18.0 FINAL SAFETY ANALYSIS REPORT SUPPLEMENT FOR LICENSE RENEWAL

18.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment and evaluation of time-limited aging analyses (TLAA) 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 and identifies those programs that have been determined to be consistent with NUREG-1801 (Reference 18.1.0-1).

Three elements common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the HNP Quality Assurance (QA) Program, which implements the requirements of 10 CFR 50, Appendix B.

In accordance with the guidance of NUREG-1801, regarding aging management of reactor vessel internals components for aging mechanisms, such as embrittlement and void swelling, HNP will: (1) participate in the industry programs for investigating and managing aging effects on reactor internals (such as Westinghouse Owner's Group and Electric Power Research Institute materials programs), (2) evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

In accordance with the guidance of NUREG-1801, regarding activities for managing the aging of nickel alloy and nickel-clad components susceptible to primary water stress corrosion cracking, HNP will comply with applicable NRC Orders and will implement: (1) applicable Bulletins and Generic letters, and (2) staff-accepted industry guidelines.

Prior to the period of extended operation, HNP will replace the subject elastomeric and thermoplastic components referenced in Requests for Additional Information 3.4-2, 3.4-3, 3.4-4, 3.4-5, and 3.4-7 from NRC letter, dated January 7, 2008 (Reference 18.1.0-2), and add them to the Preventive Maintenance Program. HNP will perform an evaluation to determine the frequency of periodic replacement of the components during the period of extended operation based on the guidance in the HNP Preventive Maintenance Program.

Prior to the period of extended operation, HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary.

Upon issuance of the renewed license, guidance will be incorporated into administrative control procedures that manage the HNP configuration control process to ensure that the requirements of 10 CFR 54.37(b) are met.

18.1.1 AGING MANAGEMENT PROGRAMS

18.1.1.1 ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program

The American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program is an existing program that consists of periodic volumetric, surface, and/or visual examinations of components for assessment, signs of degradation, and corrective actions. The HNP program is implemented in accordance with the requirements of 10 CFR 50.55a and the applicable Edition and Addenda of the ASME B&PV Code, Section XI, as required by 10 CFR 50.55a(g)(4)(ii).

18.1.1.2 Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as a process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion and cracking. The HNP Water Chemistry 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 HNP Water Chemistry Program is currently based on the latest version of the Electric Power Research Institute (EPRI) pressurized water reactor guidelines, "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1 and 2, Revision 5," (Reference 18.1.1-1) and "Pressurized Water Reactor Secondary Water Chemistry Guidelines – Revision 7" (Reference 18.1.1-2). The HNP Water Chemistry Program will be updated as revisions to the guidelines are released. The HNP Water Chemistry Program is an existing program and is consistent with the corresponding program described in NUREG-1801.

18.1.1.3 Reactor Head Closure Studs Program

The HNP Reactor Head Closure Studs Program is an existing condition monitoring program which is implemented primarily through the HNP ASME Section XI Inservice Inspection Program. In addition, the program includes certain preventive measures recommended by Regulatory Guide 1.65, "Material and Inspection for Reactor Vessel Closure Studs" (Reference 18.1.1-3). This program is credited for aging management of the Reactor Vessel Closure Head Stud Assembly (Closure Studs and Closure Nuts) for cracking due to stress corrosion cracking and loss of material due to wear.

18.1.1.4 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 includes components that may be susceptible to exposure to boric acid including mechanical, structural, and electrical components. The Boric Acid Corrosion Control Program includes plant-specific reactor coolant pressure boundary (RCPB) boric acid leakage identification and inspection procedures to ensure that leaking borated coolant does not lead to degradation of the leakage source or adjacent structures, and provides assurance that the RCPB will continue to perform its intended functions consistent with the CLB for the period of extended operation. The Boric Acid Corrosion Control Program is an existing program and is consistent with the corresponding program described in NUREG-1801.

18.1.1.5 Nickel-alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure heads of Pressurized Water Reactors Program

The HNP Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is an existing program that provides for the periodic inspection of the Reactor Pressure Vessel head and Vessel Head Penetration nozzles. This program effectively manages cracking in the Vessel Head Penetration (VHP) nozzles by identifying cracking in the upper penetration nozzles or the J-groove welds prior to loss of intended function. The required inspections are performed in the HNP ISI Program as augmented inspections.

Prior to the period of extended operation, the ISI Program administrative controls will be enhanced to specifically identify the requirements of NRC Order EA-03-009. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.6 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The HNP Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program that will manage loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement of CASS reactor vessel internals. This program will be performed as augmented inspections to visual inspections already required by the ASME Code. Components within the scope of this Program include CASS reactor vessel internals components that have been determined to be potentially susceptible to thermal aging and/or are subjected to neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV).

18.1.1.7 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion (FAC) Program is an existing program that provides for prediction, inspection, and monitoring of piping, valves, and fittings for a loss of material due to FAC so that timely and appropriate action may be taken to minimize the probability of experiencing a FAC-induced consequential leak or rupture. The FAC Program elements are based on the recommendations identified in NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program" (Reference 18.1.1-4), which require controls to assure the structural integrity of carbon steel lines containing high-energy fluids (two phase as well as single phase). The HNP FAC Program manages loss of material in carbon steel piping and fittings.

Prior to the period of extended operation, the HNP FAC Program will be enhanced to provide a consolidated exclusion bases document (i.e., a FAC susceptibility analysis). The exclusion bases document will include an evaluation of the Steam Generator Feedwater Nozzles to determine their susceptibility to FAC. Following enhancement, the Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.8 Bolting Integrity Program

The Bolting Integrity Program addresses aging management requirements for bolting on mechanical components within the scope of License Renewal. The HNP Bolting Integrity Program utilizes industry recommendations and EPRI guidance that considers material properties, joint/gasket design, chemical control, service requirements, and industry and site operating experience in specifying torque and closure requirements. The program relies on recommendations for a Bolting Integrity Program, as delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," (Reference 18.1.1-5) and industry recommendations, as delineated in EPRI reports NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," (Reference 18.1.1-6) and TR-104213, "Bolted Joint Maintenance & Applications Guide," (Reference 18.1.1-7) for pressure retaining bolting within the scope of License Renewal. Bolting and closures inspections, monitoring and trending, and repair and replacement are performed under ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program and External Surfaces Monitoring Program requirements, as applicable. Degraded conditions are also subject to the Corrective Action Program. The Structures Monitoring Program and the ASME Section XI Inservice Inspection, Subsection IWF Program are credited for aging management of structural bolting.

Prior to the period of extended operation a precautionary note will be added to plant bolting guidelines to prohibit the use of molybdenum disulfide lubricants.

18.1.1.9 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program is credited for aging management of the tubes, tube plugs, tube supports, and the secondary-side components whose failure could prevent the steam generator from fulfilling its intended safety function for the period of extended operation. The Steam Generator Tube Integrity Program is based on an existing program, the Steam Generator Integrity Program, that has been established to meet the intent of the Steam Generator Program guidance in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2 (Reference 18.1.1-8). The Steam Generator Integrity Program is based on Technical Specification requirements and NEI 97-06.

Prior to the period of extended operation the Program implementing procedure will be enhanced to include a description of the instructions for implementing corrective actions if tube plugs or secondary-side components (e.g., tube supports) are found to be degraded.

18.1.1.10 Open-Cycle Cooling Water System Program

The Open-Cycle Cooling Water System Program addresses the aging effects of material loss, flow blockage, and reduction in heat transfer due to micro- or macro-organisms and various corrosion mechanisms in raw water piping systems. This Program was originally developed in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The Program includes monitoring, inspecting, and testing to verify to verify the safety related portion of the service water systems can perform their intended functions. The Program provides assurance that aging effects for the Open-Cycle Cooling Water System can be managed during the period of extended operation. The Open-Cycle Cooling Water System

Program is an existing program that is consistent with the corresponding program described in NUREG-1801.

18.1.1.11 Closed-Cycle Cooling Water System Program

The Closed-Cycle Cooling Water System Program is an existing program that addresses aging management of components in the Component Cooling Water and Essential Services Chilled Water Systems and components in other systems cooled by these systems. This Program also manages the jacket water components associated with the Emergency Diesel Generators, Diesel Driven Fire Pump, and Security Diesel. These systems are closed cooling loops with controlled chemistry, consistent with the NUREG-1801 description of a closed cycle cooling water system. These systems employ an effective chemistry program augmented by component testing and inspection based on "Closed Cooling Water Chemistry Guideline: Revision 1 to TR-107396, Closed Cooling Water Chemistry Guideline," EPRI, Palo Alto, CA: 2004, 1007820 (Reference 18.1.1-9) to assure the License Renewal intended function(s) are maintained.

18.1.1.12 Boraflex Monitoring Program

The Boraflex Monitoring Program is implemented to assure that no unexpected degradation of the Boraflex neutron absorbing material would compromise the criticality analysis for spent fuel storage racks. The criticality analysis for Pressurized Water Reactor spent fuel racks contained in pools A and B currently reflects a zero Boraflex credit. HNP plans to perform new criticality analysis to eliminate credit for Boraflex in the Boiling Water Reactor (BWR) spent fuel racks. Until such time as this analysis has been completed and approved by the NRC, the Boraflex Monitoring Program will continue to be implemented. The Boraflex Monitoring Program is an existing program that relies on periodic inspection, testing and analysis of test coupons and monitoring of silicon levels to assure that the required 5% subcriticality margin is maintained.

Unless an approved analysis exists that eliminates credit for the Boraflex in the BWR fuel racks, prior to the period of extended operation, the Program will be enhanced to: (1) include measurements of actual boron areal density using in-situ techniques, (2) include neutron attenuation testing ("blackness testing"), to determine gap formation in Boraflex panels, and (3) include the use of the EPRI RACKLIFE predictive code or its equivalent. Following enhancement, the Boraflex Monitoring Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.13 Inspection of Overhead Heavy Load and Light Load Handling Systems Program

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program manages the aging effects of corrosion of structural components and wear of crane rails for the Containment Building Polar Crane, Jib Cranes, Reactor Cavity Manipulator Crane and the Fuel Cask Handling Crane, Fuel Handling Building Auxiliary Crane, and the Fuel Handling Bridge Crane. The Program is an existing program that provides for the periodic inspection of rails and structural members for degradation.

Administrative controls for the Program will be enhanced, prior to the period of extended operation to: (1) include in the Program all cranes that are within the scope of License Renewal; (2) require the responsible engineer to be notified of unsatisfactory crane inspection results; (3) specify an annual inspection frequency for the Fuel Cask Handling Crane, Fuel Handling Bridge Crane, and Fuel handling Building Auxiliary Crane, and every refuel cycle for the Polar Crane, Jib Cranes, and Reactor Cavity Manipulator Crane, and (4) include a requirement to inspect for bent or damaged members, loose bolts/components, broken welds, abnormal wear of rails, and corrosion (other than minor surface corrosion) of steel members and connections. The enhanced Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.14 Fire Protection Program

The HNP Fire Protection Program is an existing program that provides aging management of the diesel-driven fire pump fuel oil supply line and credited fire barrier assemblies including fire doors, penetration seals, fire wrap, barrier walls, barrier ceilings and floors, and seismic joint filler. The HNP Fire Protection Program is an effective program that will adequately manage cracking and loss of material so that system intended functions will be maintained consistent with the CLB for the period of extended operation.

Prior to the period of extended operation, the program will be enhanced to: (1) include inspection criteria as described in NUREG-1801 for penetration seals, (2) provide specific procedural guidance for inspecting fire barrier walls, ceilings and floors, (3) include a visual inspection of the diesel-driven fire pump fuel oil supply piping for signs of leakage, and (4) include minimum qualification requirements for inspectors performing inspections required by this Program. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.15 Fire Water System Program

The Fire Water System Program is an existing program that includes system pressure monitoring, wall thickness evaluations, periodic flow and pressure testing in accordance with applicable NFPA commitments, and periodic visual inspection of overall system condition. These activities provide an effective means to determine whether corrosion and biofouling are occurring. Inspections of the sprinkler heads assure that corrosion products that could block flow are not accumulating. These measures will allow timely corrective action in the event of system degradation to ensure the capability of the water-based Fire Suppression System to perform its intended functions.

Prior to the period of extended operation, the Program will be revised to: (1) incorporate a requirement to perform one or a combination of the following two activities: (a) Perform non-intrusive baseline pipe thickness measurements at various locations, prior to the expiration of current license and trended through the period of extended operation. The plant-specific inspection intervals will be determined by engineering evaluation performed after each inspection of the fire protection piping to detect degradation prior to the loss of intended function, or (b) Perform flow testing meeting the general flow requirements (intent) of NFPA 25, (2) either replace the sprinkler heads prior to reaching their 50-year service life or revise site procedures to perform field service testing, by a recognized testing laboratory, of representative samples from one or more sample areas, and (3) for in-scope spray nozzles, either (a) add a requirement to perform flow testing to ensure proper spray pattern or (b) add a modification to prevent blockage from external sources. Following enhancement, the Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.16 Fuel Oil Chemistry Program

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality by monitoring and controlling fuel oil contamination in accordance with the guidelines of the American Society for Testing Materials (ASTM) Standards specified in the HNP Technical Specifications Surveillance Requirements and chemistry program procedures for fuel oil testing. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by verifying the quality of new oil and the addition of a stabilizer, which contains a biocide and corrosion inhibitors, before the fuel oil is added to the storage tanks that supply the Emergency Diesel Generator and Security Power Diesel Generator. Continued quality levels are assured by periodically checking for and removing water from tank drains, sampling to confirm that the bulk properties of water and sediment, particulate contamination, and biological growth are within administrative target values or Technical Specification limits. Confirmatory samples are periodically taken from the day tanks to assess the quality of fuel in the components downstream of the storage tanks by testing for bulk properties of water and sediment and particulate contamination. The day tank sampling frequency is adjusted based on site operating experience. The effectiveness of the program is verified using visual inspections of system tanks to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

Prior to the period of extended operation the program administrative controls will be enhanced to: (1) add requirements to enter an item into the corrective action program whenever an administrative value or control limit for parameters relevant to this program are exceeded or water is drained from a fuel oil tank in the scope of this program; (2) establish administrative values for fuel oil chemistry parameters relating to corrosion; (3) require Diesel Fuel Oil System chemistry controls to include semiannual monitoring and trending of water and sediment and particulates from an appropriate sample point for the day tanks and semiannual monitoring and trending of biological growth in the main storage tanks; (4) require Security Power System fuel oil chemistry controls to include semiannual monitoring and trending of biological growth in the fuel oil in the buried storage tank and periodic inspecting of the internal surfaces of the buried storage tank and the aboveground day tank or require UT or other NDE of the tanks if inspection proves inadequate or indeterminate; (5) require Site Fire Protection System fuel oil chemistry controls for the Diesel Driven Fire Pump fuel oil storage tank to include quarterly monitoring and trending of particulates and semiannual monitoring and trending of biological growth, to check and remove water quarterly, to periodically inspect the tank or require UT or other NDE of the tank if inspection proves inadequate or indeterminate; and to revise chemistry sampling procedures to address positive results for biological growth including as one option the use of biocides; and (6) verify the condition of the Diesel Fuel Oil Storage Tank Building Tank Liners by means of bottom thickness measurements under the One Time Inspection Program. Day tank sampling for water, sediment, and particulate contamination is considered to be confirmatory of components outside the main storage tanks, and its frequency may be adjusted based on site operating experience.

18.1.1.17 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program is an existing program that manages the reduction of fracture toughness of the reactor vessel beltline materials due to neutron embrittlement. The Program fulfills the intent and scope of 10 CFR 50, Appendix H. The Program evaluates neutron embrittlement by projecting upper shelf energy for all reactor materials with projected neutron exposure greater than 10^{17} n/cm^2 (E > 1.0 MeV) after 60 years of operation and by the development of pressure-temperature limit curves. Embrittlement information is obtained in accordance with NRC Regulatory Guide 1.99, Rev. 2, chemistry tables and through the use of surveillance capsules. The surveillance program design, capsule withdrawal schedule, and evaluation of test results are in accordance with ASTM E 185-82.

Prior to the period of extended operation, the program will be enhanced to: (1) include a provision that tested and untested specimens from all capsules pulled from the reactor vessel must be kept in storage to permit future reconstitution use, and that the identity, traceability, and recovery of the capsule specimens shall be maintained throughout testing and storage, (2) include a provision that withdrawal of the next capsule (i.e., Capsule Y or Z) will occur during Refueling Outage 21, at which time the capsule fluence is projected to be 9.39 x 10^{19} n/cm^2 (E > 1.0 MeV), which is between one and two times the EOL estimated vessel fluence for the 60-year license period. The remaining Capsule and archived test specimens available for reconstitution will be available for the monitoring of neutron exposure if additional license renewals are sought, and (3) include a provision that, if future plant operations exceed the limitations in Section 1.3 of Regulatory Guide 1.99, Revision 2, or the applicable bounds, e.g., cold leg operating temperature and neutron fluence, as applied to the surveillance capsules, the impact of these plant operation changes on the extent of reactor vessel embrittlement will be evaluated, and the NRC will be notified. Following enhancement, the Program will be consistent with the corresponding program described in NUREG-1801. Additional requirements regarding surveillance capsule management are provided in Section 2.K of Renewed License No. NPF-63, dated October 24, 2008 (NRC Accession Number ML083120237).

18.1.1.18 One-Time Inspection Program

The One-Time Inspection Program uses one-time inspections to verify the effectiveness of other aging management programs or to confirm the slow progression or the absence of an aging effect. The program scope includes Water Chemistry Program, Fuel Oil Chemistry Program and Lubrication Analysis Program verifications specified by NUREG-1801, as well as plant specific inspections. The One-Time Inspection Program will be completed by the addition of procedural controls for implementation and tracking. The One-Time Inspection Program is a new program that is consistent with the corresponding program described in NUREG-1801.

18.1.1.19 Selective Leaching of Materials Program

The Selective Leaching of Materials Program includes one-time inspections and qualitative determinations of selected components that may be susceptible to selective leaching. A sample population of susceptible components will be selected for the inspections prior to commencing the period of extended operation. The inspections will determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function(s) for the period of extended operation. This new Program includes an exception to the corresponding program described in NUREG-1801 involving the use of qualitative determinations, other than Brinell hardness testing, to identify the presence of selective leaching.

18.1.1.20 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program manages the aging effect of loss of material for the external surfaces of buried piping components in HNP systems within the scope of License Renewal. There are no buried tanks in this program. The program includes preventive measures to mitigate corrosion by protecting the external surface of buried piping components through use of coating or wrapping. The program includes visual examination of buried piping components made accessible by excavation. Program requirements based on NUREG-1801 guidance include: (1) incorporate a requirement to ensure an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this Program is exposed, with a minimum frequency of at least one buried piping inspection each 10 years, (2) verify that there is at least one opportunistic or focused inspection performed within the 10-year period prior to the period of extended operation, (3) specify that an inspection checklist will be developed, (4) require inspection results to be documented, (5) add precautions concerning excavation and use of backfill to the excavation procedure with precautions for License Renewal piping, (6) add a requirement that buried pipe coating inspection shall be performed, when excavated, by qualified personnel to assess its condition, and (7) add a requirement that a coating engineer or other qualified individual should assist in evaluation of any pipe coating damage and/or degradation found during the inspection. This new Program is consistent with the corresponding program described in NUREG-1801.

18.1.1.21 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program

The HNP One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a new program that will manage cracking in small-bore (less than NPS 4) Class 1 piping through the use of a combination of volumetric examinations and visual inspections. The program will manage the aging effect through the identification and evaluation of cracking in small-bore Class 1 piping with the exception that volumetric examinations for small-bore socket-welds will not be done. In lieu of performing volumetric inspections of socket welds, the program will include one-time volumetric examinations of a sample of Class 1 butt welds for pipe less than NPS 4. Any cracking identified in small-bore Class 1 piping resulting from stress corrosion or thermal and mechanical loading will result in periodic inspections. The HNP One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be implemented and inspections completed and evaluated prior to the period of extended operation. The Program will effectively manage the aging effect by identifying and evaluating cracking in small-bore Class 1 piping prior to loss of intended function.

18.1.1.22 External Surfaces Monitoring Program

The External Surfaces Monitoring Program is an existing program based on system inspections and walkdowns. The Program consists of periodic visual inspections of components such as piping, piping components, ducting, and other equipment within the scope of License Renewal and subject to aging management review in order to manage aging effects.

Prior to the period of extended operation, the program will be enhanced to: (1) include a specific list of systems managed by the program for License Renewal, (2) provide specific guidance for insulated/jacketed pipe and piping components to identify signs of leakage and provide criteria for determining whether the insulation/jacket should be removed to inspect for corrosion, (3) provide inspection criteria for components not readily accessible during plant operations or refueling outages, (4) provide specific guidance for visual inspections of elastomers for cracking, chafing, or changes in material properties due to wear, and (5) incorporate a checklist for evaluating inspection findings, with qualified dispositions. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.23 Flux Thimble Tube Inspection Program

The Flux Thimble Tube Inspection Program is an existing program that provides for eddy current testing of flux thimble tubes, evaluation of test results, wear prediction, and determination of acceptable inspection frequencies. The program manages loss of flux thimble wall thickness so that timely and appropriate action may be taken so that a through wall leak or rupture of a flux thimble tube does not occur. The Flux Thimble Tube Inspection Program elements are based on the recommendations identified in NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," (Reference 18.1.1-10) and the methodology for predicting wear as identified in WCAP-12866, "Bottom Mounted Instrumentation Flux Thimble Wear" (Reference 18.1.1-11).

Prior to the period of extended operation, the HNP Flux Thimble Tube Inspection Program will be enhanced: (1) to require an evaluation of historic plant-specific test data in order to ensure that conservative wear rates are used so that a loss of intended function will not occur, (2) to provide guidance for treatment of flux thimbles that could not be inspected due to restriction, defect or other reason, and (3) to require test results and evaluations be formally documented as QA records. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.24 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program relies upon work order tasks that provide opportunities for the visual inspection of internal surfaces of piping, piping elements, ducting, and components. The Internal Surfaces in Miscellaneous Piping and Ducting Components Program work task activities will monitor parameters that may be detected by visual inspection and include change in material properties, cracking, flow blockage, loss of material and reduction of heat transfer effectiveness. The extent and schedule of inspections and testing assure detection of component degradation prior to loss of intended functions.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program and is consistent with the corresponding program described in NUREG-1801.

18.1.1.25 Lubricating Oil Analysis Program

The Lubricating Oil Analysis Program is an existing program, which maintains lubricating oil quality by periodic sampling for contamination per site program procedures. Exposure to contaminants, such as water and particulates, is minimized by monitoring the lube oil quality and periodic changing of the oil at fixed schedules or when monitored parameters trend toward unacceptable or administrative limits. Lubricating oil analysis of old oil is performed prior to disposal to confirm that water or particulates levels do not indicate a loss of material or reduction in heat transfer is occurring in the system. Effectiveness of the Program is verified under the One-Time Inspection Program prior to the period of extended operation.

Prior to the period of extended operation the program will be enhanced as follows: (1) a review and revision of work documents and analysis requirements will be performed to ensure that the used oil from appropriate component types in the scope of License Renewal is analyzed to determine particle count and moisture, and if oil is not changed in accordance with the manufacturer's recommendation, then additional analyses for viscosity, neutralization number, and flash point will be performed (this activity will ensure that used oil is visually checked for water); and (2) the program administrative controls will be enhanced to include a requirement to perform ferrography or elemental analysis to identify wear particles or products of corrosion when particle count exceeds an established level or when considered appropriate. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.26 ASME Section XI, Subsection IWE Program

The ASME Section XI, Subsection IWE Program is an existing aging management program used for the aging management of accessible and inaccessible pressure retaining Containment Structure Class MC components. The HNP program is implemented in accordance with the requirements of 10 CFR 50.55a and the applicable Edition and Addenda of the ASME B&PV Code, Section XI, as required by 10 CFR 50.55a(g)(4)(ii).

Prior to the period of extended operation, the ASME Section XI, Subsection IWE Program implementing procedure will be enhanced to: (1) include additional recordable conditions, (2) include moisture barrier and applicable aging effects, (3) include pressure retaining bolting and aging effects, and (4) include a discussion of augmented examinations.

18.1.1.27 ASME Section XI, Subsection IWL Program

The ASME Section XI, Subsection IWL Program is an existing aging management program used for the aging management of accessible and inaccessible pressure retaining Primary Containment concrete. The HNP containment structure does not use prestressing tendons. Therefore, ASME Section XI, Subsection IWL rules regarding post-tensioning systems are not applicable. The HNP program is implemented in accordance with the requirements of 10 CFR 50.55a and the applicable Edition and Addenda of the ASME B&PV Code, Section XI, as required by 10 CFR 50.55a(g)(4)(ii).

18.1.1.28 ASME Section XI, Subsection IWF Program

The ASME Section XI, Subsection IWF Program consists of periodic visual examination of component supports for loss of material and loss of mechanical function. The Program is an existing program implemented through plant procedures, which provide for visual examination of ISI Class 1, 2, and 3 supports. The HNP program is implemented in accordance with the requirements of 10 CFR 50.55a and the applicable Edition and Addenda of the ASME B&PV Code, Section XI, as required by 10 CFR 50.55a(g)(4)(ii).

18.1.1.29 10 CFR 50, Appendix J Program

The 10 CFR Part 50, Appendix J Program is an existing program that consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings, and access openings to detect degradation of the pressure boundary. An evaluation is performed and appropriate corrective actions are taken if leakage rates exceed acceptance criteria. For the Integrated Leak Rate Testing, this Program is implemented in accordance with Option B (performance based leak testing) of 10 CFR Part 50, Appendix J; Regulatory Guide 1.163 (Reference 18.1.1-13); and NEI 94-01, "Industry Guideline for Implementing Performance Based Option of

10 CFR Part 50, Appendix J" (Reference 18.1.1-14). For Local Leak Rate Testing, the Program is in accordance with the prescriptive requirements of 10 CFR Part 50, Appendix J Option A.

Prior to the period of extended operation, the program will be enhanced to describe in the implementing procedures the evaluation and corrective actions to be taken when leakage rates do not meet their specified acceptance criteria. Following enhancement, the Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.30 Masonry Wall Program

The HNP Masonry Wall Program is an existing program designed to manage aging effects of masonry walls. For License Renewal, the Program will assure that the evaluation basis established for each masonry wall within the scope of License Renewal remains valid through the period of extended operation. The program includes masonry walls identified as performing License Renewal intended functions within the Containment Building, Reactor Auxiliary Building, Diesel Generator Building, Fuel Handling Building, HVAC Equipment Room, Security Building, Tank Area/Building (including Unit 1 and Unit 2 buildings), Turbine Building and the Waste Processing Building. The Program is a condition monitoring program with the inspection frequencies established such that no loss of intended function would occur between inspections.

Prior to the period of extended operation, Program administrative controls will be enhanced to identify the structures that have masonry walls in the scope of License Renewal. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.31 Structures Monitoring Program

The Structures Monitoring Program consists of periodic inspection and monitoring of the condition of structures and structure component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This Program an existing program that is implemented in accordance with the Maintenance Rule, 10 CFR 50.65; NEI 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 18.1.1-15) and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 18.1.1-16). The inspection criteria are based on American Concrete Institute Standard ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures;" (Reference 18.1.1-17) and American Society of Civil Engineers, ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings;" (Reference 18.1.1-18) as well as, Institute for Nuclear Power Operations (INPO) Good Practice document 85-033, "Use of System Engineers;" (Reference 18.1.1-19) and NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Plants" (Reference 18.1.1-20).

Prior to the period of extended operation, the Structures Monitoring Program implementing procedures will be enhanced to: (1) identify the License Renewal structures and systems that credit the program for aging management, (2) require notification of the responsible engineer when below-grade concrete is exposed so an inspection may be performed prior to backfilling, (3) require periodic groundwater chemistry monitoring including consideration for potential seasonal variations., (4) define the term “structures of a system” in the system walkdown procedure and specify the condition monitoring parameters that apply to "structures of a system," (5) include the corporate structures monitoring procedure as a reference in the plant implementing procedures and specify that forms from the corporate procedure be used for inspections, (6) identify additional civil/structural commodities and associated inspection attributes required for License Renewal, and (7) require inspection of inaccessible surfaces of reinforced concrete pipe when exposed by removal of backfill. Following enhancement, the Structures Monitoring Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.32 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program

The Regulatory Guide (RG) 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program is an existing program that consists of an inspection and surveillance program used to manage the aging effects of the dams and spillways, dikes, canals, reservoirs, and the intake, screening and discharge structures associated with emergency cooling water system. The Program meets the requirements of RG 1.127, Revision 1.

Prior to the period of extended operation, the Program will be enhanced to: (1) require an evaluation of any concrete deficiencies in accordance with the acceptance criteria provided in the corporate inspection procedure, (2) require initiation of a Nuclear Condition Report (NCR) for degraded plant conditions and require, as a minimum, the initiation of an NCR for any condition that constitutes an "unacceptable" condition based on the acceptance criteria specified, and (3) require documentation of a visual inspection of the miscellaneous steel at the Main Dam and Spillway. Following enhancement, the program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program credited for the aging management of cables and connections not included in the HNP Environmental Qualification (EQ) Program. Under this Program, accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination, which are precursor indications of conductor insulation aging degradation from heat, radiation or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection. This Program is consistent with the corresponding program described in NUREG-1801.

18.1.1.34 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new program credited for the aging management of radiation monitoring and nuclear instrumentation cables not included in the HNP EQ Program. Exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced insulation resistance (IR). A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies. For radiation monitoring circuits, the review of calibration results or findings of surveillance testing will be used to identify the potential existence of cable system aging degradation. This review will be performed at least once every 10 years, with the first review to be completed before the end of the current license term. Cable systems used in Excore Nuclear Instrumentation Systems will be tested at a frequency not to exceed 10 years based on engineering evaluation, with the first testing to be completed before the end of the current license term. The scope includes source, intermediate, and power range nuclear instrumentation circuits, and the RG 1.97 wide range neutron flux monitoring circuits. Testing may include IR tests, time domain reflectometry tests, current versus voltage testing, or other testing judged to be effective in determining cable system insulation condition. This Program is consistent with the corresponding program described in NUREG-1801.

18.1.1.35 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program credited for the aging management of cables not included in the HNP EQ Program. In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than 25% of the time. Manholes associated with inaccessible non-EQ medium-voltage cables will be inspected for water accumulation and drained, as needed. The manhole inspection frequency will be based on actual field data and shall not exceed two years. The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is consistent with the corresponding program described in NUREG-1801.

18.1.1.36 Metal Enclosed Bus Program

The Metal Enclosed Bus Program is a new program credited for the aging management of the iso-phase bus as well as non-segregated 6.9KV and 480V metal enclosed bus within the scope of License Renewal. The program involves various activities conducted at least once every 10 years to identify the potential existence of aging degradation. In this aging management program, a sample of accessible bolted connections will be checked for loose connection by using thermography or by measuring connection resistance using a low range ohmmeter. In addition, the internal portions of the bus enclosure will be visually inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports will be visually inspected for structural integrity and signs of cracks. The Metal Enclosed Bus Program is consistent with the corresponding program described in NUREG-1801.

18.1.1.37 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program credited for the aging management of cable connections not included in the HNP EQ Program. The program will be implemented as a one-time inspection on a representative sample of non-EQ cables connections within the scope of License Renewal prior to the period of extended operation to provide an indication of the integrity of the cable connections. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting loose connections, such as thermography, contact resistance testing, bridge balance testing, or other appropriate testing judged to be effective in determining cable connection integrity. The factors considered for sample selection are application (high, medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.) in both indoor and outdoor environments. The technical basis for the sample selections of cable connections to be tested will be provided. In addition, the program will include the bolted connections on the overhead transmissions conductors from the high voltage bushings on the main power transformers to the switchyard bus.

18.1.1.38 Reactor Coolant Pressure Boundary Fatigue Monitoring Program

The Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program is an existing program that includes preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the RCPB. This is accomplished by monitoring and tracking the significant thermal and pressure transients for limiting RCPB components in order to prevent the fatigue design limit from being exceeded. The Program addresses the effects of the reactor coolant environment on component fatigue life by including, within the program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260, "Application of NUREG/CR-5999, Interim Fatigue Curves to Selected Nuclear Power Plant Components."

Prior to the period of extended operation, the Program will be enhanced to: (1) expand the program scope to include an evaluation of selected RCPB components beyond the reactor pressure vessel (including auxiliary system components such as the pressurizer lower head, pressurizer surge line, and CVCS piping and heat exchanger), and to include the NUREG/CR-6260 locations analyzed for environmental effects, (2) provide preventive actions to include, prior to a monitored location exceeding a cumulative usage factor limit of 1.0, evaluation of operational changes to reduce the number or severity of future transients, (3) include a provision to utilize online fatigue analysis software for the periodic updating (not to exceed once every 18 months) of cumulative usage, (4) describe the acceptance criteria for maintaining fatigue usage below the design limit, and (5) address corrective actions, to be implemented through the Corrective Action Program, for components that have exceeded alarm limits, with options to include a revised fatigue analysis or repair or replacement of the component and for piping systems that have exceeded their cyclic alarm limit to require a review of the pertinent design calculations to determine if any additional locations should be designated as postulated high energy line breaks. Following enhancement, the RCPB Fatigue Monitoring Program will be consistent with the corresponding program described in NUREG-1801.

18.1.1.39 Environmental Qualification (EQ) Program

The existing HNP EQ Program, which implements the requirements of 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. 10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). Time-Limited Aging Analysis (TLAA) demonstration option 10 CFR §54.21(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, has been chosen. The EQ Program will manage the aging effects of the components associated with the environmental qualification TLAA.

18.1.1.40 Deleted by Amendment 60

18.1.1.41 Compressed Air Monitoring Program

The Compressed Air Monitoring Program is an existing program that ensures that instrument air supplied to components is maintained free of water and significant contaminants, thereby preserving an environment that is not conducive to loss of material. Dew point and particulate contamination are periodically checked to verify the instrument air quality is maintained. Periodic and opportunistic inspections of accessible internal surfaces are performed for signs of corrosion and abnormal corrosion products that might indicate loss of material within the system.

REFERENCES: SECTION 18.1

18.1.0-1 NUREG-1801, "Generic Aging Lessons Learned (GALL)," Rev. 1, U.S. NRC, September 2005.

18.1.0-2 Letter from Maurice Heath (NRC) to Robert J. Duncan II, "Requests for Additional Information for the Review of the Shearon Harris Nuclear Power Plant, Unit 1, License Renewal Application," dated January 7, 2008.

18.1.1-1 "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1 and 2, Revision 5," EPRI, Palo Alto, CA: 2003, 1002884.

18.1.1-2 "Pressurized Water Reactor Secondary Water Chemistry Guidelines – Revision 6," EPRI, Palo Alto, CA: 2004, 1008224.

18.1.1-3 NRC Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," October 1973.

18.1.1-4 NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," April 1999.

18.1.1-5 NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," June 1990.

18.1.1-6 EPRI report NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," April 1988.

18.1.1-7 EPRI report TR-104213, "Bolted Joint Maintenance & Applications Guide," December 1995.

18.1.1-8 Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2.

18.1.1-9 "Closed Cooling Water Chemistry Guideline: Revision 1 to TR-107396, Closed Cooling Water Chemistry Guideline," EPRI, Palo Alto, CA: 2004, 1007820.

18.1.1-10 NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 1988.

18.1.1-11 WCAP-12866, "Bottom Mounted Instrumentation Flux Thimble Wear," January 1991.

18.1.1-12 NUREG-1916, "Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1," August 2008.

18.1.1-13 NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

18.1.1-14 NEI 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," Nuclear Energy Institute, July 1995.

18.1.1-15 NEI (NUMARC) 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2, April 1996.

18.1.1-16 Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2, March 1997.

18.1.1-17 American Concrete Institute Standard ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures."

18.1.1-18 American Society of Civil Engineers, ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings."

18.1.1-19 Institute for Nuclear Power Operations (INPO) Good Practice document 85-033, "Use of System Engineers."

18.1.1-20 NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Plants." December 1996.

18.1.1-21 HNP-P/LR-0009, License Renewal Benchmark Update, Revision 1.

18.2 EVALUATION OF TIME LIMITED AGING ANALYSES

18.2.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

18.2.1.1 Neutron Fluence

Calculations have been performed to determine neutron fluence projections applicable to the reactor vessel and internals at 55 EFPY, which bounds 60 years of operation, using an NRC-approved methodology applicable to HNP. The methodology used is documented in the AREVA report, BAW-2241, "Fluence and Uncertainty Methodologies," Revision 1 (Reference 18.2.1-1). Using the AREVA methodology, the data from Capsule X, which was removed from the reactor vessel at the end of cycle 8, and a value of 55 EFPY, projected values of neutron flux were obtained for use in the fluence-related analyses. In addition, the reactor pressure vessel boundary components outside the beltline region have been evaluated to determine whether additional materials should be considered "beltline" material for the period of extended operation. The beltline, as defined by 10 CFR 50.61(a)(3), is the region of the reactor pressure vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor pressure vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material with regard to radiation damage. The threshold fluence for beltline material is 1 x 10^{17} n/cm^2, E > 1.0 MeV.

The materials outside of the traditional beltline region which are expected to receive fluence values greater than 10^{17} n/cm^2 were evaluated. The evaluation found five reactor vessel locations outside the traditional beltline region with 55 EFPY fluence values greater than 10^{17} n/cm^2 that were not previously analyzed for irradiation damage. The locations were:

1. Upper to Intermediate Circumferential Weld AC,
2. Upper Shell,
3. Inlet Nozzle Weld 15-A, 15-B, 15-C,
4. Inlet Nozzle, and
5. Upper Shell Longitudinal Welds BE/BF.

The materials at these locations were analyzed for Upper Shelf Energy (USE), Pressurized Thermal Shock (PTS), and Adjusted Reference Temperature (ART) and were found not to be limiting. The controlling material for these analyses remains the intermediate shell plate, heat number B4197-2, which is a location inside the traditional beltline region.

Therefore, the neutron fluence has been projected to the end of the period of extended operation using a methodology previously approved by the NRC. The 55 EFPY fluence projections will be used for evaluating the following fluence-based analyses.

18.2.1.2 Upper Shelf Energy Evaluation

10 CFR 50, Appendix G, contains screening criteria that limit the degree that the upper shelf energy (USE) value for a reactor pressure vessel material may be allowed to drop due to neutron radiation exposure. The regulation requires the initial USE for a reactor pressure vessel material to be greater than 75 ft.-lb. when the material is in the unirradiated condition, and for the USE to remain above 50 ft.-lb. in the fully irradiated condition throughout the licensed life of the vessel, unless it is demonstrated that lower values of energy will provide margins of safety against fracture equivalent to those required by the ASME Code, Section XI, Appendix G.

An evaluation of the HNP RPV for the License Renewal period (55 EFPY) USE for the HNP reactor vessel beltline materials was performed using the guidelines in RG 1.99, Revision 2. The evaluations for the decreases in USE of the HNP reactor vessel were performed at the 1/4T wall location of each beltline material using the respective copper contents and Figure 2 of RG 1.99, Revision 2. The HNP RV beltline material with the lowest predicted USE is the intermediate shell plate, heat number B4197-2; however, the predicted value for this material is not projected to fall below the required 50 ft-lb limit. Therefore, the analyses for the decreases in USE of the HNP RV have been projected to the end of the 60-year period of extended operation. The analyses demonstrate that, for the most limiting material, the lowest predicted USE is greater than the 10 CFR 50, Appendix G, limit of 50 ft-lbs.

Therefore, the HNP USE analysis has been projected to the end of the License Renewal period of extended operation.

18.2.1.3 Pressurized Thermal Shock Analysis

10 CFR 50.61 defines screening criteria for embrittlement of reactor pressure vessel materials in pressurized-water reactors, as well as actions that are required if these screening criteria are exceeded. The screening criteria limit the degree that a vessel material may increase in its reference temperature for pressurized thermal shock - RT_{PTS}, following neutron irradiation of the reactor pressure vessel. For circumferential welds, the pressurized thermal shock (PTS) screening criterion is 300°F maximum. For plates, forgings, and axial weld materials, the screening criterion is 270°F maximum. The projected EOL RT_{PTS} values must be shown to remain below the applicable screening temperature.

A PTS evaluation for the HNP RV beltline materials was performed in accordance with 10 CFR 50.61. The PTS reference temperature, RT_{PTS}, values are calculated by adding the initial nil-ductility reference temperature, RT_{NDT}, to the predicted radiation-induced ΔRT_{NDT} and the margin term to cover the uncertainties in the values of initial RT_{NDT} copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective RV beltline materials copper and nickel contents and the neutron fluence applicable to the HNP RV for License Renewal at 55 EFPY.

The evaluations for the HNP RT_{PTS} values were performed for each HNP reactor vessel beltline material with chemistry factors determined from Tables 1 and 2 in 10 CFR 50.61. In addition, the chemistry factors for the intermediate shell plate, heat no. B4197-2, and the intermediate shell to lower shell circumferential weld are recalculated using the available HNP surveillance data.

The RT_{PTS} values for the HNP RV beltline materials at 55 EFPY were determined. The results of the PTS evaluation demonstrate that the HNP RV beltline materials will not exceed the PTS screening criteria before the end of the period of extended operation. The controlling beltline material for the HNP reactor vessel with respect to PTS is the intermediate shell plate, heat number B4197-2, with a RT_{PTS} value of 199.9°F which is well below the PTS screening criterion of 270°F.

Therefore, the analyses for the shift in PTS reference temperature of the HNP reactor vessel have been projected to the end of the period of extended operation.

18.2.1.4 Operating Pressure-Temperature (P-T) Limits Analysis

The Adjusted Reference Temperature (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 temperature at which a reactor vessel is allowed to be pressurized increases over the licensed period. The ART of the limiting beltline material is used to correct the beltline Pressure-Temperature (P-T) limits to account for radiation effects. 10 CFR Part 50 Appendix G requires reactor vessel thermal limit analyses to determine operating P-T limits for boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, 2) non-nuclear heat-up/cooldown and low level physics tests, and 3) core critical operation.

The ART values for the HNP reactor vessel beltline region materials are calculated in accordance with Regulatory Guide 1.99, Revision 2, by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT}, and a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation-induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 55 EFPY. The evaluations for the HNP ART were performed at the 1/4T and 3/4T wall location of each beltline material with chemistry factors determined from Tables 1 and 2 in Regulatory Guide 1.99, Revision 2. In addition, the chemistry factors for the intermediate shell plate, heat no. B4197-2, and the intermediate shell to lower shell circumferential weld are recalculated using the available HNP surveillance data.

In this manner, ART results for the HNP reactor vessel beltline region materials applicable to 55 EFPY were determined. Based on these results, the controlling beltline material for the HNP reactor vessel is the intermediate shell plate, heat no. B4197-2. The P-T limit curves show that adequate operating margins will be provided. Therefore, the P-T limits analysis for HNP has been projected to the end of the period of extended operation. HNP has implemented changes in the P-T curves throughout the current period of operation using the license amendment process, and expects to continue to use to license amendment process to implement future changes in P-T curves for the remainder of the current period of operation and for the extended period of operation.

18.2.1.5 Low-Temperature Overpressure Limits Analysis

ASME Section XI, Appendix G, establishes procedures and limits for RCS pressure and temperature primarily for low temperature conditions to provide protection against non-ductile failure of the reactor vessel. The Low Temperature Overpressure Protection System (LTOPS) assures that these limits are not exceeded when it is enabled at low temperatures.

No analysis of LTOP setpoints has been performed to support operation to the end of the period of extended operation for License Renewal. The LTOP setpoint analysis will be recalculated following the removal of one of the remaining surveillance capsules from the vessel. The surveillance capsule will be removed when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall comparable to an 80-year fluence. HNP has implemented changes in the LTOP setpoints throughout the current period of operation using the license amendment process, and expects to continue to use to license amendment process to implement future changes in LTOP setpoints for the remainder of the current period of operation and for the extended period of operation.

18.2.2 METAL FATIGUE

The HNP approach is to identify the latest design fatigue analyses associated with each NSSS component within the RCPB in order to demonstrate that the design analyses will remain bounding through the period of extended operation.

The first step in the evaluation was to establish the current design basis for the major NSSS components. This was done by reviewing the current design transients and past operational transients. This review showed that the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections 18.2.2.6 and 18.2.2.7). Therefore the governing transients "NSSS Design Transients" are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design Cumulative Usage Factor (CUF) values were compiled from design documents including the steam generator replacement/uprating analysis.

The next step in the HNP approach was to factor the effects of the reactor water environment on fatigue. The following subsections provide a summary of the evaluation results for each of the major NSSS components evaluated.

18.2.2.1 Reactor Vessel Fatigue Analyses

TLAAs have been identified for several sub-components of the Reactor Vessel. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections 18.2.2.6 and 18.2.2.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis. The reactor vessel fatigue analysis demonstrated that if the reactor vessel components were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0. For the components that comprise the Reactor Vessel, the highest 40-year design fatigue usage value is 0.37 for the closure studs. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.56. This value does not exceed the design limit of 1.0 and is therefore acceptable. This 60-year fatigue usage bounds the maximum environmentally adjusted usage factor of 0.1740 for the Reactor Vessel Outlet Nozzles. Therefore, the Reactor Vessel Fatigue Analyses have been projected to the end of the period of extended operation.

18.2.2.2 Reactor Vessel Internals Fatigue Analyses

A TLAA has been identified for the Reactor Vessel Internals. NSSS Design Transients were identified and forty-year design CUF values determined as part of the HNP steam generator replacement/uprating analysis. The Reactor Vessel Internals fatigue analysis demonstrated that if the Reactor Vessel Internals were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0. The 40-year design fatigue usage value is 0.52 for the core internals. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.78. This value does not exceed the design limit of 1.0. Therefore, the Reactor Vessel Internals Fatigue Analyses have been projected to the end of the period of extended operation.

18.2.2.3 Control Rod Drive Mechanism Fatigue Analysis

TLAAs have been identified for several sub-components of the Control Rod Drive Mechanism (CRDM). NSSS Design Transients were identified and forty-year design CUF values determined as part of the HNP steam generator replacement/uprating analysis. For the CRDM, the highest 40-year design fatigue usage value is 0.99 for the "Lower Joint Canopy Area." Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.49. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the CRDM will be managed for the period of extended operation.

18.2.2.4 Reactor Coolant Pumps Fatigue Analysis

The Reactor Coolant Pumps (RCPs) have been designed and analyzed to meet the ASME Code of record. The original design fatigue analysis was performed using fatigue waiver requirements. The pumps were therefore identified as having a TLAA. The RCP fatigue analysis demonstrated that if the RCPs were exposed to a bounding set of postulated transient cycles, the fatigue waiver would remain valid. The HNP RCP evaluation showed that the fatigue analysis was performed using the Code [NB-3222.4(d)] waiver of fatigue requirements. The original fatigue waiver evaluation was found to be bounding for the SGR/Uprating evaluation. Therefore, it was unnecessary to determine a 40-year or 60-year fatigue usage factor for the RCPs. Therefore, the RCPs Fatigue Analyses have been determined to remain valid for the period of extended operation.

18.2.2.5 Steam Generators Fatigue Analysis

TLAAs have been identified for several sub-components of the Steam Generators. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment. Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis. The Steam Generator fatigue analysis demonstrated that if the Steam Generator subcomponents were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0 with the exception of the Secondary Manway Bolts and the 4 in. Inspection Port Bolts. These components are discussed in more detail below.

Other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts, the highest 40-year design fatigue usage value is 0.98 for minor shell taps. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the steam generator (other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts) will be managed for the period of extended operation. The Steam Generator Secondary

Manway Bolts and 4 in. Inspection Port Bolts were identified as having 40-year design fatigue usage factors over 1.0. These components were characterized as "to be replaced based on a replacement schedule." However, HNP reanalyzed the Steam Generator Secondary Manway Cover Bolts and 4 in. Inspection Port Bolts to remove unnecessary conservatisms. The updated evaluation changed only the number of Unit Loading and Unit Unloading transient cycles relative to the previous design analysis. Each transient was considered to occur 2000 times over the life of the plant, a number which is still greater than the best estimate number provided in the previous design analysis. Fatigue usage values for the bolts based on the reduced Unit Loading and Unit Unloading cycles are as follows:

- Secondary Manway Cover Bolts: Fatigue Usage = 0.83
- 4 in. Inspection Port Bolts: Fatigue Usage = 0.81

Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a Fatigue Usage of 1.245 for the Secondary Manway Cover Bolt and a Fatigue Usage of 1.215 for the 4 in. Inspection Port Bolts. These values exceed the design limit of 1.0; and, therefore, an aging management program is required. The HNP Reactor Coolant Pressure Boundary Fatigue Management Program will ensure that the design allowable cycles for all transients (except Unit Loading and Unit Unloading) and the reduced number of Unit Loading and Unit Unloading transients is not exceeded or that appropriate re-evaluation or corrective action is taken. Therefore, the effects of fatigue on the Secondary Manway Bolts and the 4 in. Inspection Port Bolts will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

18.2.2.6 Pressurizer Fatigue Analysis

TLAAs have been identified for several sub-components of the Pressurizer. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see below and Subsection 18.2.2.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the steam generator replacement/uprating analysis. The Pressurizer fatigue analysis demonstrated that if the Pressurizer subcomponents were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0 for all components. However, certain locations of the Pressurizer lower head are not bounded by the original design fatigue analysis, because the original fatigue analysis did not consider insurge/outsurge transients that were identified subsequent to the original fatigue analysis.

For the Pressurizer (other than the lower head and surge line nozzle), the highest 40-year design fatigue usage value is 1.00 for the Trunnion Bolt Hole. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.50.

Recommendations of the Westinghouse Owners Group (WOG) were used to address operational pressurizer insurge/outsurge transients. These include reviewing plant operating records in sufficient detail to determine pressurizer insurge/outsurge transients for past operation, updating pressurizer lower head and surge nozzle transients to reflect past and projected future operations, and evaluating the impact of the updated transients on the structural integrity of the pressurizer. The WOG also recommended operating strategies that may be used in the future to address the insurge/outsurge issue. On January 20, 1994, HNP adopted the Modified Operating Procedures (MOP) recommended by the WOG to mitigate pressurizer insurge/outsurge transients.

Plant data was used to establish pre-MOP and post-MOP transients that represent past plant heat-up and cooldown operations. Fatigue evaluations of the pressurizer lower head and surge line nozzle were performed using the online monitoring and Westinghouse proprietary design analysis features of the WESTEMS™ Integrated Diagnostics and Monitoring System. The fatigue evaluations follow the procedures of ASME Code, Section III, NB-3200. The stress ranges, cycle pairing, and fatigue usage factors were calculated using WESTEMS™, consistent with the ASME Code and WOG recommendations. The fatigue evaluations were performed at critical locations of the pressurizer lower head (including the pressurizer surge line nozzle) and of the surge line RCS hot leg nozzle. The evaluations were based upon pre-MOP transients in conjunction with the post-MOP transients that include the effects of insurge/outsurge and surge line stratification. These transients were developed based upon plant-specific data and the information and guidelines provided by the WOG.

For the 40 years of plant life, the pressurizer lower head has the highest fatigue usage of 0.36 at the inside surface of the lower head at the heater penetration region. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a fatigue usage of 0.54. This location was also evaluated to account for the effects of reactor water environment on fatigue, and the 60-year fatigue usage for this location is 1.35. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the design limit fatigue usage is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the pressurizer will be managed for the period of extended operation.

18.2.2.7 Reactor Coolant Pressure Boundary Piping (ASME Class 1) Fatigue Analysis

TLAAs have been identified for components of the Reactor Coolant Pressure Boundary (RCPB) Piping. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately. Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis. The RCPB Piping fatigue analysis demonstrated that, if the RCPB piping components were exposed to a bounding set of postulated transient cycles, the CUF values for the components do not exceed 1.0. However, the Pressurizer Surge Line is not bounded by the original design fatigue analysis. In response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," (Reference 18.2.2-1), HNP evaluated the pressurizer surge line stratification transients separately for 40 years of operation.

For the components that are part of the RCPB Piping, the highest 40-year design fatigue usage value is 0.98 for the Pressurizer Spray Piping before the evaluation of the effects of reactor water environment on fatigue. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47.

Accounting for the effects of reactor water environment on fatigue, the highest 60-year fatigue usage is 2.120 for the pressurizer surge line. Since these values exceed the design limit of 1.0, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the design limit fatigue usage is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the reactor coolant pressure boundary piping will be managed for the period of extended operation

18.2.2.8 ASME Class 2 and 3 Piping Fatigue Analysis

HNP auxiliary piping that was designed to ASME Section III, Code Class 2 and 3 requirements did not require an explicit fatigue evaluation. Instead, for Class 2 and 3 piping, the Code includes implicit treatment of fatigue using a stress range reduction factor, f, which is a function of the total number of thermal expansion stress range cycles. The factor is equal to 1.0 for up to 7,000 cycles. For greater number of cycles, f may be further reduced, thereby reducing the thermal expansion range stress allowable. The affected Class 2 and 3 piping is effectively an extension of the adjacent Class 1 piping. Therefore, the cycle count depends closely on reactor operating cycles. Of those Normal Conditions listed that are likely to produce full-range thermal cycles in a 40-year plant lifetime are the 200 Heatup and Cooldown cycles. Assuming that all Upset Conditions lead to full-range thermal cycles adds an additional 980 cycles for a total of 1180 occurrences. The 980 cycles are equal to the summation of Upset Condition transients 1 through 12 plus 5 OBEs at 10 cycles each. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 1,770. This is only a fraction of the 7,000 full-range thermal cycles associated with a stress range reduction factor of 1.0. Therefore, the analysis for Class 2 and 3 piping remains valid for the period of extended operation using 10 CFR 54.21(c)(1) method (i).

18.2.2.9 ANSI B31.1 Piping Fatigue Analysis

In addition to ASME Class 2 and Class 3 piping, the scope of License Renewal at HNP includes non-safety related piping designed to ANSI B31.1. HNP auxiliary piping that was designed to ANSI B31.1 requirements did not require an explicit fatigue evaluation. Instead, for ANSI B31.1 piping, the "Power Piping" Code includes implicit treatment of fatigue using a stress allowable reduction factor, f, which is a function of the total number of thermal expansion stress range cycles. The factor is equal to 1.0 for up to 7,000 cycles. For greater number of cycles, f may be further reduced, thereby reducing the thermal expansion range stress allowable. For the Main Feedwater System, and associated systems such as the Condensate System, and Main Steam System, and associated systems such as the Steam Generator System, the thermal cycles anticipated correspond to Heatup and Cooldown cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 300. Therefore, main feedwater and main steam components will not experience 7,000 cycles during the period of extended operation.

The Auxiliary Feedwater System provides an alternate to the Feedwater System during startup, hot standby and cooldown and also functions as an Engineered Safeguards System. The total number of cycles expected in 40 years of operation are as follows: 200 Heatup and Cooldown cycles, 2,000 cycles of feedwater cycling at hot standby, 980 cycles associated with all Upset Conditions, 240 cycles of quarterly AFW pump tests in accordance with ASME Section XI, and 40 cycles of tests per the plant Technical Specifications. This yields a total of 3,460 cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 5,190. Therefore, auxiliary feedwater components will not experience 7,000 cycles during the period of extended operation.

The diesel generators in the Emergency Diesel Generator System undergo monthly surveillance tests in accordance with plant Technical Specifications. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 720. Therefore, the emergency diesel generator diesel exhaust piping will experience significantly fewer than 7,000 equivalent full temperature cycles during the period of extended operation.

The diesel generator in the Security Power System undergoes a monthly surveillance test to satisfy fire protection program surveillance requirements. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 720. Therefore, the security diesel generator diesel exhaust piping will experience significantly fewer than 7,000 equivalent full temperature cycles during the period of extended operation.

The diesel-driven fire pump in the Fire Protection System undergoes a monthly test to satisfy fire protection program surveillance requirements. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 720. Therefore, the diesel-driven fire pump piping will experience significantly fewer than 7,000 equivalent full temperature cycles during the period of extended operation.

Therefore, the analysis for B31.1 piping remains valid for the period of extended operation using 10 CFR 54.21(c)(1) method (i).

18.2.2.10 Environmentally-Assisted Fatigue Analysis

The effects of reactor water environment on fatigue were evaluated for a subset of representative components based on the evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Design Curves to Selected Nuclear Power Plant Components" (Reference 18.2.2-2). Since HNP Class 1 piping was designed in the more recent history of Westinghouse plant design, the locations corresponding to the "Westinghouse Newer Vintage Plant" were selected:

- Reactor Vessel Shell and Lower Head
- Reactor Vessel Inlet and Outlet Nozzles
- Pressurizer Surge Line
- Charging Nozzle
- Safety Injection Nozzle
- Residual Heat Removal (RHR) System Class 1 Piping

In addition to the above, locations in the pressurizer lower head that are potentially subject to insurge/outsurge transients were evaluated considering reactor water environmental effects. The environmental effects on fatigue were evaluated based on NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels" (Reference 18.2.2-3), NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels" (Reference 18.2.2-4), and NUREG/CR-6717, "Environmental Effects of Fatigue Crack Initiation in Piping and Pressure Vessel Steels" (Reference 18.2.2-5). Environmental fatigue penalty factors (Fen) were used to obtain adjusted cumulative fatigue usage (Uen) which includes the effects of reactor water environments.

For the charging nozzle, additional analyses were required for several "partial cycle" transients. This was to account for transients of much less severity than design, so that the less severe transients would not be counted as full design cycles. The ANSI B31.1 Power Piping Code, 1967 Edition, Section 102.3.2, provides the following equation and methodology for mathematically determining the number of equivalent full temperature range changes that result from the number of lesser temperature range changes:

$$N = N_E + r_1^5 N_1 + r_2^5 N_2 + \ldots r_n^5 N_n$$

Where:

N = the number of equivalent full temperature cycles,

NE = number of cycles at full temperature change for which expansion stress has been calculated

N_1, N_2_N_n = number of cycles at lesser temperature changes

r_1, r_2_r_n = ratio of lesser temperature cycles to the cycle for which the expansion stress has been calculated.

For this evaluation, the partial range cycles were converted to the equivalent number of full severity cycles, and the totals were adjusted accordingly. Using this approach, the results showed a considerable reduction in the number of equivalent full temperature range cycles. The methodology is not applicable to ASME Section III Class 1 piping. However, HNP performed a technical evaluation comparing the results from an independent ASME Section III, Division I, Subsection NB fatigue evaluation against the application of the ANSI B31.1 cycle reduction methodology to show that it was reasonable to use in the specific case of the charging nozzles.

The Reactor Vessel Shell and Lower Head and Reactor Vessel Inlet and Outlet Nozzles are addressed in 18.2.2.1 and their analyses has been projected through the period of extended operation using 10 CFR 54.21(c)(1) method (ii). Reactor Coolant Pressure Boundary Piping (ASME Class 1) components are addressed in 18.2.2.7. For these components, the effects of fatigue will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The pressurizer lower head and surge nozzle are addressed in 18.2.2.6, and the effects of fatigue will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

18.2.2.11 RCS Loop Piping Leak-Before-Break Analysis

In accordance with the CLB, a Leak-Before-Break (LBB) analysis was performed to show that any potential leaks that develop in the Reactor Coolant System (RCS) loop piping can be detected by plant leak monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. LBB evaluations postulate a surface flaw at a limiting stress location, and demonstrate that a through-wall crack will not result following exposure to a lifetime of design transients. A separate evaluation assumes a through-wall crack of sufficient size, such that the resultant leakage can be easily detected by the existing leakage monitoring system, and then demonstrates that, even under maximum faulted loads, this crack is much smaller than a critical flaw size that could grow to pipe failure. The aging effects to be addressed during the period of extended operation include thermal aging of the primary loop piping components and fatigue crack growth.

WCAP-14549-P, Addendum 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Harris Nuclear Plant for the License Renewal Program" (Reference 18.2.2-6), is a new LBB calculation applicable to HNP large bore 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. This calculation concluded:

1. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation. Currently an EPRI Material Reliability Program is underway to address the Alloy 82/182 Primary Water Stress Corrosion Cracking issue for the industry due to the V. C. Summer cracking incident. However, plant-specific calculations for Alloy 82/182 locations were performed to account for PWSCC crack morphology, and a conservative factor between fatigue cracking and PWSCC was used.

2. Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.

3. The effects of low and high cycle fatigue on the integrity of the primary piping are negligible. The fatigue crack growth evaluated is insignificant.

4. A margin of 10 exists between the leak rate of small stable leakage flaws and the capability (1 gpm) of the HNP RCS pressure boundary Leakage Detection System.

5. A margin of two or more exists between the small stable leakage flaw sizes of and the larger critical stable flaws.

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. Therefore, the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation. When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds" (Reference 18.2.2-7), has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.

18.2.2.12 Primary Sample Lines Fatigue Analysis

Portions of the Primary Sampling System may be subjected to thermal transients that are unrelated to those experienced by Class 1 components. There are three sample line penetrations involved - RCS Hot Leg (M-78A), Pressurizer Liquid Space (M-78B), and Pressurizer Steam Space (M-78C). The following analyses determined the number of cycles to which the relevant equipment would be subjected and compared them to the implicit fatigue analysis acceptance criterion of 7,000 cycles. The applied cycles are based on the manner in which they are used.

18.2.2.12.1 Penetration M-78A - RCS hot legs

The piping downstream of M-78A include three parallel branch lines that supply the Post-Accident Sample Panel in the Post-Accident Sampling System (PASS), the Primary Sample Panel in the Reactor Coolant Sample System, and the Gross Failed Fuel Detector in the Gross Failed Fuel Detection System. Depending on sample system operations, the number of thermal cycles experienced would be greater than the number of cycles caused by reactor shutdowns. Additional cycles would occur when sampling points are swapped or sampling equipment is isolated.

The number of cycles due to reactor shutdowns and the number of penetration M-78A isolations that would result in a thermal cycle were estimated based on a review of plant data. Extrapolating this data to a 60-year period, the number of cycles has been conservatively estimated to be:

- 1,000 thermal cycles from sample point swapping,
- 81 thermal cycles from reactor shutdown, and
- 270 thermal cycles from penetration isolations.

Therefore, total number of Hot Leg Thermal Cycles for penetration M-78A is 1,351 cycles, which is less than the criterion of 7,000 cycles.

18.2.2.12.2 Penetration M-78B - Pressurizer liquid space

This penetration supplies the primary sample panel and is cycled every time it is used to sample the Pressurizer liquid space. The number of thermal cycles was estimated to be 3,120 based on weekly sampling over a 60-year period. Because this line is connected to the Pressurizer, the estimated number of reactor thermal cycles over 60 years, i.e., 81, was added to the above value. This results in 3,201 thermal cycles, which is less than the criterion of 7,000 cycles.

18.2.2.12.3 Penetration M-78C - Pressurizer steam space

The sample line associated with the Pressurizer steam space penetration (M-78C) is normally used during degassing for reactor shutdown conditions and also for sampling following a postulated accident. A portion of the sample line is exposed to steam from the Pressurizer. Every time condensate is removed, the line has an opportunity to heat up again. It is conservatively assumed that the condensate will be drained at least once an operating cycle during testing of the isolation valves. Consequently, the number of thermal cycles for the sample lines is estimated to be approximately 81 cycles, which is less than the requisite 7,000 cycles. Since the total number of thermal cycles for the sample lines is less than 7,000 cycles, no reanalysis of the piping design is necessary.

Based on the above, the primary sample line design analyses of record remain valid for the period of extended operation.

18.2.2.13 Steam Generator Blowdown Lines Fatigue Analysis

Steam Generator blowdown flow is normally maintained during operation in order to maintain Steam Generator water chemistry. A thermal cycle in the Blowdown lines may result whenever blowdown flow is interrupted. These interruptions have the potential to result in thermal cycles over and above the heat-up and cooldown cycles of the RCS.

Blowdown interruptions were determined using actual plant data and projecting the number of interruptions through the period of extended operation. A conservative method was chosen such that one cycle was counted when blowdown flow was interrupted for more than 30 minutes. For the purposes of thermal fatigue, a complete thermal cycle is defined as a heat up from ambient to operating temperature followed by a cooldown to ambient temperature. The criterion adopted for counting thermal cycles is conservative because it includes interruptions of blowdown flow in which a significant decrease in temperature is not expected. This is based on the operating practice for reestablishing blowdown flow following a isolation which requires the downstream piping to be warmed-up prior to opening the isolation valves if the isolation valves were closed for more than 30 minutes. It follows then that if an isolation valve is closed for less than 30 minutes this does not constitute a significant cooldown period.

The number of cycles due to reactor shutdowns was added to the projected isolation cycles and resulted in a 60-year value of 404 cycles. Since the total number of projected thermal cycles for the Steam Generator Blowdown lines is less than 7,000 cycles, no reanalysis of the piping design calculations is necessary. Therefore, the Steam Generator Blowdown Line fatigue analysis remains valid for the period of extended operation.

18.2.3 ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT

The existing HNP EQ Program is credited for aging management of electric equipment important to safety in accordance with the requirements of 10 CFR 50.49. 10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). TLAA demonstration option 10 CFR §54.21(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, has been chosen. The EQ Program will manage the aging effects of the components associated with the environmental qualification TLAA.

18.2.4 CONTAINMENT MECHANICAL PENETRATION BELLOWS FATIGUE

18.2.4.1 Mechanical Penetration Bellows - Valve Chambers

The four mechanical penetration bellows addressed by this section are the Containment Spray and Safety Injection System Recirculation Valve Chamber Bellows associated with containment penetrations M-47 through M-50. Per the plant specifications, the valve chamber bellows expansion joint design is in accordance with ASME Section III, Paragraph NC-3649.1 so that no single corrugation is permitted to deflect more than its maximum allowable amount. Each bellows is designed to withstand a total of 7,000 cycles of expansion and compression over its lifetime due to maximum normal operating conditions plus 10 cycles of movement due to safe shutdown earthquake condition.

Operating cycles of expansion and compression due to maximum normal operating conditions was calculated by adding the number of containment cycles corresponding to RCS heat-up and cooldown cycles plus the number of times the containment is pressurized during Type A Integrated Leak Rate Testing (ILRT) plus the number times a Type B local leak rate test (LLRT) is performed.

The expansion bellows is the barrier between the valve chamber and the Reactor Auxiliary Building. The containment isolation valves associated with these chambers isolate the containment sumps from the Containment Spray and RHR Systems and, therefore, do not normally experience any fluid flow. Operation of RHR during cool-down of the RCS would have a negligible impact on the bellows due to the piping configuration but are included since operation of RHR would typically correspond to the Reactor Coolant System (Class 1) cycles.

The number of Reactor Thermal Cycles projected over 60 years is 81 cycles. The containment ILRT is performed infrequently, i.e., once every 10 years. Conservatively assuming an ILRT will be performed once every 5 years rather than the maximum period of 10 years yields 12 cycles. Per Type B Local Leak Rate Test program the maximum test interval for this equipment is 24 months. Since this is the maximum interval, the minimum will be conservatively assumed to be yearly resulting in an additional 60 cycles. The total number of cycles anticipated for 60 years is

81 + 12 + 60 = 153 cycles.

Since the total number of thermal cycles for the Containment Spray and Safety Injection System Recirculation Valve Chamber Bellows is less than 7,000 cycles, no reanalysis of the design calculations is necessary. Therefore, the Containment Spray and Safety Injection System Recirculation Valve Chamber

Bellows design analyses of record remain valid for the period of extended operation.

18.2.4.2 Mechanical Penetration Bellows - Fuel Transfer Tube Bellows Expansion Joint

Per plant specifications, the Fuel Transfer Tube bellows-expansion-joint design is in accordance with ASME Section III, Paragraph NC-3649.1, and such that no single corrugation is permitted to deflect more than its maximum allowable amount. Each bellows shall be designed to withstand a total of 7,000 cycles of expansion and compression over its service lifetime, due to maximum normal operating conditions, plus 10 cycles of movement due to the safe shutdown earthquake condition.

The expansion cycles would occur when the tube is flooded between the transfer canal in the Containment Building and the Fuel Handling Building. This typically occurs twice every refueling cycle. The maximum number of operating cycles projected to be experienced over a 60-year period is 80 cycles assuming a refueling outage every 1.5 years. Since the total number of thermal cycles for the Fuel Transfer Tube Bellows Expansion Joint is less than 7,000 cycles, no reanalysis of the design calculations is necessary. Therefore, an evaluation was performed and was successful in demonstrating that the Fuel Transfer Tube Bellows Expansion Joint design analyses of record remain valid for the period of extended operation.

18.2.5 TURBINE ROTOR MISSILE GENERATION ANALYSIS

Please reference section 3.5.1.3.2 for a summary of the Turbine Missile Analysis conducted as part of the Low Pressure Turbine Modernization.

18.2.6 CRANE CYCLIC LOAD ANALYSIS

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

The Polar Crane, Jib Cranes, and Reactor Cavity Manipulator Crane in the Containment Building and the Fuel Cask Handling Crane, Fuel Handling Bridge Crane, and Fuel Handling Building Auxiliary Crane in the Fuel Handling Building have been identified as involving 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.

18.2.6.1 Polar Crane

The Polar Crane is a low-cycle lifting device. The total number of load cycles projected for 60-years for the Main Hook is 4,020 and for the Auxiliary Hook is 1,600. The combined total is 5,620 cycles applied to the crane structure. This is below the 20,000 to 100,000 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Polar Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.6.2 Jib Cranes

The Containment Jib Cranes are low load capacity lifting devices used for refueling and maintenance activities. The total number of load cycles projected for 60-years is 18,800. This is less than the 20,000 to 100,000 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Jib Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.6.3 Reactor Cavity Manipulator Crane

The Reactor Cavity Manipulator Crane provides the flexibility to grip, remove, and replace fuel assemblies to support refueling operations. The total number of load cycles projected for 60-years is 16,824. This is less than the 107 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Reactor Cavity Manipulator Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.6.4 Fuel Cask Handling Crane

Fuel Cask Handling Crane is a low cycle lifting device. The total number of load cycles projected for 60-years is 8,750. This is less than the 20,000 to 100,000 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Fuel Cask Handling Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.6.5 Fuel Handling Bridge Crane

The Fuel Handling Bridge Crane provides the ability to place, remove, and replace fuel assemblies and appurtenances to support fuel handling operations. The total number of load cycles projected for 60-years is 27,558. This is less than the 107 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Fuel Handling Bridge Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.6.6 Fuel Handling Building Auxiliary Crane

The Fuel Handling Building Auxiliary Crane is used to support the refueling process by handling of the removable barrier, pool gates, fuel racks and other miscellaneous items. The total number of load cycles projected for 60-years is 15,380. This is less than the 20,000 to 100,000 permissible cycles originally projected for 40 years and is therefore acceptable. Based on the forgoing, the HNP Fuel Handling Building Auxiliary Crane has been evaluated, and the fatigue analysis remains valid for 60-years.

18.2.7 MAIN AND AUXILIARY RESERVOIR SEDIMENTATION ANALYSIS

The Auxiliary Reservoir functions as the ultimate heat sink for HNP, and the Main Reservoir functions as a backup in case the Auxiliary Reservoir is not available. The HNP FSAR addresses sedimentation and concludes that the effects of sediment deposit on the reservoir operations and cooling capacities will be negligible for the current 40-year operating license. Therefore, sedimentation in the Main and Auxiliary Reservoirs at HNP was considered to be a TLAA, since the sedimentation effects are based on a 40-year plant life. A simple calculation of sedimentation based on the ratio of 60 years to 40 years projects values that would have a negligible effect on the capability of the reservoirs. However, HNP intends to use the Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program, to manage the potential effects of sedimentation. The Program monitors the Main and Auxiliary Reservoir shoreline, reservoirs, and drainage area for landslides, excessive sedimentation, or developments in the drainage basin that could cause a sudden increase in sediment load, which would reduce the reservoir capacity. The frequency of the inspection of the Auxiliary and Main Reservoirs is every five years. Therefore, the continued implementation of the HNP Regulatory Guide 1.127 Program, will manage the effects of sedimentation in the Main and Auxiliary Reservoirs during the period of extended operation.

18.2.8 HIGH ENERGY LINE BREAK LOCATION POSTULATION BASED ON FATIGUE CUMULATIVE USAGE FACTOR

Section 3.6 of the FSAR describes the design bases and measures that are taken to demonstrate that the systems, components and structures required to safely shutdown and maintain the reactor in a cold shutdown condition are adequately protected against the effects of blowdown jets, reactive forces, and pipe whip resulting from postulated rupture of piping both inside and outside Containment.

RG 1.46 has been followed in all matters except for the postulation of break points. The criteria of NRC Branch Technical Position MEB 3-1 for Class 1 piping has been adapted such that pipe breaks are postulated to occur at:

1. Terminal ends.

2. Intermediate locations where the maximum stress range as calculated by Eq. (10) and either (12) or (13) exceeds 2.4 Sm.

3. Intermediate locations where the cumulative usage factor exceeds 0.1.

The calculation of cumulative usage factors used design cycles associated with a 40-year design life. Since the design cycles used in these evaluations are associated with a 40-year design life, the high-energy line-break postulation based on cumulative usage factor is considered a TLAA.

Original fatigue design calculations assumed a large number of design transients, corresponding to relatively severe system dynamics over the original 40-year design life. The current design fatigue usage factors will remain valid during the period of extended operation as long as the number of design transients is not exceeded.

The HNP Fatigue Monitoring Program will identify when piping systems are approaching the original 40-year number of design transients. Prior to any piping system exceeding its original number of design transients, the pertinent design calculations for that system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of Section 3.6 of the FSAR. If other locations are determined to require consideration as postulated break locations, appropriate actions will be taken to address the new break locations.

REFERENCES: SECTION 18.2

18.2.1-1 BAW-2241P-A, "Fluence and Uncertainty Methodologies," Rev. 1, AREVA, December 1999.

18.2.2-1 NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," December 1988.

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TABLE	TITLE
N/	THERE ARE NO TABLES FOR CHAPTER 18