.

In accordance with 10 CFR 54.37(b), additional portions of the Nitrogen Supply System that were not included in the initial scope of License Renewal constitute “newly identified SSCs”. These SSCs include the liquid nitrogen tank and high pressure storage bottle banks. The affected SSCs are constructed of carbon steel, stainless steel and aluminum, and have an external environment of outdoor air and an internal environment of purified nitrogen. Additionally, the drain piping downstream of the main steam before seat drains to the atmospheric vent header have been added to the scope of License Renewal as “newly identified SSCs”. These SSCs are carbon steel piping and stainless steel braided flexible hoses with an internal and external environment of outside air. The aging management review identified that the effects of aging will be controlled through the Systems Monitoring Program. Aging effects associated with these components have been added to the administrative controls for System Walkdown Program.

18.1.26 PREVENTIVE MAINTENANCE PROGRAM

The Preventive Maintenance (PM) Program assures that various aging effects are managed for a wide range of components. PM activities include periodic component replacement, inspections, and testing, and may be used to manage aging effects and mechanisms.

Administrative controls for the program have been enhanced to: (1) include aging effects/mechanisms identified in the aging management reviews, and (2) incorporate specific aging management activities identified in the aging management reviews into the program.

In accordance with 10 CFR 54.37(b), the Chemical and Volume Control System (CVCS) charging pump speed control air tubing constitutes a “newly identified SSC”. The aging management review identified that the effects of aging for this tubing will be controlled through the Preventive Maintenance Program, which will ensure periodic inspection and replacement as needed. The component commodity consists of copper tubing and fittings in an indoor environment with a dry instrument air internal environment. The tubing is subject to cracking due to vibration induced fatigue. As such, periodic inspection and replacement is an appropriate aging management program.

18.1.27 METAL FATIQUE OF REACTOR COOLANT PRESSURE BOUNDARY (FATIQUE MONITORING PROGRAM

The Metal Fatigue of the Reactor Coolant Pressure Boundary (Fatigue Monitoring Program) monitors the bounding primary system transient cycles to assure that transient limits are not exceeded for in scope components. These monitoring results are considered bounding for most reactor coolant pressure boundary components and various secondary side components. The acceptance criteria for the Fatigue Monitoring Program are the transient limits, which are based upon the fatigue evaluations for RCS components. By maintaining the actual transient counts below the transient limits, the fatigue usage is kept below the design code limit.

As a result of the license renewal review, the plant load/unload transient limit was reduced to provide the margin needed for consideration of reactor water environmental effects.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.28 NICKEL-ALLOY NOZZLES AND PENETRATIONS PROGRAM

The program includes (1) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (2) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (3) inservice inspection of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry-wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter 97-01. This program includes the augmented requirements in Code Case N-729-1 for the RNP upper reactor vessel head and its vessel head penetration nozzles.

The Nickel-Alloy Nozzles and Penetrations Program has incorporate the following:
(1) evaluations of indications are performed under the ASME Section XI program, (2) corrective actions for augmented inspections are performed in accordance with repair and replacement procedures equivalent to those requirements in ASME Section XI, (3) RNP maintains its involvement in industry initiatives and will systematically assess for implementation applicable programmatic enhancements, that are agreed upon between the NRC and the nuclear power industry to monitor for, detect, evaluate, and correct cracking in the vessel head penetration nozzles and other nickel-based alloy components, including base-metals and welds, during the extended period of operation, and (4) RNP submitted, for review and approval, its inspection plan for the Nickel-Alloy Nozzles and Penetrations Program, as it will be implemented from participation in industry initiatives, on July 29, 2009.

This program is consistent with the corresponding program described in the GALL Report.

18.1.29 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEAL PROGRAM

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is credited for aging management of CASS components within Class 1 boundaries of the reactor coolant system and connected systems at RNP. The aging effect/mechanism of concern is Loss of Fracture Toughness Due to Thermal Embrittlement of Cast Austenitic Stainless Steel.

Flaw tolerance evaluations for pump casings and primary loop CASS components have been done on the basis of fracture toughness methodology approved by the NRC. Consistent with NRC guidance, the RNP program does not include additional inspections of pump casings, valve bodies, or piping. RNP will continue to perform inservice inspection examinations for the primary coolant loop piping, valve and pump casings as required by Table IWB-2500-1 of Section XI of the ASME Code, unless relief has been granted by the NRC under applicable provisions in 10 CFR 50.55a from meeting the staff's ISI requirements of 10 CFR 50.55a(g)(4). Relief has been granted for several ISI requirements as documented in NRC letter dated September 26, 2002. The granted relief has been authorized for the Fourth Ten-Year Interval for RNP, which began on February 19, 2002, and is scheduled to end on February 18, 2012.

This program is consistent with the corresponding program described in the GALL Report.

18.1.30 PWR VESSEL INTERNALS PROGRAM

The PWR Vessel Internals Program includes (a) participation in industry programs and initiatives to determine appropriate inspection techniques for use in managing aging effects, and (b) monitoring and control of reactor coolant water chemistry in accordance with the Water Chemistry Program to ensure the long-term integrity and safe operation of pressurized water reactor vessel internal components. This is a new program that will incorporate the following commitments: (1) to address change in dimensions due to void swelling, RNP will continue to participate in industry programs to investigate this aging effect and determine the appropriate AMP, (2) to address baffle and former assembly issues, RNP will continue to participate in industry programs and will implement appropriate program enhancements to manage the aging effects associated with the baffle and former assembly, (3) as WOG and EPRI Materials Reliability Project (MRP) research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program. The expected results include identification of components which are the most limiting and most susceptible and identification of appropriate inspection techniques, and (4) RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements, will become part of the ASME Section XI program. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME Section XI.

RNP submitted, for review and approval, its inspection plan for the PWR Vessel Internals Program, as it will be implemented from participation in industry initiatives, on September 24, 2009, 24 months prior to the augmented inspection scheduled for 2011.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.31 ONE-TIME INSPECTION PROGRAM

Special inspections of components within the scope of license renewal have been performed in accordance with the One-Time Inspection Program. The program is used to verify the effectiveness of the aging management activities and to determine the present condition of components. One-Time Inspection Program activities consist of inspecting (1) the CCW heat exchanger tubes, (2) miscellaneous piping protected by the Water Chemistry Program, (3) small bore RCS and connected piping, (4) emergency diesel generator exhaust silencers, (5) containment liner plate and moisture barrier, (6) the diesel-driven fire pump and fuel oil tank, and (7) steam generator feed ring/J-nozzles.

In accordance with 10 CFR 54.37(b), the main steam line before seat drains have been added to the scope of License Renewal as “newly identified SSCs”. The affected SSCs are constructed of carbon steel and have an internal and external environment of outdoor air. One-time inspections, consisting of ultrasonic piping thickness examinations, determined that the condition of the drain piping remains acceptable.

This program is consistent with the corresponding program described in the GALL Report.

18.1.32 SELECTIVE LEACHING OF MATERIALS PROGRAM

The program includes mechanical testing to determine the properties of selected components that may be susceptible to selective leaching to determine whether loss of materials is occurring and whether the process will affect the ability of the components to perform their intended function for the period of extended operation. Mechanical means include resonance when struck by another object, scraping, or chipping. These techniques provide a valid method of identification and subsequent management of selective leaching.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.33 NON-EQ INSULATED CABLES AND CONNECTIONS PROGRAM

The Non-EQ Insulated Cables and Connections Program is credited for aging management of cables and connections not included in the RNP EQ Program. The non-EQ insulated cables and connections managed by this program include those used for power, instrumentation, control, and communication, including cables sensitive to reduction in insulation resistance such as radiation monitoring and nuclear instrumentation. The program involves periodic, visual inspections of accessible cables and connections installed in adverse localized environments to detect embrittlement, cracking, melting, discoloration, or swelling that could lead to reduced insulation resistance or electrical failure. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection.

This program is consistent with the corresponding program described in the GALL Report.

18.1.34 AGING MANAGEMENT PROGRAM FOR NON-EQ ELECTRICAL CABLES USED IN INSTRUMENTATION CIRCUITS

In this aging management program, calibration results or findings of surveillance testing programs are used to identify the potential existence of aging degradation. Exposure of electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument loop.

This program applies to the cables used in CV high-range radiation monitoring instrumentation circuits.

This program is consistent with the corresponding program described in the GALL Report, with exceptions as justified in docketed correspondence.

18.1.35 AGING MANAGEMENT PROGRAM FOR NEUTRON FLUX INSTRUMENTATION CIRCUITS

In this aging management program, an appropriate test, such as insulation resistance tests, time domain reflectometry (TDR) tests, or I/V testing will be used to identify the potential existence of a reduction in cable insulation resistance (IR). Exposure of electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in reduced IR. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument loop. This program applies to the cables used for the Source Range, Intermediate Range, Power Range, and Gamma-Metrics circuits of the Excore Nuclear Instrumentation System (NIS).

18.1.36 FUSE HOLDER PROGRAM

The Fuse Holder Program focuses on the metallic clamp (or clip) portion of the fuse holder. The parameters monitored include thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent manipulation of the fuse itself or vibration, chemical contamination, corrosion, and oxidation. The program utilizes thermography or other appropriate testing (to be determined prior to implementation) to identify the potential existence of aging degradation such as high contact resistance. The program applies to fuse holders located outside of active devices. Fuse holders inside an active component, such as switchgear, power supplies, inverters, battery chargers, control panels, and circuit boards, are considered to be parts of the larger assembly. Since piece parts and subcomponents in such an enclosure are inspected regularly and maintained as part of the plant's normal maintenance and surveillance activities, they are not within the scope of this program.

18.1.37 AGING MANAGEMENT PROGRAM FOR BUS DUCTS

RNP will implement an aging management program to check a sampling of bolted connections of bus ducts. The sample of accessible bolted connections will be checked for loose connections by using thermography or by measuring connection resistance using a low-range ohmmeter. Visual inspections of the bus ducts for signs of cracks, corrosion, foreign debris, excessive dust buildup, evidence of water intrusion or discoloration, which may indicate overheating, will also be performed to identify the potential existence of aging degradation. The program applies to the iso-phase bus duct, as well as all non-segregated 4.16 KV and 480 V bus ducts within the scope of license renewal. Industry experience has shown that the bus ducts exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connections related to repeated cycling of connected loads or the ambient temperature environment.

18.1.38 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIAL COMPONENTS

The existing RNP EQ Program has been established in accordance with the requirements of 10 CFR 50.49. The program will adequately manage aging of EQ equipment for the period of extended operation. Components that have been determined by EQ evaluation to have age-related limitations or restrictions are refurbished, re-qualified, or replaced prior to becoming incapable of performing their intended functions.

REFERENCES: SECTION 18.1

18.1.1 NUREG-1801, Revision 0, Generic Aging Lessons Learned (GALL) Report, July 2001

18.2 EVALUATION OF TIME LIMITED AGING ANALYSES

18.2.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

18.2.1.1 Pressurized Thermal Shock

10 CFR 50.61 requires the reference temperature (RT_{PTS}) for reactor vessel beltline materials to be less than the "PTS screening criteria" at the expiration date of the Operating License unless otherwise approved by the NRC. The screening criteria limit the amount that the material reference temperature, RT_{PTS}, may increase following neutron irradiation.

WCAP-15828, Revision 0, provides an evaluation of PTS for RNP that incorporates the results of the surveillance Capsule X evaluation. The calculated RT_{PTS} temperatures for reactor vessel beltline materials, including plates, forgings, axial welds, inlet nozzles, outlet nozzles, and nozzle welds, have been demonstrated to remain below the 270°F PTS screening criterion throughout the 60-year period of extended operation. The limiting location is Circumferential Weld Seam 10-273, which has an RT_{PTS} temperature of 297°F.

Therefore, the TLAA for Pressurized Thermal Shock has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.1.2 Upper Shelf Energy

10 CFR Part 50, Appendix G, paragraph IV.A.1, requires that reactor vessel beltline materials have a Charpy upper-shelf energy (USE) of no less than 50 ft-lbs (68 J) throughout the life of the reactor vessel unless otherwise approved by the NRC.

WCAP-15828, Revision 0, Appendix A, provides an evaluation of USE for the RNP incorporating the results of the surveillance Capsule X evaluation. WCAP-15828, Appendix A, Table A-3, provides predicted end-of-extended-license (50 EFPY) USE values for the beltline region materials. The limiting value is for Upper Shell Plate W-10201-3, which has a predicted 60-year USE of 48.4 ft-lbs. This exceeds the applicable 42 ft-lbs minimum requirement from the Equivalent Margins Analysis provided in WCAP-13587, Revision 1, for this material.

Based on the foregoing discussion, the TLAA for reactor pressure vessel USE has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.2 METAL FATIGUE

18.2.2.1 Fatigue Analysis (Design)

The reactor vessel, pressurizer, steam generators (primary side), and reactor coolant pumps have been designed to ASME Section III, Class A (now Class 1), requirements which include analyses to address fatigue and establish limits such that initiation of fatigue cracks is precluded. The fatigue analyses are contained in the stress reports for each of these components. Fatigue usage factors for critical locations in NSSS components were determined using design cycles specified during the design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements.

Additional explicit fatigue analyses have been prepared since original design to address (1) thermal stratification of the pressurizer surge line, (2) reactor vessel internals holddown spring and alignment pins, (3) insurge/outsurge flow between the pressurizer and surge line, (4) containment bellows, and (5) thermal cycling of auxiliary feedwater to main feedwater connections. These analyses also determined fatigue usage factors using design cycles specified during the design process.

In addition, the RNP Class 1 piping was designed in accordance with USAS B31.1 Power Piping Code – 1965 Edition. These rules provide an implicit fatigue design basis because cyclic loading is required to be considered when applying the code rules, but explicit fatigue analyses are not required. Most RNP piping has been designed in accordance with USAS B31.1. Auxiliary heat exchangers at RNP were designed in accordance with Westinghouse specifications and ASME Section III, Class C, or ASME Section VIII, requirements that have fatigue design rules essentially identical to B31.1.

Experience has shown actual plant operation often is very conservatively represented by the assumed design cycle count. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the fatigue analyses remain valid for the period of extended operation, the operational cycle set for plant components was assembled. The actual frequency of occurrence for the design cycles was determined and compared to the design cycle set. The severity of the actual plant transients, e.g., partial cycles, was compared to the severity of the assumed design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis, the design cycle profiles envelope actual plant operation. The result of this evaluation was the set of adjusted cumulative transient cycle counts. These data were used as a basis for 60-year projections, along with trending data from the past operational periods. Some projected cycle counts were adjusted to account for the decrease in the number of cycles experienced during recent plant operations from the high number of cycles experiences during the early years of plant operation. The resulting 60-year transient projections were then compared to the 40-year design transient limits to determine the remaining margin above the projected values.

The evaluation concluded that, with one exception, the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation, and, therefore, the TLAAs for fatigue remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i). The exception involves a fatigue evaluation performed for feedwater line connection reinforcement pads. The fatigue analysis was based in part upon the number of surveillance tests performed during refueling cycles. Since the number of refueling cycles is increased during the license renewal period, this fatigue analysis requires projection for 60 years.

The three connections downstream from the steam-driven auxiliary feedwater pump could not be qualified for the full 40-year design transient set, so a reduced number of design transients were postulated. The number of transients used in the analysis will be used as limits tracked by the Fatigue Monitoring Program (FMP). The components will be either reanalyzed or replaced prior to exceeding the transient limits tracked by the FMP.

18.2.2.2 Environmentally Assisted Fatigue

Plant-specific environmental fatigue calculations were performed for a sample of high fatigue locations to demonstrate that adequate conservatism exists within the fatigue TLAAs to account for reactor water environmental effects. These sample locations include the seven specified in NUREG/CR-6260 for older-vintage Westinghouse plants. Environmentally Assisted Fatigue (EAF) relationships developed in NUREG/CR-6583, for carbon and low alloy steels, and NUREG/CR-5704, for stainless steels, were used. Since the pressurizer surge line was not shown to have an EAF-adjusted CUF value below 1.0, an increased sample of high fatigue locations was evaluated for environmental fatigue, including seven additional locations in the pressurizer for which plant-specific fatigue analyses exist. All sample locations were shown to have an environmentally-adjusted CUF value below 1.0, indicating acceptability, except for the pressurizer surge line and the stainless steel pressurizer surge line nozzle safe end and RCS hot leg pressurizer surge line nozzle. Subsequent to license renewal approval, the fatigue analysis for the pressurizer surge line was reanalyzed and demonstrated that the environmentally-adjusted CUF value was below 1.0.

18.2.2.3 Reactor Vessel Underclad Cracking

A fracture mechanics analysis completed in 1971 concluded that fatigue growth of potential underclad flaws in reactor vessel base metal over a 40-year period would be insignificant and the structural integrity of reactor vessels had not been compromised for their intended use for a 40-year period. The underclad cracking analysis has been updated by a topical report, WCAP-15338 [Reference 18.2.1], to justify operation for 60 years. The topical report results indicated that an assumed flaw, assumed to grow under the influence of transient cycles for a period of 60 years, would remain below the most critical allowable flaw depth. Since the estimated final flaw depth is smaller than the allowable flaw depth, it was concluded that a reactor vessel with postulated underclad cracks would be acceptable for operation for 60 years. An NRC Safety

Evaluation [Reference 18.2.2] concluded that WCAP-15338 is acceptable for referencing as a topical report, and RNP has verified that the report is applicable to the RNP reactor vessel. Therefore, the TLAA for reactor vessel underclad cracking has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.2.4 Containment Penetration Bellows Fatigue

Fatigue TLAAs were identified for certain the flexible bellows assemblies used for hot piping penetrations through the containment wall. The current analysis assumes that the bellows assemblies will experience a number of transient cycles based on a 40-year life. The significant thermal transients that result in flexure of the hot pipe penetration bellows are those involving a full-range temperature change in the piping system. These are the plant heatup and cooldown cycles. As discussed above, an evaluation of operational transients shows that the number of heatup and cooldown cycles included within the 40-year design basis remain conservative for 60 years of operation. Therefore, the number of cycles assumed in the analysis of the penetration bellows exceeds the heatup cooldown cycles applicable for 60 years of operation, and the bellows fatigue calculations remain valid through the license renewal period. Thus, the TLAA for containment bellows fatigue are conservative and bounding for the period of extended operation, and, therefore, the TLAAs for fatigue remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

18.2.3 CRANE FATIQUE

Load lifting cranes within the scope of license renewal have service life limitations based upon the number of load cycles they can safely withstand.

The polar crane and spent fuel cask crane have been identified as having 40 year TLAAs for structural fatigue considerations. In support of license renewal, the cranes have been evaluated for structural fatigue considerations for a 60-year service period. The evaluations are summarized in the following paragraphs.

18.2.3.1 Polar Crane

The RNP polar crane is a low-cycle lifting device. While the plant is in operation, the polar crane is not operated. The polar crane is only operated during refueling/forced/maintenance outages and therefore the total number of lift cycles is directly dependent on the number of refueling/forced/maintenance outages. The total number of refueling outages for 60 years of operation has been established as 40. The total number of upper and mid-range lifts has been determined to be 110 per outage; thus, for a total of 40 outages, the number of lift cycles is 4400 (forced/maintenance outages requiring polar crane operation are very infrequent and considered bounded by margin). This is less than the 10,000 permissible lift cycles for this crane and is therefore acceptable. Based on the foregoing, the RNP polar crane has been evaluated for fatigue for 60 years, and the Polar Crane TLAA has been successfully projected for 60 years in accordance with 10 CFR 54.21(c)(1)(ii).

18.2.3.2 Spent Fuel Cask Crane

The number of lift cycles originally projected for the spent fuel cask crane during a 40-year period was 2,500. This can be multiplied by a factor of 1.5 to determine the number of cycles for the 60-year life. Therefore, the number of load cycles projected for 60 years is 3,750. This is less than the 20,000 permissible cycles and is therefore acceptable. Therefore, the RNP spent fuel cask crane has been evaluated for fatigue for 60 years, and the Spent Fuel Cask Crane TLAA has been successfully projected for 60 years in accordance with 10 CFR 54.21(c)(1)(ii).

18.2.4 ENVIRONMENTAL QUALIFICATION

The thermal, radiation, and wear cycle aging analyses, as applicable, of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as TLAAs. Equipment in the RNP Environmental Qualification (EQ) Program has been evaluated to determine if existing environmental qualification aging analyses can be projected to the end of the period of extended operation. Qualification that extends into the period of extended operation is addressed in the same manner as qualification for the current operating term. Should an analysis fail to justify a qualified life to 60-years, the equipment or affected component will be replaced prior to exceeding its qualified life in accordance with the existing provisions of the EQ Program.

Age-related service conditions that are applicable to environmentally qualified equipment were evaluated for the period of extended operation to verify that the current analyses remain bounding. The evaluations considered thermal, radiation, and wear cycle aging effects, as applicable.

Therefore, the analyses associated with environmental qualification of electrical equipment have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii), or the effects of aging will be adequately managed by periodic replacement of components in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

18.2.5 CONTAINMENT TENDON LOSS OF PRESTRESS

The vertical prestressing tendons are used to impart compressive forces in the prestressed concrete containment to resist the internal pressure that would be generated in the event of an accident. The prestressing forces generated by the tendons diminish over time due to losses in prestressing forces in the tendons and in the surrounding concrete. The prestressing force evaluation has been determined to remain valid to the end of the period of extended operation, and the final projected preload will remain above the minimum required preload to the end of this period. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

Therefore, the analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

To provide additional assurance of the tendons design capacity, testing (at integrated leak rate test pressure), similar to the Structural Integrity Test performed in 1992, will be scheduled to coincide with Appendix J containment integrated leak rate testing conducted during the period of extended operation (required frequency in accordance with 10 CFR 50, Appendix J). The monitoring criteria for these tests will be limited to deformations and cracking associated with the vertical prestressed tendons, and will not include radial monitoring. Guidelines for performing the IWL examinations for these tests will include additional emphasis on looking for a pattern of horizontal cracks, and additional cracking in the discontinuity areas.

18.2.6 THERMAL AGING EMBRITTLEMENT

Leak-Before-Break Analysis of Reactor Coolant System Piping, WCAP-15628, [Reference 18.2.3] is a new leak-before-break (LBB) calculation applicable to RNP large bore reactor coolant system (RCS) piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The new analysis uses the 40-year design basis thermal transients as an input for the fracture mechanics analyses. These transients have been shown to be conservative for the 60-year operating period. Therefore, the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.7 FRACTURE MECHANICS ANALYSIS FOR REACTOR COOLANT PUMP

WCAP-15363, Rev. 1, [Reference 18.2.4] is a new analysis that compares plant-specific loadings and materials to generic loadings and materials used in an earlier evaluation to support use of ASME Code Case N-481 for Westinghouse Model 93 pumps. WCAP-15363, Rev. 1, includes allowances for a reduction of fracture toughness of cast austenitic stainless steel during the 60-year operating period, but uses the limiting transients from the 40-year design transient set. This is acceptable because the 40-year design transients have been shown to be conservative for 60 years of plant operation. WCAP-15363, Rev. 1, uses plant-specific material property data instead of generic materials data and demonstrates that margin requirements for leakage and crack stability have been met. The new analysis permits the use of the surface examination of pump casings in lieu of volumetric examination, in accordance with the Code Case, throughout the period of extended operation. Therefore, the ASME Code Case N-481 analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.8 FOUNDATION PILE CORROSION

Corrosion of Class 1 structure foundation piles was identified as a TLAA based on the evaluation of the piles for a 40-year corrosion loss. The original analysis determined corrosion losses would be negligible based on measured soil resistivity values that minimize the possibility of active corrosion. Industry data from NUREG-1557, “Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal,” and EPRI TR-103842, “Class I Structures License Renewal Industry Report,” confirm that steel piles driven in undisturbed soils have been unaffected by corrosion, and those driven in disturbed soil experience minor to moderate corrosion to a small area of metal.

A reanalysis of foundation pile corrosion for license renewal determined that corrosion losses would remain non-significant for the period of extended operation and will not prevent the foundation piles from performing their license renewal intended functions. This conclusion is consistent with the recommendations and findings of NUREG-1557 and EPRI Report TR-103842, and is in accordance the estimated corrosion losses developed in the original analysis. Therefore, the foundation pile corrosion analysis results have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

18.2.9 ELIMINATION OF CONTAINMENT PENETRATION COOLERS

This TLAA has been eliminated. See RNP letter Serial: RNP-RA/03-0094, dated August 13, 2003. Also, see the License Renewal Safety Evaluation Report, Section 4.6.3, transmitted by NRC letter, Serial: RRA-04-0011, dated January 20, 2004.

18.2.10 AGING OD BORAFLEX IN SPENT FUEL POOL

This TLAA has been eliminated. See License Amendment No. 198, transmitted by NRC letter, Serial: RRA-03-0128, dated December 22, 2003. Also, see the License Renewal Safety Evaluation Report, Section 4.6.4, transmitted by NRC letter, Serial: RRA-04-0011, dated January 20, 2004.

REFERENCES SECTION 18.2

18.2.1 WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, March, 2000.

18.2.2 USNRC Safety Evaluation of WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, October 15, 2001.

18.2.3 WCAP-15628, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program, July 2001.

18.2.4 WCAP-15363, Rev. 1, A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H.B. Robinson Unit 2 for the License Renewal Program, July 2001.