Progress Energy

AUG 2 0 2007

SERIAL: HNP-07-112
10 CFR 54

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

Subject: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / LICENSE NO. NPF-63

LICENSE RENEWAL APPLICATION, AMENDMENT 1: CHANGES RESULTING FROM RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION, SITE AUDIT QUESTIONS, AND APPLICANT-IDENTIFIED CHANGES

References:
1. Letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006
2. Letter from Dave Corlett to the U. S. Nuclear Regulatory Commission (Serial: HNP-07-032), "Response to Request for Additional Information 2.3.3.31-1 and 2.3.3.31-2 for License Renewal," dated March 23, 2007
3. Letter from Christopher L. Burton to the U. S. Nuclear Regulatory Commission (Serial: HNP-07-087), "License Renewal Application - Responses to Requests For Additional Information 2.1-1 and 3.0-1," dated July 10, 2007
4. Letter from T. J. Natale to the U. S. Nuclear Regulatory Commission (Serial: HNP-07-104), "Responses to Requests for Additional Information - License Renewal Application Section 4.2 and Subsection B.2.17," dated August 16, 2007

Ladies and Gentlemen:

On November 14, 2006, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, requested renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

By letters dated March 23, July 10, and August 16, 2007, Progress Energy Carolinas responded to NRC Requests for Additional Information (RAIs) regarding License Renewal issues. In addition, from May 21 to 25 and from June 25 to 29, 2007, the NRC conducted audits of HNP License Renewal activities related to aging management programs (AMPs)

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and aging management reviews (AMRs). In the course of these audits, questions were identified by the auditors. Responses to the audit questions are included in this letter.

Also, enclosed is the list of regulatory commitments supporting License Renewal modified to reflect the information provided in the responses to the audit questions and RAIs. Any other actions discussed should be considered intended or planned actions; they are included for informational purposes but are not considered to be regulatory commitments.

Based on the above activities and internal HNP reviews, required changes to the HNP License Renewal Application (LRA) have been identified. This LRA amendment consists of three enclosures. Enclosure 1 is the revised list of License Renewal Commitments. Enclosure 2 is a table that identifies the changes to the LRA and the source of those changes. Enclosure 3 is a report of AMP-related and AMR-related questions and responses from the NRC audits.

Please refer any questions regarding this submittal to Mr. Roger Stewart, Supervisor - License Renewal, at (843) 857-5375.

I declare, under penalty of perjury, that the foregoing is true and correct (Executed on AUG 2 0 2007).

Sincerely,

Christopher L. Burton

Christopher L. Burton
Director – Site Operations
Harris Nuclear Plant

CLB/mhf

Enclosures:

1. HNP License Renewal Commitments, Revision 1
2. Amendment 1 Changes to the License Renewal Application
3. Harris Nuclear Plant License Renewal Audit Question and Response Database Report

Mr. P. B. O'Bryan (NRC Senior Resident Inspector, HNP)
Ms. B. O. Hall (Section Chief, N.C. DENR)
Mr. M. L. Heath (NRC License Renewal Project Manager, HNP)
Ms. M. G. Vaaler (NRC Project Manager, HNP)
Dr. W. D. Travers (NRC Regional Administrator, Region II)

HARRIS NUCLEAR PLANT LICENSE RENEWAL COMMITMENTS, REVISION 1				
ITEM NO.	COMMITMENT	FINAL SAFETY ANALYSIS REPORT (FSAR) SUPPLEMENT LOCATION	PROGRAM IMPLEMENTATION SCHEDULE	LICENSE RENEWAL APPLICATION (LRA) SOURCE
1	In accordance with the guidance of NUREG-1801, Rev. 1, regarding aging management of reactor vessel internals components, 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.	A.1.1	As stated in the commitment	Reactor Vessel Internals Aging Management Activities LRA Section A.1.1
2	In accordance with the guidance of NUREG-1801, Rev. 1, regarding aging management 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.	A.1.1	As stated in the commitment	Primary Water Stress Corrosion Cracking of Nickel Alloys LRA Section A.1.1
3	Program inspections are performed as augmented inspections in the HNP Inservice Inspection (ISI) Program. The ISI Program administrative controls will be enhanced to specifically identify the requirements of NRC Order EA-03-009.	A.1.1.5	Prior to the period of extended operation	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program LRA Section B.2.5
4	The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program to be implemented.	A.1.1.6	Prior to the period of extended operation	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program LRA Section B.2.6

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5	The 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.	A.1.1.7	Prior to the period of extended operation	The Flow-Accelerated Corrosion (FAC) Program LRA Section B.2.7
6	A precautionary note will be added to plant bolting guidelines to prohibit the use of molybdenum disulfide lubricants.	A.1.1.8	Prior to the period of extended operation	Bolting Integrity Program LRA Section B.2.8
7	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.	A.1.1.9	Prior to the period of extended operation	Steam Generator Tube Integrity Program LRA Section B.2.9
8	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.	A.1.1.12	Prior to the period of extended operation, unless an approved analysis exists that eliminates credit for the Boraflex in the BWR fuel racks	Boraflex Monitoring Program LRA Section B.2.12
9	The Program will be enhanced to: (1) include in the Program all cranes 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.	A.1.1.13	Prior to the period of extended operation	Inspection of Overhead Heavy Load and Light Load Handling Systems Program LRA Section B.2.13; Response to Audit Question B.2.13-JW-01.

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10	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.	A.1.1.14	Prior to the period of extended operation	Fire Protection Program LRA Section B.2.14
11	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.	A.1.1.15	Prior to the period of extended operation	Fire Water System Program LRA Section B.2.15 Commitment (1)(b) and the option of using a combination of (1)(a) and (1)(b) were added in the response to Audit Question B.2.15-PB-01. Commitment (3) was added per Audit Question 3.3.1-70-MK-01

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12	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 and 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.	A.1.1.16	Prior to the period of extended operation	Fuel Oil Chemistry Program LRA Section B.2.16, Response to Audit Question B.2.16-MK-12.

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13	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 W) will occur during Refueling Outage 16, at which time the capsule fluence is projected to be equivalent to the 60-year maximum vessel fluence of 6.8×10^{19} n/cm^2 in accordance with ASTM E 185-82, (3) include a provision that analysis of Capsule W be used to evaluate neutron exposure for remaining Capsules Y and Z, as required by 10 CFR 50 Appendix H. The withdrawal schedule for one of the remaining capsules will be adjusted, based on the analysis of Capsule W, so that the capsule fluence will not exceed twice the 60-year maximum vessel fluence in accordance with ASTM E 185-82. The neutron exposure and withdrawal schedule for the last capsule will be optimized to provide meaningful metallurgical data. If the last capsule is projected to significantly exceed a meaningful fluence value, it will either be relocated to a lower flux position or withdrawn for possible testing or re-insertion. Capsules Y and Z and archived test specimens available for reconstitution will be available for the monitoring of neutron exposure if additional license renewals are sought, and (4) 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.	A.1.1.17	Prior to the period of extended operation	Reactor Vessel Surveillance Program LRA Section B.2.17, RAI-B.2.17
14	The One-Time Inspection Program is a new program to be implemented.	A.1.1.18	Prior to the period of extended operation	One-Time Inspection Program LRA Section B.2.18

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15	The Selective Leaching of Materials Program is a new program to be implemented	A.1.1.19	Prior to the period of extended operation	Selective Leaching of Materials Program LRA Section B.2.19
16	The Buried Piping and Tanks Inspection Program is a new program to be implemented.	A.1.1.20	Prior to the period of extended operation	Buried Piping and Tanks Inspection Program LRA Section B.2.20
17	The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a new program to be implemented.	A.1.1.21	Prior to the period of extended operation	One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program LRA Section B.2.21
18	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.	A.1.1.22	Prior to the period of extended operation	External Surfaces Monitoring Program LRA Section B.2.22
19	The 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.	A.1.1.23	Prior to the period of extended operation	Flux Thimble Tube Inspection Program LRA Section B.2.23

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20	The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program to be implemented.	A.1.1.24	Prior to the period of extended operation	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program LRA Section B.2.24
21	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.	A.1.1.25	Prior to the period of extended operation	Lubricating Oil Analysis Program LRA Section B.2.25
22	The 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.	A.1.1.26	Prior to the period of extended operation	ASME Section XI, Subsection IWE Program LRA Section B.2.26
23	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.	A.1.1.29	Prior to the period of extended operation	10 CFR Part 50, Appendix J Program LRA Section B.2.29
24	Program administrative controls will be enhanced to identify the structures that have masonry walls in the scope of License Renewal.	A.1.1.30	Prior to the period of extended operation	Masonry Wall Program LRA Section B.2.30

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25	The 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.	A.1.1.31	Prior to the period of extended operation	Structures Monitoring Program LRA Section B.2.31
26	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.	A.1.1.32	Prior to the period of extended operation	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program LRA Section B.2.32
27	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.33	Prior to the period of extended operation	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program LRA Section B.2.33

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ITEM NO.	COMMITMENT	FINAL SAFETY ANALYSIS REPORT (FSAR) SUPPLEMENT LOCATION	PROGRAM IMPLEMENTATION SCHEDULE	LICENSE RENEWAL APPLICATION (LRA) SOURCE
28	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new program to be implemented.	A.1.1.34	Prior to the period of extended operation	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program LRA Section B.2.34
29	The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.35	Prior to the period of extended operation	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program LRA Section B.2.35
30	The Metal Enclosed Bus Program is a new program to be implemented.	A.1.1.36	Prior to the period of extended operation	Metal Enclosed Bus Program LRA Section B.2.36
31	The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.37	Prior to the period of extended operation	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program LRA Section B.2.37

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32	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 of cumulative usage, (4) describe the acceptance criteria for maintaining fatigue usage below the design limit, and (5) address corrective actions for components approaching design limits, with options to include a revised fatigue analysis or repair or replacement of the component.	A.1.1.38	Prior to the period of extended operation	Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program LRA Section B.3.1
33	The Low Temperature Overpressure (LTOP) setpoint analysis will be recalculated following removal of one of the remaining surveillance capsules from the reactor vessel.	A.1.2.1.4	After capsule fast neutron exposure comparable to the end of the period of extended operation	TLAA – Low temperature Over-Pressure Limits LRA Section 4.2.5
34	The Oil-Filled Cable Testing Program is a new program to be implemented.	A.1.1.40	Prior to the period of extended operation	Oil-Filled Cable Testing Program LRA Section B.2.38, Response to Audit Question LRA-3.6.2-1-RM-02

Amendment 1 Changes to the License Renewal Application

Source of Change	License Renewal Application Amendment 1 Changes
Progress Energy Letter to NRC, HNP-07-032, dated March 23, 2007 RAI 2.3.3.31-1, RAI 2.3.3.31-2	As indicated in the response to the Request for Additional Information (RAI) in Progress Energy letter to NRC, (Serial HNP-07-032): "Response to Request for Additional Information 2.3.3.31-1 and 2.3.3.31-2 for License Renewal," add the commodity "Fire Barrier Assemblies," with a "C-4 Fire Barrier" intended function, to LRA Table 2.4.2-11, COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DIESEL FUEL OIL STORAGE TANK BUILDING
Progress Energy Letter to NRC, HNP-07-087, dated July 10, 2007 RAI 2.1-1	The response to RAI 2.1-1 in Progress Energy letter to NRC, (Serial HNP-07-087): "License Renewal Application - Responses to Requests For Additional Information 2.1-1 and 3.0-1," resulted in the following changes to LRA Subsection 2.1.1.2, Seismic-Connected Piping. Revise the second sentence on Page 2.1-13 to read: The following quotes are from that discussion (excluding the Main Steam and Feedwater interface restraints): In addition, insert the following paragraph on Page 3.1-13 prior to the heading 10 CFR 54.4(a)(2) Scoping Summary: The system of interface restraints for the Main Steam and Feedwater lines are attached to the outside wall of the Reactor Auxiliary Building as described in FSAR Section 3.6.2.1.2. Also, see Figures 3.6.2-1 and 1.2.2-80 in the FSAR. This design supports the creation of a pipe break exclusion zone.
Progress Energy Letter to NRC, HNP-07-104, dated August 16, 2007 RAI B.2.17, RAI 4.2.3, RAI 4.2.4, RAI 4.2.5, and RAI 4.2.6	Revise Subsection B.2.17 and Subsections 4.2.1, 4.2.3, 4.2.4, and 4.2.5 of the LRA to include changes conforming to the responses to RAIs contained in Progress Energy letter to NRC, (Serial HNP-07-104): "Responses to Requests for Additional Information - License Renewal application Sections 4.2 and Subsection B.2.17." Refer to Enclosure 2, Attachment 4. This change impacts License Renewal Commitment #13.
Audit Question B.2.6-FS-02	Delete the last sentence in LRA Subsection A.1.1.6, and Revise LRA Subsection B.2.6, NUREG-1801 Consistency, to read: The HNP Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program that is consistent with NUREG-1801, Section XI.M13, with exception. Add an exception to Subsection B.2.6 by replacing "None" with the following in the Exceptions to NUREG-1801 discussion: Program Elements Affected • **Corrective Actions** HNP is updating the ISI Program to ASME Section XI, 2001 edition with addenda through 2003, per 10 CFR 50.55a. This differs from the Code edition specified in NUREG-1801. In the newer edition/addenda of the Code, repair/replacement activities are controlled under Article IWA-4000, and it is stated that the term repair/replacement activity includes those activities previously known as repair, replacement, modification, or alteration. The IWX-4000 and IWX-7000 articles of Section XI have been incorporated into IWA-4000.

Source of Change	License Renewal Application Amendment 1 Changes
Audit Question B.2.10-MK-01	Change the NUREG-1801 Consistency Statement in LRA Subsection B.2.10 to read: The Open-Cycle Cooling Water System Program is an existing program that is consistent with NUREG-1801, Section XI.M20.
Audit Question B.2.15-PB-01	Enhancement (1) of LRA Subsection A.1.1.15 and the enhancements for LRA B.2.15 under program element for **Parameters Monitored/Inspected** are changed to read: Revise the Program to 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. This change impacts License Renewal Commitment #11.
Audit Questions B.2.16-MK-11 and B.2.16-MK-12	Replace the second, third, and fourth sentences of LRA Subsection A.1.1.16 and the fifth and sixth sentences of LRA Subsection B.2.16 Program Description with the following: 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. In Subsection B.2.16 under the Exceptions to NUREG-1801 for Preventive Actions, change item 1) to a Note, renumber the remaining items, and revise new item 3) as shown: Note: A stabilizer containing a biocide and corrosion inhibitor is added to new fuel before it is added to the storage tanks in the Diesel Generator Fuel Oil Storage and Transfer System and the Security Power System. 3) The use of stabilizers, biocides, and corrosion inhibitors in the diesel-driven fire pump fuel oil tank is not warranted, as fuel oil is frequently refreshed. The consumption of fuel oil is the result of the monthly requirement in the Fire Protection Program to run the pump for 30 minutes on relief flow. The frequent addition of diesel fuel oil obviates the need for these additives. Additionally, site operating experience does not show adverse trends in corrosion in the fuel oil components. (continued)

Source of Change	License Renewal Application Amendment 1 Changes
Audit Questions B.2.16-MK-11 and B.2.16-MK-12 (continued)	In Subsection B.2.16 under the Exceptions to NUREG-1801 for Monitoring and Trending, item 2) is changed to say: The Security Power System buried tank is monitored semiannually not quarterly. This exception is acceptable because operating experience shows no evidence of corrosion or biological growth since the installation of the new tank, use of the Diesel Grade No. 1-D, and use of a fuel oil stabilizer containing a biocide and corrosion inhibitors. This covers about eight years of operating experience. If Diesel No. 2-D is used in the future, the monitoring, except for biological growth, will be performed on a quarterly basis for the main storage tank only. (Note: Replacement of the tank was done to comply with more stringent state and federal codes for buried fuel oil tanks.) Add the following to Subsection A.1.1.16, paragraph two, after item (6) and to LRA Subsection B.2 16, Enhancements, Monitoring and Trending items 1) and 3). This change impacts License Renewal Commitment #12. Day tank sampling for water and 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.
Audit Question B.2.23-CM-01	Monitoring and Trending enhancement 1) for Subsection B.2.23 of the LRA is changed to delete the statement: If the generic value of 0.67 is used for "n", a basis must be provided for using the generic value in lieu of plant-specific data.
Audit Question B.2.26-JW-04	The following statement has been added to the Scope of Program element exception in LRA Subsection B.2.26: In conformance with 10 CFR 50.55a(g)(4)(ii), the ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.
Audit Question B.2.27-SA-02	The following statement has been added to the Scope of Program element exception in LRA Subsection B.2.27: In conformance with 10 CFR 50.55a(g)(4)(ii), the ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.
Audit Question B.2.28-SA-02	Remove the discussion of snubbers, the ASME OM Code, and ASME OM Code Case OMN-13 from the LRA discussion of the ASME Section XI, Subsection IWF Program as follows: 1) Revise LRA Subsection A.1.1.28 and the Program Description of Subsection B.2.28 by deleting the words "other than snubbers" from the third sentence and by deleting the last two sentences, and 2) Revise the LRA Subsection B.2.28 Scope of Program Exception by deleting the third sentence.

<table>
<tr><th>Source of Change</th><th>License Renewal Application Amendment 1 Changes</th></tr>
<tr><td>Audit Question B.2.33-RM-01</td><td>Revise the Operating Experience (OE) discussion in LRA, Appendix B, Subsection B.2.33 to include the following paragraph:

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Similarly, the Operating Experience discussions in LRA Subsections B.2.34, B.2.35, B.2.36, and B.2.37 are revised to read as stated above. These words have been incorporated into the changes implemented under Audit Question AMP-MH-01 below.</td></tr>
<tr><td>Audit Question B.2.34-RM-02</td><td>Revise Subsection A.1.1.34 and the Program Description of Subsection B.2.34 of the LRA as follows:

Change the fourth sentence to read:

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.

Change the sixth sentence to read:

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.

Add the following new sentence after sentence six:

The scope includes source, intermediate, and power range nuclear instrumentation circuits, and the RG 1.97 wide range neutron flux monitoring circuits.</td></tr>
<tr><td>Audit Question B.2.37-XI.E6-RM-02</td><td>These changes are based on a combination of NUREG-1801, Rev. 1, AMP XI.E6 and NRC letter, dated March 16, 2007: "Staff Response to the Nuclear Energy Institute (NEI) White Paper on Generic Aging Lessons Learned (GALL) Report Aging Management Program (AMP) XI.E6, 'Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.'" The NRC letter was issued after the submittal date of the HNP LRA. Revise Subsection A.1.1.37 and B.2.37 to read as shown in Attachment 1 to this Enclosure (i.e., <u>Enclosure 2, Attachment 1</u>).</td></tr>
</table>

<table>
<tr><th>Source of Change</th><th>License Renewal Application Amendment 1 Changes</th></tr>
<tr><td>Audit Question B.3.2-X.E1-RM-02</td><td>Delete the last sentence in LRA Subsections A.1.1.39 and A.1.2.3, and revise LRA Subsection B.3.2, NUREG-1801 Consistency, to read:

The EQ Program is an existing program consistent with NUREG-1801, Section X.E1, with exception.

Add exceptions to Subsection B.3.2 by adding the following paragraph to the section entitled Exceptions to NUREG-1801 applicable to the program elements for Scope of Program and Parameters Monitored/Inspected:

The original licensing basis of the EQ Program is Regulatory Guide (RG) 1.89, Rev. 0, not RG 1.89, Rev. 1, as specified in NUREG-1801, Section X.E1. HNP was originally licensed as a NUREG-0588, Category II plant, and IEEE Std 323-1971 was the original EQ Program basis. RG 1.89, Rev. 1, had not been issued when the HNP construction permit was issued. RG 1.89, Rev. 1, describes a method acceptable to the NRC staff for complying with §50.49 of 10 CFR Part 50. The acceptable method follows the procedures described by IEEE Std 323-1974. Currently, the HNP EQ Program meets the requirements of 10 CFR 50.49 (as further defined and clarified by NUREG-0588, and RG 1.89, Rev. 1) for the applicable electrical components important to safety. Under 10 CFR §54.21(c)(1)(iii), the HNP EQ Program, which implements the requirements of 10 CFR 50.49, is viewed as an aging management program for License Renewal. Section 4.4.2.1.3 of the "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR) states that the staff evaluated the EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii).</td></tr>
<tr><td>Audit Question B.3.2-RM-02</td><td>Add the following sentence to the Program Description of LRA Subsection B.3.2:

Refer to section 4.4 for a discussion of EQ Program reanalysis attributes.</td></tr>
<tr><td>Audit Question LRA 3.5.1-26-JW-01</td><td>Add the following, to the third paragraph under Section 3.5.2.2.2.2.1, Freeze-Thaw, on LRA Page 3.5-34:

Consistent with a Structures Monitoring Program enhancement, examination of inaccessible Class 1 Concrete will be performed when excavated for any reason.</td></tr>
<tr><td>Audit Question LRA 3.5.1-32-JW-01</td><td>Remove the entire aging management review (AMR) line item associated with Table 1 Item 3.5.1-32 from AMR Tables 3.5.2-2, 3.5.2-10, 3.5.2-12, 3.5.2-17, 3.5.2-27 and 3.5.2-28 for the concrete exterior below grade and concrete foundation component/commodity groups.</td></tr>
<tr><td>Audit Question LRA 3.6.2-1-RM-02</td><td>Provide a new, plant-specific AMP for the HNP high-voltage, oil-filled cables, and revise LRA Tables 3.6.2-1, Subsection B.1.1, and Table B-1 to incorporate the new program. This change is the genesis of License Renewal Commitment #34. These LRA changes are shown in Attachment 2 to this Enclosure (i.e., <u>Enclosure 2, Attachment 2</u>).</td></tr>
<tr><td>Audit Question AMP-MH-01</td><td>Provide additional program element discussions for new AMPs. The additional information is provided in Attachment 3 to this Enclosure (i.e., <u>Enclosure 2, Attachment 3</u>).</td></tr>
<tr><td>Audit Question 3.1-FS-05</td><td>Revise LRA Table 3.1.2-1, on Page 3.1-46, for the Reactor Vessel; CRDM Head Penetration Thermal Sleeves, to replace the AMR line item showing no aging effects with the following:

<table>
<tr><td>Cracking due to SCC</td><td>Water Chemistry and One-Time Inspection</td><td>VIII.B1-5 (SP-17)</td><td>3.4.1-14</td><td>C</td></tr>
<tr><td>Loss of Material due to Crevice Corrosion
Loss of Material due to Pitting Corrosion</td><td>Water Chemistry</td><td>IV.A2-14 (RP-28)</td><td>3.1.1-83</td><td>A</td></tr>
</table></td></tr>
</table>

<table>
<tr><td>Audit Question 3.1-FS-06</td><td>
Revise LRA Table 3.1.2-1, on Page 3.1-48, for the Reactor Vessel; Primary Nozzles in Treated Water (Inside), to add the following AMR line item:
<table>
<tr><td>Loss of Fracture Toughness due to Neutron Irradiation Embrittlement</td><td>TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG1.99</td><td>IV.A2-16 (R-81)</td><td>3.1.1-17</td><td>A, 126</td></tr>
</table>
In addition, add new Plant-Specific Note 126 to read:
The HNP reactor vessel does not have safety injection nozzles. The reactor vessel outlet nozzles were not identified as components expected to receive fluences greater than 10^{17} n/cm^2 (E > 1.0 MeV). Therefore, the reactor vessel outlet nozzles do not apply to GALL Report, Volume 2, Item IV.A2-16 (R-81). Five other reactor coolant pressure boundary components outside the beltline region are expected to receive fluences greater than 10^{17} n/cm^2, (E > 1.0 MeV). These components include: (1) the circumferential weld that is between the upper and intermediate shells, (2) the upper shell, (3) the inlet nozzle welds, (4) the inlet nozzle, and (5) upper shell longitudinal welds. These components were evaluated and none of these materials were determined to be limiting in ART, C_VUSE or RT_{PTS} values.
</td></tr>
<tr><td>Audit Question 3.1-FS-31</td><td>
Add the following new AMR lines applicable to Nickel Base Alloys in Treated Water (Inside) for the Pressurizer Spray Nozzle Safe End and Pressurizer Relief Nozzle Safe end, on LRA Page 3.1-121, and the Pressurizer Safety Nozzle Safe End on LRA Page 3.1-122:
<table>
<tr><td>Cracking due to Thermal Fatigue</td><td>TLAA, evaluated in accordance with 10 CFR 54.21(c)</td><td>IV.C2-25 (R-223)</td><td>3.1.1-08</td><td>A, 125</td></tr>
<tr><td>Cracking due to SCC</td><td>ASME Section XI Inservice Inspection and Water Chemistry</td><td>IV.C2-24 (R-22)</td><td>3.1.1-31</td><td>B, 125</td></tr>
<tr><td>Loss of Material due to Crevice Corrosion
Loss of Material due to Pitting Corrosion</td><td>Water Chemistry</td><td>IV.C2-15 (R-23)</td><td>3.1.1-83</td><td>A, 125</td></tr>
</table>
Also, replace Plant-Specific Note 124 with Plant-Specific Note 125 for the above nozzle safe end Nickel Base Alloy material in an Air-Indoor (Outside) environment.
</td></tr>
<tr><td>Audit Question 3.1-FS-35</td><td>Revise LRA Table 3.1.2-6, on Page 3.1-136, for the AMR line item applicable to Loss of Material due to Flow Accelerated Corrosion for the Feedwater Nozzle to change Table 1 Item from 3.3.1-59 to 3.1.1-59.</td></tr>
<tr><td>Audit Question 3.1-FS-50</td><td>
Amend LRA Table 3.1.2-6 for Feedwater Nozzle Thermal Sleeve on Page 3.1-137 and for Auxiliary Nozzle Thermal Sleeve on Page 3.1-138 with intended function M-6. For these two thermal sleeve entries replaces the AMR line item showing no aging effects with the following for both Treated Water (Inside) and Treated Water (Outside) environments. Note that Auxiliary Nozzle Thermal Sleeve should be changed to Auxiliary <u>Feedwater</u> Nozzle Thermal Sleeve, also, use Plant-Specific Note 121 instead of 120 for the Auxiliary Feedwater Nozzle Thermal Sleeve.
<table>
<tr><td>Loss of Material due to Crevice Corrosion
Loss of Material due to Pitting Corrosion</td><td>Water Chemistry and One-Time Inspection</td><td>VIII.D1-4 (SP-16)</td><td></td><td>F, 120</td></tr>
<tr><td>Cracking due to SCC</td><td>Water Chemistry and One-Time Inspection</td><td>VIII.D1-5 (SP-17)</td><td></td><td>F, 120</td></tr>
</table>
</td></tr>
</table>

<table>
<tr><td>Audit Question 3.2.2-01-SA-02</td><td>Amend LRA Table 3.2.2-1 on Page 3.2-28, for Piping, piping components, piping elements, in environment Air/Gas (Wetted Inside), to include the Water Chemistry Program as well as the One-Time Inspection Program to manage loss of material.

Also, in LRA Subsection B.2.18, for the One-Time Inspection Program, revise the table in the Program Description on Page B-58 to add Containment Spray to the Building Structure/System column in the first line of the table associated with the verifying the effectiveness of the Water Chemistry Program. In addition, remove "Piping Components" from the line item for Piping Components and Refueling Water Storage Tank on Page B-59.</td></tr>
<tr><td>Audit Question 3.3.1-13-MK-01</td><td>Revise LRA Table 3.5.2-17, on Page 3.5-139, for the Spent Fuel Storage Rack Boral material to change the Aging Effect Requiring Management and the AMP entries to "None." Change the Table 3.3.1, Item 3.3.1-13 Discussion to read:

Reduction of neutron-absorbing capacity and loss of material due to general corrosion do not require aging management. Further evaluation is documented in Subsection 3.3.2.2.6.

Also, revise LRA Subsection 3.3.2.2.6 to state:

The AMR evaluation determined that there has been no adverse operating experience at HNP for reduction of neutron absorbing capacity and loss of material due to general corrosion for Boral used in the spent fuel storage racks exposed to treated water or treated borated water. Both Virgil C. Summer Nuclear Plant and Brunswick Steam Electric Plant have been evaluated for these aging effects by the NRC staff. The Safety Evaluation Reports for License Renewal (NUREG-1787, for Summer, and NUREG-1856, for Brunswick) determined the aging effect to be insignificant. Therefore, it is concluded that these aging effects for Boral do not require aging management.

In addition, change Note 570 on LRA Page 3.5-204 to read:

The HNP methodology determined that there are no aging effects for boral. There has been no HNP adverse operating experience recorded. Both the V. C. Summer Nuclear Plant and the Brunswick Steam Electric Plant have been evaluated by the staff for these aging effects, and the Safety Evaluation Reports for License Renewal for those plants have determined the aging effects to be insignificant.</td></tr>
<tr><td>Audit Question 3.3.1-34-CM-01</td><td>The addition of a new Plant-Specific Program for Oil-Filled Cable Testing requires a revision to the discussion of AMPs in LRA Subsection B.1.1 and LRA Table B-1. The changes are shown on Page 3 of 3 of <u>Enclosure 2, Attachment 2</u>.</td></tr>
<tr><td>Audit Question 3.3.1-70-MK-01</td><td>Revise LRA Table 3.3.2-27 for Spray Nozzles and Sprinkler Heads on Pages 3.3-287 and 3.3-289 as follows:

<u>Spray Nozzles</u>
Delete the AMR line items for Air-Indoor (Outside) and add the following line item applicable to Air-Outdoor (Inside) affecting Intended Function M-8:<table><tr><td>Flow Blockage due to Fouling</td><td>Fire Water System</td><td></td><td></td><td>J, 738</td></tr></table>
<u>Sprinkler Heads</u>
Replace the AMR line item for the M-8 function in Air-Indoor (Outside) associated with Table 1 Item 3.4.1-41 with the following line item applicable to Air-Indoor (Outside):<table><tr><td>Flow Blockage due to Fouling</td><td>External Surfaces Monitoring</td><td></td><td></td><td>J, 738</td></tr></table>
(continued)</td></tr>
</table>

Audit Question 3.3.1-70-MK-01 (continued)	Add new Plant-Specific Note 738 to read: This aging effect was identified while benchmarking activities with other Utilities (Industry OE) regarding the effect of insect nesting on spray pattern, e.g., wasps building nests in spray nozzles. Sprinkler heads are subject to external debris that could affect the spray coverage area. As part of system walkdown activities, sprinkler heads are visually inspected to be clear of debris or interference with coverage area. Also, add new enhancement (3) to LRA Subsection A.1.1.15 to read: 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. In addition, add the following to the **Enhancements** section of LRA Subsection B.2.15: Program Elements Affected • **Preventive Actions** Revise the Program for in-scope spray nozzles, to 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. This change impacts License Renewal Commitment #11.
Audit Question 3.3.1-77-MK-01	Revise LRA Table 3.3.2-2, on Page 3.3-134, to change the AMP for the Raw Water (Inside) line item from the One-Time Inspection Program to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program and include the latter program in the list of AMPs in Subsection 3.3.2.1.2. Also, remove Plant-Specific Note 369 from this line item. In addition, in the Discussion column for LRA Table 3.3.1, line item 3.3.1-77, change the last sentence of the second paragraph to read: The aging effect is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.
Audit Question 3.3.1-89-MK-02	Revise LRA Table 3.3.2-14, on Page 3.3-201, to add the environment "Air – Indoor (outside)" to the line items for carbon steel piping, piping components, piping elements that align with Table 1 Items 3.3.1-59 (External Surfaces Monitoring Program) and 3.3.1-89 (Boric Acid Corrosion Program).
Audit Question 3.3.1-90-MK-01	Revise the LRA to delete the AMR line item for Cracking due to SCC from the AMR results for Piping, piping components, and piping elements on Table 3.3.2-49 on Page 3.3-352.
Audit Question 3.3.2-01-MK-02	Revise the LRA Subsection B.2.24 Program Description to add the statement: Inspections of elastomeric components will include physical manipulation to detect aging effects in addition to visual inspection.
Audit Question 3.4.2-2-SA-01	Revise LRA Table 3.4.2-2, on Page 3.4-38, to change the AMP for Piping, piping components, and piping elements in Treated Water (Inside) from the One-Time Inspection Program to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Also, add the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program in LRA Subsection 3.4.2.1.2. In addition, revise Plant-Specific Note 413 to read: Item represents piping components that are water-filled. The water source is from treated water.

<table>
<tr><td>Audit Question
3.4.2-12-SA-01</td><td>Revise LRA Table 3.4.2-12, on Page 3.4-74, to replace the AMR line item for Stainless Steel Piping, piping components, and piping elements in Treated Water (Inside) and the One-Time Inspection Program with the following three AMR line items.

<table>
<tr><td>Loss of Material due to Crevice Corrosion
Loss of Material due to Pitting Corrosion</td><td>Inspection of Internal Surfaces in Miscel-laneous Piping and Ducting Components</td><td>VIII.D1-4 (SP-16)</td><td></td><td>J, 419</td></tr>
<tr><td>Loss of Material due to Crevice Corrosion
Loss of Material due to Pitting Corrosion</td><td>Water Chemistry and One-Time Inspection</td><td>VIII.D1-4 (SP-16)</td><td>3.4.1-16</td><td>C, 420</td></tr>
<tr><td>Cracking due to SCC</td><td>Water Chemistry and One-Time Inspection</td><td>VIII.D1-5 (SP-17)</td><td>3.4.1-14</td><td>C, 420</td></tr>
</table>

In LRA Subsection B.2.18, for the One-Time Inspection Program, revise the table in the Program Description on Page B-58 to add Steam Generator Wet Lay Up to the Building Structure/System column in the first line of the table associated with verifying the effectiveness of the Water Chemistry Program.

Also, add the Water Chemistry Program to the AMPs in LRA Subsection 3.4.2.1.12. In addition, add new Plant-Specific Notes 419 and 420 to read:

419. Item represents piping components located in the Reactor Auxiliary Building that are water-filled but not used on a regular basis. The water source is from treated water.

420. Item represents piping components that are water-filled and associated with the Steam Generators. The water source is from treated water.

Finally, add the Steam Generator Wet Lay Up System immediately after the Radwaste Sampling System in the list of systems in the discussion column of Table 3.4.1, line item 3.4.1-16, and in Subsection 3.4.2.2.7.1.</td></tr>
<tr><td>Audit Question
3.4.2-12-SA-02</td><td>Revise LRA Table 3.4.2-12, on Page 3.4-73, to change the AMP for Piping, piping components, and piping elements in Raw Water (Inside) from the One-Time Inspection Program to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. In addition, revise Plant-Specific Note 412 to read:

Commodity and environment represent a sample cooler with cooling water supplied with service water. The item represents a piping component that is water-filled but not used on a regular basis.</td></tr>
<tr><td>Audit Question
3.4.2-12-SA-03</td><td>Revise LRA Table 3.4.2-12, on Page 3.4-73, to change the AMP for Piping, piping components, and piping elements in Treated Water (Inside) from the One-Time Inspection Program to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. In addition, revise Plant-Specific Note 414 to read:

Item represents piping components that are water-filled but not used on a regular basis. The water source is from treated water.</td></tr>
<tr><td>Audit Question
3.5.2-2-JW-01
3.5.2-2-JW-02
3.5.2-26-JW-03
and
3.5.2-26-JW-04</td><td>Revise LRA Note 572 on Page 3.5-204 to state:

The components "Phase Bus Enclosure Assemblies" are aligned with III.B2-7 or III.B3-2 or III.B3-5 because they have the same material, environment, aging effect and aging management program although they are not the same NUREG-1801 component "Support members; welds; bolted connections; support anchorage to building structure."</td></tr>
</table>

Audit Question 3.5.2-10-JW-01 3.5.2-17-JW-02 3.5.2-26-JW-02 and 3.5.2-27-JW-01	Amend the LRA to remove Plant-Specific Note 537 from the line item for Concrete: Exterior below grade in a Soil environment from the following AMR tables: Table 3.5.2-10 on Page 3.5-102, Table 3.5.2-17 on Page 3.5-131, and Table 3.5.2-26 on Page 3.5-173. In addition, amend the LRA to remove Plant-Specific Note 537 from the line item for Concrete: Foundation in a soil environment in Table 3.5.2-27 on Page 3.5-181.
Audit Question 3.5.2-17-JW-01	Amend the line item for Canal and Pool Gates in an Air-Indoor environment in LRA Table 3.5.2-17 on Page 3.5-129 to delete Plant-Specific Note 545 and add Plant-Specific Note 540. Also, revise Plant-Specific Note 540 to read: The components "Steel Components: All structural steel," "Steel Components: Fuel Pool Liner," "Floor Drains," "Sump Screens" or "Canal and Pool Gates" are aligned with III.B5-5 and/or III.B5-6 as "Miscellaneous Structures" because they have the same material, environment, aging effect and aging management program although they are not the same NUREG-1801 component "Support members; welds; bolted connections; support anchorage to building structure."
Audit Question 3.5.2-17-JW-03	Amend the AMR line item for Concrete: Interior in Table 3.5.2-17 on Page 3.5-132, where Table 1 item 3.5.1-33 and Notes I and 502 are referenced, to change the aging effect requiring management and AMP columns to "None."
Audit Question 3.5.2-17-JW-04	Revise Table 3.5.2-17 on Page 3.5-134 of the LRA for component Fire Hose Stations in a Borated Water Leakage environment, to change Plant-Specific Note 544 to 539. Also, revise Table 3.5.2-2 for Steel Components: All structural steel in a Borated Water Leakage environment to change the Notes column from "A" to "C, 539" to be consistent throughout the LRA. In addition, revise Plant-Specific Note 539 by deleting the last sentence in the Note.
Audit Question 3.5.2-17-JW-05	Revise LRA Table 3.5.2-17 on Page 3.5-136 for the component New Fuel Storage Rack, to replace NUREG-1801 Volume 2 Item VII.A1-1 (A-94) with III.B5-5 (TP-5). In addition, amend Plant-Specific Note 545 to read: NUREG-1801 assumes New Fuel Storage Racks are carbon steel, in an Air-Indoor environment, with aging effects (NUREG-1801, Item VII.A1-1). However, the HNP New Fuel Storage Racks are stainless steel. Stainless steel in an Air-Indoor environment has no aging effects. The New Fuel Storage Racks are aligned with NUREG-1801, Item III.B5-5 (TP-5) because the New Fuel Storage Racks have the same material, environment, aging effect (none), and aging management program (none) although they are not the same NUREG-1801 component "Support members; welds; bolted connections; support anchorage to building structure."
Audit Question 3.5.2-17-JW-06 3.5.2-27-JW-04	Amend Plant-Specific Note 553 on LRA Page 3.5-202 to read: The AMR methodology concluded that elastomers in this environment are susceptible to the aging effect of cracking and change of material properties due to temperature. However, the elastomers are in the Group 3 structures, the Group 5 structure, or the Group 8 structures, rather than a Group 6 structure (III.A6-12). Cracking and change in material properties for elastomers results in loss of sealing and is considered an equivalent aging effect.
Audit Question 3.5.2-21-JW-01	Amend the LRA to add Plant-Specific Note 530 to the aging effect of "Cracking" associated with Table 1, Item 3.5.1-28, in the following locations: 1) Table 3.5.2-21 for Concrete: Exterior Below Grade and Concrete Foundation on Pages 3.5.2-148 and 3.5.2-149, respectively, and 2) Table 3.5.2-24 for Concrete Foundation on Page 3.5-164.
Audit Question 3.5.2-26-JW-01	Amend LRA Table 3.5.2-26 on Page 3.5-173 for Concrete: Exterior Above Grade to delete the AMR line associated with cracking and Table 1 item 3.5.1-32.

<table>
<tr><td>Audit Question
3.5.2-27-JW-02
3.5.2-27-JW-03</td><td>Amend LRA Table 3.5.2-27 on Page 3.5-182 for Concrete Roof Slab to revise the first two AMR rows. The first row should be revised to incorporate III.A8-5 (T-01) and 3.5.1-26, and the second row should be revised to incorporate III.A8-1 (T-03) as follows:
<table>
<tr><td>Loss of Material
Cracking</td><td>Structures Monitoring</td><td>III.A8-5
(T-01)</td><td>3.5.1-26</td><td>A</td></tr>
<tr><td>Cracking</td><td>Structures Monitoring</td><td>III.A8-1
(T-03)</td><td>3.5.1-27</td><td>A, 504</td></tr>
</table></td></tr>
<tr><td>Audit Question
3.5.2-28-JW-01</td><td>Amend LRA Table 3.5.2-28 on Page 3.5-191 for Seismic Joint Filler to remove NUREG-1801 Volume 2 Item VII.G-2 (A-20).</td></tr>
<tr><td>Audit Question
3.5.2-29-JW-01</td><td>Amend LRA Table 3.5.2-29 on Page 3.5-192 for Cable Tray, Conduit, etc., to replace the Structures Monitoring Program with None for carbon and stainless steel in a soil environment.

In addition, revise Plant-Specific Note 555 on LRA Page 3.5-202 to read:

Buried conduits that connect to Class I manholes have a designed water tight clamping arrangement. The clamping arrangement provided includes a carbon steel support structure, stainless steel clamps, and a neoprene boot. The purpose of the clamping arrangement is to prevent water intrusion into the manholes. Due to the inaccessible location of the arrangement and potential damage to the safety related cable, no direct visual inspection is planned for the buried clamping arrangement in the soil. However, degradation of the clamping arrangement leading to water intrusion into the manholes can be determined from inspections performed from inside the manhole. The interior of the manholes (included with commodity, Concrete: Interior) will continue to be inspected by the Structures Monitoring Program for water intrusion, including the area where the buried conduits connect to the Class I manholes.

The HNP AMR methodology concluded that the neoprene boot material has no aging effect in soil and that carbon steel and stainless steel in soil have the aging effect of loss of material.</td></tr>
<tr><td>Audit Question
3.5.2-29-JW-02</td><td>Amend LRA Table 3.5.2-29 on Page 3.5-197 for Siding, to replace Note A with Note C. In addition, revise Plant-Specific Note 544 on LRA Page 3.5-201 to include "Siding," as well as, "Non-fire Doors," "Floor Drains," and "Fire Hose Stations."</td></tr>
<tr><td>Audit Question
3.5.2-29-JW-03</td><td>Revise LRA Table 3.5.2-29 on Page 3.5-196 of the LRA for component Pipe, material Reinforced Concrete, in a Soil environment to delete the Buried Piping and Tanks Inspection Program and add the Structures Monitoring Program. Delete the Buried Piping and Tanks Inspection Program from the list of AMPs in Subsection 3.5.2.1.29. In addition, revise Note 547 on Page 3.5-202 of the LRA as follows:

The reinforced concrete pipe and asbestos cement manifold header are mechanical components utilizing civil materials which do not align with NUREG-1801. The HNP AMR methodology concluded that reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments and reinforced concrete pipe in a Soil environment have the same aging effects as structural concrete. The Structures Monitoring Program was selected to manage the aging effects of reinforced concrete pipe in a Soil environment and Mechanical aging management programs were selected to manage the aging effects of reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments.</td></tr>
<tr><td>Progress Energy
Self-Identified
Change 1</td><td>Revise system intended function 1 for the Radwaste Sampling System in Subsection 2.3.3.48, on Page 2.3-157, to read:

Components that are safety related and are relied upon to remain functional during and following design basis events.</td></tr>
</table>

Progress Energy Self-Identified Change 2	Revise the annotation at the top of Page 3.3-440 from: "Notes for Tables 3.3.2-1 through 3.3.2-68" to "Notes for Tables 3.3.2-1 through 3.3.2-71."
Progress Energy Self-Identified Change 3	Revise the annotation at the top of Page 3.5-198 from: "Notes for Tables 3.5.2-1 through 3.5.2-26" to "Notes for Tables 3.5.2-1 through 3.5.2-29."
Progress Energy Self-Identified Change 4	The projected gamma dose to the inside face of the Primary Shield Wall for 60 years has been determined to be 1.29 E+10 Rads through a refined calculation completed after the submittal of the LRA. Therefore, the following LRA changes are required: Revise Plant-Specific Note 535 to read: The HNP AMR methodology concluded that there are no aging effects requiring management to the primary shield wall inside face due to irradiation. The primary shield wall inside face is subject to 1.29 E+10 Rads over 60 years. On Page 3.5-67 of the LRA, revise the line item associeated with Concrete: Containment Internal and Note 535 to insert "None" in the columns for Aging Effect Requiring Management and Aging Management Program.
Progress Energy Self-Identified Change 5	In Table 4.1-2, delete the repeated phrase in the fifth row from the bottom. The corrected statement reads: Ductility reduction of fracture toughness for the reactor vessel internals.
Progress Energy Self-Identified Change 6	Revise the Program Description in LRA Subsection B.2.19 to include copper alloys with aluminum content greater than 8%, by replacing the first sentence with the following: The Selective Leaching of Materials Program ensures the integrity of components and/or commodities (such as piping, pump casings, valve bodies and heat exchanger components) made of copper alloys with zinc content greater than 15% or aluminum content greater than 8% and gray cast iron exposed to a raw water, treated water, lubricating oil or hydraulic fluid, fuel oil, wetted air/gas, or soil environment at HNP. Also, revise the Operating Experience discussion in LRA Subsection B.2.19 to acknowledge two instances of occurrence of selective leaching at HNP. This operating experience discussion has been incorporated into the discussion of B.2.19 program elements in Attachment 3 to this enclosure, (i.e., Enclosure 2, Attachment 3).
Progress Energy Self-Identified Change 7	In the fifth paragraph of the **Operating Experience** section in LRA Subsection B.2.2 , change "Nonconformance Reports" to "Nuclear Condition Reports" and change the last sentence to: However, trending data for these contaminants indicate that for the vast majority of the time the levels were well below the threshold for age related degradation.
Progress Energy Self-Identified Change 8	Change Auxiliary Nozzle Thermal Sleeve to read Auxiliary Feedwater Nozzle Thermal Sleeve in LRA Table 2.3.1-6, Page 2.3-25, and on Table 3.2.1-6, Page 3.1-137.
Progress Energy Self-Identified Change 9	Change the third sentence in LRA Subsection 2.5.4.8 to delete the word "bolted." The revised sentence will read: Connections are commonly made with welds or mechanical type connectors, which include compression and wedge-type devices.

Enclosure 2, Attachment 1

A.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.

B.2.37 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Program Description

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is credited for 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 cable 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 aging effect/mechanism of concern is:

- Loosening of Cable Connections

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.

NUREG-1801 Consistency

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program consistent with NUREG-1801, Section XI.E6, with exception.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Scope of Program**
  NUREG-1801, Rev. 1, AMP XI.E6 states that connections associated with cables in scope of license renewal are part of this program, regardless of their association with active or passive components. However, HNP has applied the clarification provided in NRC letter to NEI, dated March 16, 2007, that

revises the scope to include only external cable connections terminating at an active device such as motor, motor control center, switchgear, or of a passive device such as a fuse cabinet. Wiring connections internal to an active assembly installed by manufacturers are considered a part of the active assembly; and, therefore, are not within the scope of this program.

- **Detection of Aging Effects**
  NUREG-1801, Rev. 1, AMP XI.E6 specifies periodic testing of connections using thermography, contact resistance testing, or other appropriate testing methods. However, consistent with the test frequency flexibility provided in the NRC letter to NEI, dated March 16, 2007, this element 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. Inspection methods may include thermography, contact resistance testing, bridge balance testing, or other appropriate testing methods. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Enhancements

None.

Discussion of Program Elements

Scope of Program

The scope of this program includes only external cable connections terminating at an active device such as motor, motor control center, switchgear or of a passive device such as a fuse cabinet. Wiring connections internal to an active assembly installed by manufacturers are considered a part of the active assembly and therefore are not within the scope of this program.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The program will focus on the metallic portions of non-EQ cable connections exposed to inside and outside air environments. The monitoring includes loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation. A representative sample of electrical cable connections will be tested and will consider the following factors: application (high, medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selected will be provided.

Detection of Aging Effects

This program will be implemented as a one-time inspection on a representative sample of non-EQ cable connections within the scope of license renewal prior to the period of extended operation. Inspection methods may include thermography, contact resistance testing, bridge balance testing, or other appropriate testing methods. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Monitoring and Trending

Trending actions are not part of the program because this is a one-time inspection program. However, trending of discrepancies is performed as required in accordance with the Corrective Action Program.

Acceptance Criteria

The acceptance criteria for each test are defined by the specific type of test performed and the specific type of cable connections tested.

Corrective Actions

An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the cable connections can be maintained. Such an evaluation will consider the operability of the component, the potential root causes for not meeting the test acceptance criteria, the corrective action necessary, and possible changes to the one-time inspection program such as increased frequency and sample size. When an unacceptable condition or situation is identified, a determination is made on whether the same condition or situation is applicable to other in-scope cable connections not tested. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program. Industry operating experience has shown that loosening of connections and corrosion of connections are aging mechanisms that could lead to a loss of electrical continuity without proper installation and maintenance activities. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide reasonable assurance that electrical connections within the scope of License Renewal will be maintained consistent with the CLB through the period of extended operation.

Enclosure 2, Attachment 2

A.1.1.40 Oil-Filled Cable Testing Program

The Oil-Filled Cable Testing Program is credited for aging management of the high-voltage, oil-filled cables which connect the HNP 230KV Switchyard to the Startup Transformers. Periodic cable testing will be performed at least once every four years to provide an indication of the condition of the cable insulation properties. 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, such as power factor (Doble), partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed. The program provides assurance that a loss of dielectric strength is an aging effect that will be managed such that the oil filled cables will perform their intended function for the period of extended operation.

B.2.38 OIL-FILLED CABLE TESTING PROGRAM

Program Description

The Oil-Filled Cable Testing Program is credited for aging management of the high-voltage, oil-filled cables which connect the HNP 230KV Switchyard to the Startup Transformers. Periodic cable testing will be performed at least once every four years to provide an indication of the condition of the cable insulation properties. 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, such as power factor (Doble), partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed. The program will verify that the effects of aging from a loss of dielectric strength caused by thermal/thermoxidative degradation of organics, radiation-induced oxidation (radiolysis) of organics, voltage (partial discharge), moisture, or the presence of other impurities will be managed during the period of extended operation.

Aging Management Program Elements

The results of an evaluation of the aging management activities for the Oil-Filled Cable Testing Program against the ten elements described in Appendix A of NUREG-1800 is provided below.

- **Scope of Program**
  This program applies to the high-voltage, oil-filled cables which connect the HNP 230KV Switchyard to the Startup Transformers.

- **Preventive Actions**
  This is a condition monitoring program; therefore, no actions are taken as part of this program to prevent or mitigate aging degradation.

- **Parameters Monitored/Inspected**
  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 such as power factor (Doble), partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed.

  The parameters monitored include a loss of dielectric strength caused by thermal/ thermoxidative degradation of organics, radiation-induced oxidation (radiolysis) of organics, voltage (partial discharge), moisture, or the presence of other impurities.

- **Detection of Aging Effects**
  The high-voltage, oil-filled cables within the scope of this program are to be tested at least once every four years. This is an adequate period to detect aging effects before a loss of component intended function since experience has shown that aging degradation is a slow process. A four-year testing interval will provide

multiple data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed prior to the period of extended operation.

- **Monitoring and Trending**
  Trending actions are not part of this program. However, trending of discrepancies is performed (as required) in accordance with the HNP Corrective Action Program. The Corrective Action Program is implemented by the HNP QA Program in accordance with 10 CFR 50, Appendix B.

- **Acceptance Criteria**
  The acceptance criteria will be determined based on the test selected for this program.

- **Corrective Actions**
  Corrective actions as required will be implemented through the HNP Corrective Action Program and may include, but are not limited to, additional cable testing, testing the insulating oil in the cable system, or repair or replacement of the affected cable or connection. The Corrective Action Program is implemented by the HNP QA Program in accordance with 10 CFR 50, Appendix B.

- **Confirmation Process**
  This attribute is discussed in Subsection B.1.3.

- **Administrative Controls**
  This attribute is discussed in Subsection B.1.3.

- **Operating Experience**
  The program is a new program with no site-specific operating experience (OE) history. Plant-specific and industry-wide OE was considered in the development of this program. This review confirms the reliability of high-voltage, oil-filled cables in underground applications. However, periodic cable testing will be performed to assure that the effects of aging will be managed during the period of extended operation. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that this program will continue to be effective in the management of aging effects.

Conclusion

The Oil-Filled Cable Testing Program will be effective for managing aging effects since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls. The Oil-Filled Cable Testing Program will provide reasonable assurance that the applicable high-voltage, oil-filled cables and connections will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

Revisions to LRA Table 3.6.2-1 on Page 3.6-19:

Component Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	AMP	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
High-Voltage Power Cables	Electrical continuity E-1	Oil-impregnated paper (insulation) Lead (sheath) Organic polymers (outer jacket)	Air-Outdoor	Loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure	Oil-Filled Cable Testing Program			J, 602

Plant-specific Notes:

602 HNP cables are high-voltage, oil-filled, paper insulated, lead-sheathed cables. Lead sheath cables are designed for submergence for extended periods. The impregnation of the paper tape improves the insulation's electrical resistance and provides an extra layer of defense against moisture ingress. The highly refined oil used for the insulating medium also serves to dissipate heat from the conductors and cools the cable when operating under load. HNP cables have an Okolene (black polyethylene) outer jacket. Site and industry operating experience has shown this design to be extremely reliable in underground applications. Periodic cable testing will be performed to assure that the effects of aging will be managed during the period of extended operation.

In addition, add the Oil-Filled Cable Testing Program to the list of AMPs in LRA Subsection 3.6.2.1.1.

Revisions to LRA Subsection B.1.1 on Page B-4:

Delete the last two sentences of Subsection B.1.1.

Revision to LRA Table B-1 on Page B-11:

Add the following rows to the table.

Plant-Specific			
NA	Plant-specific Program	Oil-Filled Cable Testing Program See Subsection B.2.38	Not applicable

Enclosure 2, Attachment 3

There are 11 new programs; ten are addressed in the following LRA Subsections. B.2.37 is also a new program but is discussed in Enclosure 2, Attachment 1 above. For the remaining new programs, amend the LRA by replacing the **Operating Experience** discussion in each of the indicated LRA Subsections with the following discussion of program elements.

B.2.6 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

Discussion of Program Elements

Scope of Program

This aging management program (AMP) will manage loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement in CASS reactor vessel internals components within the scope of License Renewal. The program will consider potential synergistic effects from thermal aging and neutron embrittlement and will be implemented as a supplemental examination (augmented inspection) as part of an existing ASME Code 10-year inspection interval Inservice Inspection plan.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The components within the scope of this AMP have been evaluated and determined to be potentially susceptible to thermal aging and neutron irradiation embrittlement based on materials of construction and neutron fluence values.

Detection of Aging Effects

The program allows for a component-specific evaluation, including a mechanical loading assessment to determine if the loading is compressive or low enough to preclude fracture. The component inspection techniques have not been determined at this time; however, they will be selected based on the recommendations of NUREG-1801, Section XI.M13.

Monitoring and Trending

Supplemental inspections performed under the program will meet the requirements of IWB-2400 using reliable methods.

Acceptance Criteria

Flaw detection criteria for the CASS components will be in accordance with the recommendations of NUREG-1801, Section XI.M13.

Corrective Actions

The Corrective Action program element is addressed generically in Subsection B.1.3.

Currently, HNP uses the 1989 Edition of ASME Section XI (the Code) for determining repair and replacement requirements. Repairs are performed in accordance with Article IWA-4000 and the corresponding IWX-4000 of the IWB/C/D portions of the Code. Replacements are performed in accordance with Article IWA-7000 and the corresponding IWX-7000 of the IWB/C/D portions of the code.

HNP is currently updating the ISI Program to the ASME Section XI, 2001 Edition with addenda through 2003, per 10 CFR 50.55a. In this edition/addenda of the Code, all Repair/Replacement Activities are controlled under Article IWA-4000. Section XI states, "The term repair/replacement activity includes those activities previously known as repair, replacements, modification, or alteration." All of the IWX-4000 and IWX-7000 articles of Section XI have been incorporated into IWA-4000.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new AMP for thermal aging and neutron irradiation embrittlement of CASS. The recommendations of NUREG-1801, Section XI.M13, were developed using research data obtained on both laboratory-aged and service-aged materials. The program will be updated periodically to incorporate industry knowledge. NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.18 ONE-TIME INSPECTION PROGRAM

Discussion of Program Elements

Scope of Program

This AMP is used to determine the need for additional aging management for structures and components that are managed by other programs. This is done by confirmation that either 1) the aging effect is not occurring, or 2) the aging effect is occurring very slowly so as not to affect the intended function of the structures and components during the period of extended operation. The program includes the one-time inspections specified in NUREG-1801 as well as inspections identified during plant-specific aging management reviews. One-time inspections of small-bore Class 1 piping is addressed in a separate AMP.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

Program activities will be directed towards aging effects/mechanisms that were identified during aging management reviews. Each structure or component requiring a one-time inspection is evaluated to determine: sample size (refer to the program element Detection of Aging Effects), inspection locations, and examination techniques. Typically the method of inspection would include:

Aging Mechanism	Typical Inspection Method
Crevice/Pitting Corrosion	Visual (e.g., VT-1) and/or volumetric
General/Galvanic/MIC	Visual (e.g., VT-3) and/or volumetric
Erosion	Visual (e.g., VT-3) and/or volumetric
Fouling	Visual (e.g., VT-3) and/or volumetric
Cracking	Enhanced Visual and/or volumetric

Detection of Aging Effects

Sample size would be based on considerations, such as, accessibility, leading or bounding locations, safety significance, severity of operating conditions, and design margins. Progress Energy non-destructive examination (NDE) procedures and personnel qualifications meet the requirements of the ASME Code, where applicable. Administrative controls and quality assurance requirements for NDE activities are implemented in accordance with 10 CFR 50, Appendix B. Inspections may be performed together with ASME inservice inspection activities, and they will be designed to ascertain if detrimental aging effects are occurring. In general, inspections will be scheduled to be accomplished no earlier than 10 years prior to the period of extended operation.

Monitoring and Trending

The One-Time Inspection Program is not intended to be a monitoring or trending program; should degradation be encountered, it would be evaluated, and if required, monitored or trended, under the Corrective Action Program.

Acceptance Criteria

Any indication or relevant conditions of degradation detected will be evaluated using defined acceptance criteria, such as minimum wall thickness for pressure boundary piping.

Corrective Actions

Unacceptable inspection findings will be evaluated in accordance with the Corrective Action Program. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The One-Time Inspection Program is a new program. The HNP aging management review process ensures that one-time inspections have been prescribed and developed with consideration of plant and industry operating experience.

NUREG-1801 is based on industry operating experience through January 2005. This program applies to potential aging effects for which there are currently no operating experience indicating the need for an aging management program. Nevertheless, the elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.19 SELECTIVE LEACHING OF MATERIALS PROGRAM

Discussion of Program Elements

Scope of Program

The program will perform examinations to determine whether loss of material due to selective leaching is occurring for those components determined to be potentially susceptible during aging management reviews.

Preventive Action

The program is an inspection/verification program; there is no preventive action.

Parameters Monitored/Inspected

HNP will establish a representative sample set of components to be managed following review of industry technology used to inspect and evaluate the identification and progression of the selective leaching aging mechanism. A qualitative determination of selective leaching will be used in lieu of Brinell hardness testing for components within the scope of this program. Follow-up of unacceptable findings will include engineering evaluation and sample expansion as appropriate. Inspection of components will be scheduled under the Work Management Process to occur just before the beginning of the license renewal period.

Detection of Aging Effects

The occurrence of selective leaching will be determined using an inspection procedure that includes detailed instructions for performing visual inspections and applying qualitative determinations of selective leaching.

Monitoring and Trending

Selective leaching will be managed by one-time inspections of a representative set of components. The program will not rely on monitoring and trending of results.

Acceptance Criteria

Evidence of selective leaching will result in an engineering evaluation and will be tracked as a corrective action. Engineering evaluations for unacceptable degradation will be addressed in accordance with the Corrective Action Program.

Corrective Actions

The Corrective Action Process provides the administrative controls for identifying, documenting, tracking, investigating, correcting, and trending significant conditions adverse to quality. The Corrective Action Program is discussed generically in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Selective Leaching of Materials Program is a new program; therefore, operating experience to verify the effectiveness of the program is not available. Two occurrences of selective leaching of materials were identified. One instance involved dezincification of a shear pin on a cable attached to the refueling cart, which failed under normal tension. A second instance occurred on a copper nickel tube in the main generator hydrogen cooler. The investigation revealed that denickelification occurred from under deposit corrosion, which presents as loss of material due to pitting, in copper-nickel heat exchanger tubes that are subjected to Harris Lake water. The report determined that using periodic eddy current testing is effective in detecting this aging effect. The Generator Hydrogen Cooler is not in scope of License Renewal. However, there are components requiring aging management (such as safety-related heat exchanger tubes) which have similar materials and are subjected to the same environment. These components are more appropriately managed by the Open-Cycle Cooling Water Program, which performs eddy current testing or inspections.

NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.20 BURIED PIPING AND TANKS INSPECTION PROGRAM

Discussion of Program Elements

Scope of Program

A new inspection procedure will be developed and will include guidance for condition monitoring of buried piping components/commodity groups that have been identified during aging management reviews.

Preventive Action

HNP has used preventive measures, such as, protective coatings or wrappings on buried steel and cast iron piping applications. The soil environment at HNP is non-aggressive.

Parameters Monitored/Inspected

Detailed procedural requirements for the Program will be developed and will include provisions to perform inspections of components following excavation and to take actions to assess coating damage and potential corrosion, and to employ appropriate care when burying piping components.

Detection of Aging Effects

Inspections will be performed at frequencies consistent with those in NUREG-1801, Rev. 1.

Monitoring and Trending

Site experience with corrosion of system components will be used to plan additional inspections.

Acceptance Criteria

The new procedure governing inspection of buried components will ensure that a coating engineer or other qualified individual assists in the evaluation of coating degradation and will include inspection checklists that will include requirements to document the results of inspections. The extent of degradation found during inspection will determine the nature of the corrective actions taken and the need for and locations of future inspections.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Buried Piping and Tanks Inspection Program is a new program applicable to buried piping at HNP. There is no existing operating experience to validate the effectiveness of this program. At HNP, leaks have occurred in the Fire Protection System and the Potable and Sanitary Water System. These were evaluated and determined not to be caused by age-related degradation. A leak occurred in a buried portion of the Diesel Generator Fuel Oil Storage and Transfer System; however, due to the inaccessible location of the leak, its cause could not be determined. These leaks were evaluated in accordance with the Corrective Action Program. Based on this site experience, it can be concluded that leaks in HNP buried piping have been detected and that appropriate corrective actions have been taken to ensure no loss of component intended function. This experience is not atypical and justifies the use of the 10-year inspection frequency for buried components endorsed by NUREG-1801.

NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.21 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING PROGRAM

Discussion of Program Elements

Scope of Program

This program is a new program that will be administratively controlled as an augmented ASME Code inservice inspection program. The program will detect cracking in small-bore piping regardless of the aging mechanism causing the cracking and will consider the guidance in EPRI Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)."

Preventive Action

The program is an inspection/verification program; there is no preventive action.

Parameters Monitored/Inspected

The program is intended to detect cracking in small-bore, ASME Class 1 piping less than NPS 4.

Detection of Aging Effects

HNP has not experienced cracking of small-bore piping due to stress corrosion or thermal/mechanical loading.

Monitoring and Trending

Inspections will be performed at a sufficient number of locations to assure an adequate sample. The sample population for the plant-specific program will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore locations. The sample prioritization will consider the potential for mechanical loading as a result of thermal stratification, piping potentially susceptible to IGSCC (i.e., normally stagnant piping), and locations identified for inspection under the RI-ISI program which considers thermal loading from plant cycles and thermal stratification.

Acceptance Criteria

Inspection results will be evaluated in accordance with ASME Code Subsection IWB-3400 and IWB-3131. Additional examinations, if required, will be performed in accordance with IWB-2430.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The industry has experienced unexpected cracking which may have been caused by aging mechanisms not originally considered in the design analyses. Additional events caused by fatigue due to thermal stratification resulted in the issuance of IE Bulletin 88-08 as supplemented. In response to IE Bulletin 88-08, an augmented inspection program was implemented during the first HNP inspection interval. Augmented volumetric examinations were performed on selected piping welds in the charging and safety injection lines considered potentially subject to thermal fatigue. No indications of cracking were found.

This is a new aging management program for the One-Time Inspection of ASME Code Class 1 Small-Bore Piping. There is no existing operating experience to validate the effectiveness of this program. Any future operating experience which may impact the program will be captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.24 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS PROGRAM

Discussion of Program Elements

Scope of Program

The program does not inspect components within the scope of License Renewal that have aging effects associated with boric acid corrosion; however, should borated water leakage be identified during inspections under this program, they would be reported and resolved under the Boric Acid Corrosion Program.

Preventive Action

The program is an inspection program; there is no preventive action.

Parameters Monitored/Inspected

Parameters monitored are those that may be detected by visual inspections, such as, cleanliness and freedom from obstructions for flame arrestors.

Detection of Aging Effects

Inspection intervals will be selected to assure detection of degradation prior to loss of intended function using existing administrative controls for scheduling preventive maintenance and surveillance, trending performance of systems and components, and equipment reliability guidance. Identification of paint or coating degradation, such as, blistering, cracking, or the presence of rust or mechanical damage, is an effective method for detection of the occurrence of corrosion on a painted steel surface. Locations of inspections will be selected to be representative of the material/ environment combinations evaluated in the aging management review for the components/commodities. Selection of locations will consider such factors as access, equipment design, radiation levels, and safety of inspectors.

Monitoring and Trending

Administrative controls for the program will ensure: (1) that qualified personnel will perform inspections under work order tasks, (2) that System Engineers confirm the effectiveness of inspection activities by monitoring, reporting, and controlling trending activities, and (3) that System Engineers periodically review the plant-specific operating history and industry operating experience applicable to the system.

Acceptance Criteria

System maintenance history will be periodically reviewed by System Engineers. Material failures or degraded material conditions will require initiation of a report and further evaluation in accordance with the Corrective Action Program.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program. However, operating experience has been incorporated into License Renewal program activities, and also NUREG-1801 is based on industry operating experience through January 2005. The HNP aging management review process ensures that the program has been prescribed and will be developed with consideration of plant and industry operating experience.

Periodic surveillance and preventive maintenance activities have been in place at HNP since the plant began operation. These activities have proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. There is a demonstrated history of detecting damaged and degraded components and causing their repair or replacement in accordance with the site Corrective Action Program.

B.2.33 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Discussion of Program Elements

Scope of Program

This program includes insulated cables and connections installed in an adverse, localized environment caused by heat or radiation in the presence of oxygen and purposely includes other plant areas for conservatism.

Preventive Action

The program is an inspection program; there is no preventive action.

Parameters Monitored/Inspected

The AMP is a representative sampling program. Accessible electrical cables and connections installed in adverse localized environments are visually inspected for cable and connection jacket surface anomalies.

Detection of Aging Effects

A representative sample of accessible insulated cables and connections installed in adverse localized environments will be inspected at least once every 10 years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking, swelling or surface contamination. The first inspection for license renewal will be completed before the period of extended operation. The HNP program utilizes plant operating experience (OE) to determine the plant areas to be inspected. HNP OE is used to identify past cable failures, cables that exhibited the effects of aging, hot spots, and adverse localized environments. Part of this OE review includes conversations with maintenance personnel and the use of environmental surveys. Based on this review of OE, the plant areas to be inspected become localized in nature, consisting of a limited area (or subset) of a much larger plant area or zone. The sample selection of cables and connections inspected within the limited plant area bound all cables and connections in the area since the inspection focuses on the worst case environments.

Monitoring and Trending

Trending of discrepancies is performed as required in accordance with the HNP Corrective Action Program.

Acceptance Criteria

The program requires that accessible cables and connections be free from unacceptable, visual indications of jacket surface anomalies, which suggest that conductor insulation or connection degradation exists.

Corrective Actions

Unacceptable visual indications of cable and connection jacket surface anomalies will be subject to an engineering evaluation. This evaluation is to consider the age and operating environment of the component, as well as the severity of the anomaly, and whether such an anomaly has previously been correlated to degradation of conductor insulation or connections. Corrective actions as required will be implemented through the Corrective Action Program and may include, but are not limited to, testing, shielding, or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination will be made as to whether this same condition or situation could be applicable to other accessible or inaccessible insulated cables and connections and whether the sample size should be expanded. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The program is a new program with no site-specific operating experience history. However, as noted in NUREG-1801, industry operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.34 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

Discussion of Program Elements

Scope of Program

This program applies to electrical cables and connections (i.e., cable system) used in circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation that are subject to aging management review.

Preventive Action

This is a surveillance testing program and no actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The parameters monitored are determined from the specific calibrations, surveillances, or testing performed and are based on the specific instrumentation circuit under surveillance or being calibrated as documented in HNP procedures.

Detection of Aging Effects

For radiation monitoring circuits, HNP will review the results or findings of calibration or surveillance tests in order to detect severe aging degradation prior to the loss of the cable and connection intended function. Calibration or surveillance results that fail to meet acceptance criteria will be reviewed for aging effects when the results are available. For nuclear instrumentation circuits, the cable system will be tested as an alternative to the review of calibration or surveillance results described above.

Monitoring and Trending

Trending actions are not part of this HNP AMP because the ability to trend test results is dependent on the specific type of test chosen. However, the trending of discrepancies is performed as required in accordance with the HNP Corrective Action Program.

Acceptance Criteria

Acceptance criteria will be set out in the applicable plant procedures.

Corrective Actions

Corrective actions will be implemented when calibration or surveillance results or findings of surveillances do not meet the acceptance criteria through the Corrective Action Program and will include an engineering evaluation to ensure that the intended functions of the electrical cable system can be maintained consistent with the CLB. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new program; there is no existing operating experience to validate the effectiveness of this program. However, as noted in NUREG-1801, industry operating experience has shown that exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced IR. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.35 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Discussion of Program Elements

Scope of Program

The program applies to inaccessible (i.e., in conduit or direct buried) medium voltage cables.

Preventive Action

Periodic actions are taken to prevent non-EQ medium voltage cables from being exposed to significant moisture. This activity includes inspecting for water collection in cable manholes and draining water as needed.

Parameters Monitored/Inspected

In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation.

Detection of Aging Effects

The first testing for license renewal will be completed before the period of extended operation. The manhole inspection frequency will be based on actual field data and shall not exceed two years. The first inspection for license renewal will be completed before the period of extended operation.

Monitoring and Trending

Trending actions are not part of the program. However, trending of discrepancies is performed as required in accordance with the Corrective Action Program.

Acceptance Criteria

The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

Corrective Actions

Engineering will perform an evaluation when acceptance criteria are not met in order to ensure that the License Renewal intended functions will be maintained. Such an evaluation will consider the significance of the test results, the operability of the component, the potential root causes, the corrective actions required, and a determination whether the condition is applicable to other in-scope medium-voltage cables. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new Aging Management Program for non-EQ inaccessible medium-voltage cables. There is no existing operating experience to validate the effectiveness of this program. As noted in NUREG-1801, industry operating experience has shown that cross linked polyethylene or high molecular weight polyethylene insulation materials are most susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage; for example, treeing is much less prevalent in 4KV cables than those operated at 13KV or 33KV. Also, minimizing exposure to moisture minimizes the potential for the development of water treeing.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.36 METAL ENCLOSED BUS PROGRAM

Discussion of Program Elements

Scope of Program

The program applies to Metal Enclosed Bus (MEB) within the scope of License Renewal.

Preventive Action

This is an inspection program; no actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The program will check a sample of accessible bolted connections for loose connection. Alternatively, bolted connections covered with heat shrink tape, sleeving, insulating boots, etc., may be visually inspected for insulation material surface anomalies.

Detection of Aging Effects

The program will check a sample of accessible bolted connections for loose connection by using thermography or by measuring connection resistance using a low range ohmmeter. Thermography will be performed while the bus is energized and loaded. Connection resistance measurements will be performed while the bus is de-energized and will consist of a sampling of accessible bolted connections. MEB internal surfaces will be visually inspected for aging degradation of insulating material and for foreign debris and excessive dust buildup, and evidence of moisture intrusion. Bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. Internal bus supports will be visually inspected for structural integrity and signs of cracks. This program will be completed before the period of extended operation and every 10 years thereafter.

As an alternative to thermography or measuring connection resistance of bolted connections, for the accessible bolted connections that are covered with heat shrink tape, sleeving, insulating boots, etc., HNP may use visual inspection of insulation material to detect surface anomalies, such as discoloration, cracking, chipping or surface contamination. If this alternative visual inspection is used to check bolted connections, the first inspection will be completed before the period of extended operation and every five years thereafter.

Monitoring and Trending

Trending actions are not part of the program, because the ability to trend inspection results is limited. However, trending of discrepancies is performed as required in accordance with the HNP Corrective Action Program.

Acceptance Criteria

Bolted connections are to be below the maximum allowed temperature for the application when thermography is used. If measuring connection resistance using a low range ohmmeter, microhm values of accessible bolted connections shall not exceed the high levels of the normal range indicated in the manufacturer's published data. If manufacturer's data is not available, the accessible bolted connection must be within 25% of the lowest value of the other sample connections. MEBs are to be free from unacceptable visual indications of surface anomalies, which suggest that conductor insulation degradation exists. In addition, no unacceptable indication of corrosion, cracks, foreign debris, excessive dust buildup or evidence of moisture intrusion is to exist. When the visual inspection alternative for bolted connections is used, the absence of discoloration, cracking, chipping, or surface contamination will provide positive indication that the bolted connections are not loose.

Corrective Actions

Corrective actions may include but are not limited to cleaning, drying, increased inspection frequency, replacement, or repair of the affected MEB components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible MEBs. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new Aging Management Program for MEB. There is no existing site specific operating experience to validate the effectiveness of this program. Industry experience has shown that failures have occurred on MEBs caused by cracked insulation and moisture or debris buildup internal to the MEB. Industry experience has also shown that MEB exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connections related to the repeated cycling of connected loads or of the ambient temperature environment. This phenomenon can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating or ambient heating) that are routinely cycled.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Enclosure 2, Attachment 4

Review of LRA Changes

This attachment includes: 1) changes to LRA Appendices A and B resulting from the responses to RAI-B.2.17 (changes are marked with margin lines); 2) changes to LRA Tables 4.2-2 and 4.2-3 resulting from the response to RAI-4.2.3; 3) changes to LRA Subsection 4.2.4 resulting from the response to RAI-4.2.4; 4) changes to LRA Subsection 4.2.5 resulting from the response to RAI-4.2.5; and 6) changes to LRA Subsection 4.2.1 resulting from the response to RAI-4.2.6.

LRA Appendix A and Appendix B Changes Resulting from the Response to RAI-B.2.17

Revise LRA Subsection A.1.1.17 to change the Program enhancements as follows

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 W) will occur during Refueling Outage (RFO)-16, at which time the capsule fluence is projected to be equivalent to the 60-year maximum vessel fluence of 6.8×10^{19} n/cm^2, in accordance with ASTM E 185-82, (3) include a provision that the analysis of Capsule W will be used to evaluate neutron exposure for remaining Capsules Y and Z, as required by 10 CFR 50, Appendix H. The withdrawal schedule for one of the remaining capsules (i.e., Capsule Y or Z) will be adjusted, based on the analysis of Capsule W, so that the capsule fluence will not exceed twice the 60-year maximum vessel fluence, in accordance with ASTM E 185-82. The neutron exposure and withdrawal schedule for the last capsule will be optimized to provide meaningful metallurgical data. If the last capsule is projected to significantly exceed a meaningful fluence value, it will either be relocated to a lower flux position or withdrawn for possible testing or re-insertion. The remaining Capsules Y and Z (and archived test specimens available for reconstitution) will be available for the monitoring of neutron exposure if additional license renewals are sought, and (4) include a provision that, if future plant operations exceed the limitations or bounds in Section 1.3 of NRC 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.

Revise LRA Subsection B.2.17 to change the **Enhancements** to **Program Element 6** to read as follows:

- **Program Element 6**
  1) Revise the program to indicate that withdrawal of the next capsule (i.e., Capsule W) will occur during Refueling Outage (RFO)-16, at which time the capsule fluence is projected to be equivalent to the 60-year maximum vessel fluence of 6.8×10^{19} n/cm^2, in accordance with ASTM E 185-82.

  2) Revise the program to indicate that the analysis of Capsule W, to be withdrawn during RFO-16, will be used to evaluate neutron exposure for remaining Capsules Y and Z, as required by 10 CFR 50, Appendix H. The withdrawal schedule for one of the remaining capsules (i.e., Capsule Y or Z) will be adjusted, based on the analysis of Capsule W, so that the capsule fluence will not exceed twice the 60-year maximum vessel fluence, in accordance with ASTM E 185-82. The neutron exposure and withdrawal schedule for the last capsule will be optimized to provide meaningful metallurgical data. If the last capsule is projected to significantly exceed a meaningful fluence value, it will either be relocated to a lower flux position or withdrawn for possible testing or re-insertion. The remaining Capsules Y and Z (and archived test specimens available for reconstitution) will be available for the monitoring of neutron exposure if additional license renewals are sought (i.e., 80 years of operation).

LRA Section 4.2 Changes Resulting from the Response to RAI-4.2.3

Add the following footnote number three to Tables 4.2-2 and 4.2-3 of the HNP LRA to modify the Chemistry Factors of 51.4 and 49.1 applicable to the Intermediate Shell Plate and the Intermediate Shell-to-Lower Shell Circumferential Weld (100%), respectively:

3. The chemistry factors for the surveillance test capsule representing the Intermediate Shell Plate (heat number B4197-2) and the Intermediate Shell-to-Lower Shell Circumferential Weld (100%) (heat number 5P6771) are derived from the surveillance test data.

LRA Subsection 4.2.4 Changes Resulting from the Response to RAI-4.2.4

Add the following paragraph at the end of the **Analysis** section of LRA Subsection 4.2.4:

The current P-T limits are valid through 36 EFPY. The P-T limits for the extended period of operation will be managed by using approved fluence calculations when there are changes in power or core design in conjunction with surveillance capsule results. P-T limits have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G. 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.

LRA Subsection 4.2.5 Changes Resulting from the Response to RAI-4.2.5

Add the following paragraph at the end of the **Analysis** section of LRA Subsection 4.2.5:

HNP will submit the appropriate analysis for LTOP setpoints that will be valid for the period of extended operation. LTOP setpoints have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G. 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.

LRA Subsection 4.2.1 Changes Resulting from the Response to RAI-4.2.6

Replace the last sentence in the second-last paragraph of the **Analysis** section in LRA Subsection 4.2.1 with the following:

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.

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Question No: 3.1-FS-01

NRC Request:
Original Question:
LRA Table B-1 states that based on a thermal aging susceptibility evaluation, the applicable CASS components are not susceptible to thermal aging. Therefore, GALL XI.M12 is not applicable to HNP. Further, LRA Table 3.1.1 item 3.1.1-57 states that the subject components have been screened and found to be not susceptible to thermal aging embrittlement based on the information provided in a letter C. I. Grimes, dated May 19, 2000. Please explain how CASS components that are exposed to treated water environment with temperature >250°C (482°F) have been screened out and why GALL XI.M12 is not applicable to HNP during the extended period of operation.

Revised Question: from email on 6/25/2007:
GALL XI.M12 is not applicable to HNP. What are the HNP's CASS components' casting method, molybdenum content, and ferrite content?

HNP Response:
A methodology for screening cast austenitic stainless steel (CASS) components for susceptibility to thermal aging embrittlement was provided in the NRC staff's evaluation and proposed resolution to License Renewal Issue No. 98-0030. The evaluation and proposed resolution was provided in a letter from C.I. Grimes (USNRC) to D. Walters (NEI), License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components, May 19, 2000. The letter is referenced in NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule (Hereinafter referred to as the "Grimes Letter"). Revision 6 of NEI 95-10 was endorsed by Revision 1 to Regulatory Guide 1.188, Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses.

Determination of the susceptibility of CASS components to thermal aging can use a screening method based upon the molybdenum (Mo) content, casting method, and ferrite content. The specific screening criteria acceptable to the staff are outlined in Table 2 of the Grimes Letter, and are applicable to all primary pressure boundary and reactor vessel internal (RVI) components constructed from SA-351 Grade CF3, CF3A, CF8, CF8A, CF3M, CF3MA, or CF8M, with service conditions above 250°C (482°F).

The significance of finding a particular component not susceptible or potentially susceptible is described in the Grimes

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Letter for each component type. The examination requirements for each component type are provided in Table 3 of the Grimes Letter.

Per Table 3 of the Grimes Letter, valve bodies and pump casings do not require a susceptibility evaluation because both susceptible and non-susceptible components are examined to ASME Section XI requirements. As shown on page 3.1-62 of the LRA, CASS components of the Reactor Vessel Internals are managed by the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program (B.2.6) for Loss of Fracture Toughness due to Thermal Embrittlement. The remaining population of CASS components that require a susceptibility review included the Reactor Coolant Loop elbows and the Pressurizer Spray Head. The d-ferrite level for the Reactor Coolant Loop elbows was calculated as part of the leak-before-break evaluation performed in 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.* The reactor coolant loop elbows are low-molybdenum statically cast components. Since the maximum calculated d-ferrite level is ≤ 20%, the elbows are not susceptible to thermal aging. For the Pressurizer Spray Head, the Certified Material Test Report (CMTR) information was reviewed and the d-ferrite level calculated. The resultant d-ferrite level was below the screening threshold regardless of casting method. Therefore, the Pressurizer Spray Head is not susceptible to thermal aging.

Since the population of components reviewed for thermal aging were shown not to be susceptible to thermal aging, the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is not required for License Renewal.

Specific details on the material composition and casting methods are available in the basis document where the preceding evaluation was performed and is available for your review during the audit.

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Question No: 3.1-FS-02

NRC Request:
LRA Table 3.1.2-1, on Page 3.1-41 identifies water chemistry and one time Inspection (OTI) programs to mange cracking due to SCC for SS vessel flange leak detection line. GALL Report Table 3.1.1 item 23 recommends a plant specific program which should be further evaluated by the staff. LRA Section 3.1.2.2.7 states that the OTI program provides an inspection that verifies either unacceptable degradation is not occurring or triggers actions. Please explain how OTI detects cracking due to SCC for this item. Please describe any site specific or industry operating experience with regard to failure of the SS vessel flange leak detection line that has been identified by HNP.

HNP Response:
Please describe any site specific or industry operating experience with regard to failure of the SS vessel flange leak detection line that has been identified by HNP.

HNP reviewed site and industry OE since January 1, 2005. No site specific or industry OE has been identified with regard to failure of the SS vessel flange leak detection line.

Please explain how OTI detects cracking due to SCC for this item.

Enhanced Visual (VT-1 or equivalent) and/or Volumetric (RT or UT).

Further, what are the corrective actions if cracks are identified by One-Time Inspection?

<u>Corrective Actions</u>

Unacceptable components/structures are processed according to the provisions of the corporate Corrective Action Program which complies with 10 CFR 50 Appendix B.

Why were the water chemistry and one time Inspection (OTI) programs chosen to manage cracking due to SCC for the vessel flange leak detection line?

The vessel flange leak detection line is not classified as ASME Code Class 1, therefore the leak detection line is not included in the One-Time Inspection of Small Bore Class 1 RCS Piping Program. Although these lines are typically dry, if any leaks were to occur at the vessel flange, the components would be exposed internally to primary water. Thus, the Water Chemistry Program is appropriate to manage cracking due to SCC. Since there has been no operating experience that identifies cracking in these lines, the One-Time Inspection program is appropriate to verify the aging effect has not occurred.

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Question No: 3.1-FS-05

NRC Request:
LRA, Table 3.1.2-1, page 3.1-46 lists stainless steel CRDM head penetration thermal sleeves in treated water (outside) environment. Notes J and 113 are used for this line items. Note J indicates that neither the component nor the material and environment combination is evaluated in GALL Report and Note 113 states that these aging effects do not affect the insulation intended function of the thermal sleeves. Therefore, the LRA identifies "None" for aging effect requiring management and aging management program. If a component does not have an intended function to be managed during the period of extended operation, then that component should be screened out and not included in the AMR tables. Please justify elimination of aging effect for stainless steel CRDM head penetration thermal sleeves in treated water in accordance with the requirements of 10 CFR 54.

HNP Response:
HNP will use a combination of the Water Chemistry Program and the One-Time Inspection Program to manage loss of material and cracking of this component.

The Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the cracking and loss of material aging effects. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

A License Renewal Application amendment is required.

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Question No: 3.1-FS-06

NRC Request:
GALL Report, Volume 2, Item IV.A2-16 (R81), lists inlet and outlet safety injection nozzles made of steel with stainless steel cladding. The GALL Report recommends a TLAA to be evaluated for this managing loss of fracture toughness due to neutron irradiation embrittlement in reactor coolant and neutron flux environment for this item. This line rolls up to GALL, Volume 1, Table 1, Line 17. Please explain why comparable line item for inlet and outlet safety injection nozzles is not included in LRA Table 3.1.2-1.

HNP Response:
As stated in Section 4.2.1 (page 4.2-3) of the HNP LRA:

> 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 1×10^{17} n/cm^2, E > 1.0 MeV. The existing AREVA neutron fluence models have been extended to facilitate this evaluation. The materials outside of the traditional beltline region which are expected to receive fluence values greater than 10^{17} n/cm2 were evaluated, and none of these materials were determined to be limiting.

The HNP reactor vessel does not have safety injection nozzles.

The reactor vessel outlet nozzles were not identified as components expected to receive fluences greater than 10^{17} n/cm^2 (E > 1.0 MeV). Therefore, the reactor vessel outlet nozzles do not apply to GALL Report, Volume 2, Item IV.A2-16 (R81).

Five other reactor coolant pressure boundary components outside the beltline region are expected to receive fluences greater than 10^{17} n/cm^2 (E > 1.0 MeV). These components include (1) the circumferential weld that is between the upper and intermediate shells, (2) the upper shell, (3) the inlet nozzle welds, (4) the inlet nozzle, and (5) upper shell longitudinal welds. These components were evaluated and none of these materials were determined to be limiting in ART, C_VUSE or RT_{PTS} values.

The LRA will be amended to include a discussion of the above components and AMR line item(s) will be added as appropriate.

A License Renewal Application amendment is required.

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Question No: 3.1-FS-07

NRC Request:
GALL Report, Volume 2, Items IV.D1-4 (R-01), IV.C2-21 (R-06), and IV.C2-21 (RP-31) list; instrument penetrations and primary side nozzles, safe ends, and welds; pressurizer instrumentation penetrations, heater sheaths and sleeves, heater bundle diaphragm plate, and manways and flanges; and piping, piping components and elements, respectively, made of nickel alloy or steel with nickel alloy cladding. The GALL Report recommends XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components, XI.M2, "Water Chemistry," for PWR primary water, and provide a commitment in the FSAR supplement to comply with applicable NRC Orders for managing cracking due to PWSCC in reactor coolant for these components. These lines roll up to GALL, Volume 1, Table 1, Line 31. Please explain why comparable line items for these components with their associated MEAP are not included in the LRA tables.

HNP Response:
The first sentence in the above question should read: GALL Report, Volume 2, Items IV.D1-4 (R-01), IV.C2-21 (R-06), and IV.C2-13 (RP-31).

IV.D1-4 (R-01)
The HNP steam generators do not have nickel base alloy instrument penetrations, primary side nozzles, safe ends or welds.

IV.C2-21 (R-06)

The HNP pressurizer does not have nickel-alloy instrumentation penetrations, heater sheaths and sleeves, heater bundle diaphragm plate, and manways and flanges.

IV.C2-13 (RP-31)

Except for components which have been more appropriately aligned to GALL Items IV.A2-12, IV.A2-19 and IV.C2-24, there are no nickel-alloy or steel with nickel-alloy cladding "piping, piping components and elements" which would align to this GALL item.

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Question No: 3.1-FS-09

NRC Request:
GALL Report, Volume 2, items IV.A2-6 (R-78), IV.A2-7 (R-79), and IV.A2-8 (R-80) identifies cracking due stress corrosion cracking; loss of material due wear; loss of preload due to thermal effects, gasket creep, and self-loosening, respectively, as aging effects for stainless steel control rod drive head penetration flange bolting in air with reactor coolant leakage environment. The GALL Report recommends XI.M18, "Bolting Integrity" for managing these aging effects. These lines roll up to GALL, Volume 1, Table 1, Line 52. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
The reactor vessel head includes 65 Control Rod Drive Mechanism (CRDM) head penetration nozzles. Of these, 52 CRDM penetrations are used for actual CRDM's, 4 CRDM penetrations are used for the Core Exit Thermocouples (CET), 8 spare CRDM penetrations are capped with a Head Adapter Plug and 1 spare CRDM penetration is used for RVLIS piping. A CRDM Head Penetration Flange is welded to top of each CRDM Head Penetration Nozzle. The top of each CRDM Head Penetration Flange is externally threaded (male) to receive an internally threaded (female) CRDM assembly, CET assembly, head adapter plug or RVLIS adapter as applicable. These components are then seal welded to the head penetration flanges. A bolted flange is not used for any of the above described locations. Therefore, GALL Report, Volume 2, items IV.A2-6 (R-78), IV.A2-7 (R-79), and IV.A2-8 (R-80) do not apply.

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Question No: 3.1-FS-10

NRC Request:
GALL Report, Volume 2, item IV.C2-11 (RP-11) identifies loss of material due to pitting, crevice, and galvanic corrosion as an aging effect for copper alloy piping, piping components, and piping elements in closed cycle cooling water environment. The GALL Report recommends XI.M21, "Closed-Cycle Cooling Water System," for managing this aging effect. This line rolls up to GALL, Volume 1, Table 1, Line 54. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
In the above question, the second sentence should read "...recommends XI.M21, Closed-Cycle Cooling Water System" for this aging effect".

The reactor coolant pumps lube oil coolers include copper alloy tubing with a component cooling water system (closed cycle cooling water) environment. However, the tubing has been identified as a copper nickel alloy with < 15% Zn. Loss of material due to pitting and crevice corrosion is not applicable because the EPRI Mechanical Tools state that these mechanisms are not applicable for copper alloys with zinc content less than 15%. Loss of material due to galvanic corrosion is not applicable because the copper alloy tubing is not in contact with a material that is higher in the galvanic series. Therefore, no aging effects are applicable for this material/environment and thus it is not appropriate to align this component with GALL Report, Volume 2, item IV.C2-11 (RP-11). No other reactor coolant system component has been identified with this material/environment combination.

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Question No: 3.1-FS-11

NRC Request:
GALL Report, Volume 2, item IV.C2-12 (RP-12) identifies loss of material due to selective leaching as an aging effect for copper alloy >15% Zn piping, piping components, and piping elements in closed cycle cooling water environment. The GALL Report recommends XI.M33, "Selective Leaching of Materials," for managing this aging effect. This line rolls up to GALL, Volume 1, Table 1, Line 56. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
In the above question, the second sentence should read "...recommends XI.M33, Selective Leaching of Material" for this aging effect".

The HNP Aging Management Review did not identify any copper alloy with > 15% Zn in a closed-cycle cooling water environment within the systems evaluated in Chapter IV of the GALL Report. Thus, GALL Report, Volume 2, item IV.C2-12 (RP-12) does not apply.

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Question No: 3.1-FS-12

NRC Request:
GALL Report, Volume 2, item IV.C2-16 (R-19) identifies cracking due to cyclic loading as an aging effect for stainless steel or steel pressurizer integral support in air with metal temperature up to 288°C (550°F) environment. The GALL Report recommends XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components for managing this aging effect. This line rolls up to GALL, Volume 1, Table 1, Line 61. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
As stated in LRA Table 3.1.1, Item 3.1.1-61, page 3.1-30, cracking due to cyclic loading is not applicable to this specific pressurizer subcomponent. Although "cracking due to cyclic loading" has not been identified as an applicable aging effect/mechanism for this particular pressurizer subcomponent, the cracking aging effect for the pressurizer is managed by the ASME Inservice Inspection, Subsections IWB, IWC, and IWD Program as stated in the discussion of Item 3.1.1-61 in Table 3.1.1 on page 3.1-30 of the LRA.

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Question No: 3.1-FS-14

NRC Request:
GALL Report, Volume 2, item IV.A2-25 (R-78) identifies loss of material due to wear as an aging effect for vessel shell flange made of steel material in reactor coolant environment. The GALL Report recommends XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components for managing this aging effect. This line rolls up to GALL, Volume 1, Table 1, Line 63. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
The first sentence should read "GALL Report, Volume 2, item IV.A2-25 (R-87) identifies..."

The HNP Aging Management Review included a review of operating experience. There is no history of wear on the HNP reactor flanges. Therefore wear is not identified as an aging effect for this component.

However, as identified in LRA Table 3.1.2-1, page 3.1-39, the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program will be used to manage cracking due to stress corrosion cracking for this component.

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Question No: 3.1-FS-16

NRC Request:
LRA Table 3.1.2-6, on Page 3.1-147 identifies cracking due to SCC, loss of material due to crevice corrosion, and loss of material due to pitting corrosion as the aging effects for stainless steel steam generator tube support plates and flow distribution baffles in treated water (outside) environment. LRA specifies Steam Generator Tube Integrity and Water Chemistry programs for managing these aging effects. Although LRA uses note "F," which means material not in NUREG-1801 for this component, it refers to the GALL Report item IV.D17 (R-42). Further, this is not consistent with the discussion for LRA Table 3.1.1, item 3.1.1-76 that states that "Ligament cracking due to corrosion of the steel steam generator tube support plate (Unique Item IV.D1-17) is not applicable to HNP. All tube support plates are made of type 405 ferritic stainless steel as described in FSAR Section 5.4.2.1.2."

(a) Please clarify the above discrepancy
(b) Please provide supporting documents and bases to demonstrate how Steam Generator Tube Integrity and Water Chemistry programs will manage the above listed aging effect for stainless steam generator tube support plates and flow distribution baffles in treated water

HNP Response:
As stated in LRA Table 3.1.2-6 (Page 3.1-147), the material for these components is "stainless steel." GALL Report item IV.D1-17 (R-42) identifies the material for the line item as "steel." As defined in GALL IX.C (Pg. IX-12), the definition for "steel" does not include "stainless steel." Therefore, since the HNP material is not in NUREG-1801 for this component, the use of Note "F" is appropriate and is consistent with the discussion in LRA Table 3.1.1, item 3.1.1-76.

The aging management strategy for this component includes the Water Chemistry Program and Steam Generator Tube Integrity Programs. The Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the loss of material and cracking aging effects. The Steam Generator Tube Integrity Program manages aging effects by providing a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

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Question No: 3.1-FS-20

NRC Request:
Shearon Harris Nuclear Power Plant (SHNPP) FSAR, Section 4.5.1.1, "Materials Specifications" (Page 4.5.1-1) states that all parts of the control rod dive mechanism (CRDM) that are exposed to reactor coolant are made of metals which resist the corrosive action of the water. Three types of materials are used exclusively: Stainless steel, nickel-chromium-iron, and cobalt based alloys. Further, FSAR Section 4.5.1.1 refers to other materials such as Haynes 25, Inconel X-750, ductile iron, and Dow Corning 302 for the coil stack assembly and latch assembly. However, most of these materials, except stainless steel, are not listed in LRA tables for CRDM assembly. Explain why these materials for CRDM are excluded from the LRA Section 3.1.

HNP Response:
Only the subcomponents of the CRDM having component intended functions were evaluated in the HNP Aging Management Review. Active sub-components are excluded from review based on 10 CFR 54.21(a)(1)(i). As stated in FSAR Section 4.5.1.1(a), "All pressure containing materials of the CRDM comply with Section III of the ASME Boiler and Pressure Vessel Code, and are fabricated from austenitic (Type 304) stainless steel." The pressure boundary components of the CRDM include only the "CRDM Latch Housings" and the "CRDM Rod Travel Housings" which are identified in FSAR Table 5.2.3-1 as type 304 stainless steel.

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Question No: 3.1-FS-22

NRC Request:
The discovery, in October 2006, of five circumferential indications of three Alloy 82/182 dissimilar metal welds (DMW) on the pressurizer at the Wolf Creek Generating Station raised safety concerns based on the size and location of the indications. Based on discussion with the NRC staff, licensee plants susceptible to this condition have committed to enhance inspection frequency and reactor coolant system leakage until actions to mitigate the potential of PWSCC in the affected welds have been completed.

Please briefly explain the commitments that you have made regarding the DMW on the pressurizer. Please explain what are your plans regarding the affected welds during the period of extended operation. Also, please explain your plans for mitigating or inspecting RCS locations with DMW , other than the pressurizer, that are potentially susceptible to PWSCC prior or during the period of extended operation.

HNP Response:
In a letter sent to the NRC (Serial HNP-07-015) on January 31, 2007, Progress Energy provided the following information:

> In October 2006, while performing inspections of pressurizer (PZR) Alloy 82/182 butt welds in accordance with MRP-139, a PWR licensee discovered several circumferential indications in the PZR surge, safety, and relief nozzles. Because of the potential importance of this issue, Carolina Power and Light Company (CP&L) doing business as Progress Energy Carolinas, Inc. commits to the following actions taken or planned at the Harris Nuclear Plant (HNP) for inspecting or mitigating Alloy 82/182 butt welds on PZR spray, surge and relief lines.
>
> Inspection of PZR Alloy 82/182 butt welds at HNP has not yet been completed, but HNP intends to complete all of the inspection and mitigation activities on these locations in refueling outage 14 (RFO-14) in the Fall 2007.
>
> Attachment 1 provides the results of completed inspections and the details of HNP's inspection and mitigation activities.
>
> Attachment 2 provides a discussion of reactor coolant system (RCS) leakage monitoring.
>
> Attachment 3 provides an example of the leakrate trend of unidentified RCS leakage.

Attachment 4 provides the commitments to this letter. This document contains new or revised regulatory commitments.

Future inspections of PZR Alloy 82/182 butt welds at HNP will be performed in accordance with industry guidance (MRP-139). The results of future inspections or mitigations of PZR Alloy 82/182 butt weld locations will be reported to the NRC within 60 days of startup from the outage during which they were performed.

The letter is available on the docket under Accession Number ML070370405.

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Question No: 3.1-FS-26

NRC Request:
LRA Table 3.1.1 Item 3.1.1-23 (Page 3.1-20), under discussion column, states that the Flux Thimble Guide Tubes are aligned to item 3.1.1-30 (IV.B2-12) for cracking due to SCC. GALL IV.A2-1, corresponding to Table 1 item 23 for bottom mounted instrument guide tubes, recommends a plant specific AMP for this line item. However, GALL Table 1 item 30 recommends water chemistry and a commitment by the applicant to follow industries development. LRA Table 3.1.2-1 refers to GALL Table 1 item for flux thimble guide tubes and seals and shows it consistent (Note A) with GALL IV.B2-12 and credits water chemistry for managing SCC aging effects for stainless steel Flux Thimble Guide Tubes and seals in treated water. Please provide bases for using GALL IV.B-12 instead of GALL IV.A2-1 for this line item.
Also justify crediting only water chemistry AMP for managing SCC aging effect for these components.

HNP Response:
All of the Bottom-Mounted Instrumentation (BMI) guide tubes are flux thimble guide tubes.

As stated in LRA 3.1.2.2.7.1, "The Flux Thimble Guide Tubes are aligned to item 3.1.1-30 (IV.B2-12) for cracking due to SCC. See further evaluation for Subsection 3.1.2.2.12."

Question No: 3.1-FS-28

NRC Request:
LRA Table 3.1.2-6 (Page 3.1-138) identifies loss of material due to erosion as an aging effect for external surfaces of "Steam generator feedwater impingement plate and support" fabricated from carbon or low alloy steel in treated water. LRA uses Note E which indicates that HNP is consistent with the GALL Table 1 item 28 and GALL IV.D1-13 (R-39) for component, material, environment and aging effect, but LRA does not credits GALL's AMP. GALL IV.D1-13 recommend using a plant-specific AMP that needs further evaluation. LRA credits One-Time Inspection AMP for managing loss of material due to erosion. Please provide bases for using OTI program for this line item. Please clearly explain how OTI manages steam generator feedwater impingement plate and support during the period of extended power.

HNP Response:
NRC Information Notice 97-88, "Experiences During Recent Steam Generator Inspections", dated December 16, 1997 stated that in May 1997, "the licensee for the Shearon Harris Nuclear Power Plant found that four perforated, carbon steel ribs in a steam generator had been extensively damaged. The ribs are welded to the feedwater impingement plate which shields the steam generator tubes from direct impact of the feedwater flow. The licensee concluded that the high flow velocities of the feedwater eroded the ligaments between the perforations on the ribs."

The Harris Westinghouse Replacement Model Delta 75 SGs do not have feedwater impingement plates as described in NRC IN 97-88. Impingement plates are associated with preheater model steam generators which were installed in the old Harris SG D4s.

The "impingement plates" identified in the LRA are ten (10) .25 inch thick carbon steel (ASME-SA-285, Gr. C) baffles which are located between the primary separator outer riser barrels and prevent direct impingement of feedwater onto the upper shell I.D. There has been no operating experience identifying erosion of the baffles or supports. As stated in the One Time Inspections Program basis document, inspections should be scheduled no earlier than 10 years prior to the period of extended operation begins.

The Steam Generators were replaced in 2001. HNP will have accumulated approximately 15 years of use before inspections under this program begin. Therefore, the One-Time Inspection Program is adequate to verify the aging effect is not occurring.

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Question No: 3.1-FS-31

NRC Request:
Volumes 1 and 2 of the GALL Report, Revision 1 include applicable AMR items to manage cracking due to the various forms of stress corrosion cracking (SCC) in nickel alloy and stainless components in the reactor coolant coolant pressure boundary (RCPB). For aging management, the GALL Report recommends, in part, that the FSAR supplement should include a commitment to implement: (1) NRC Orders, Bulletins, and Generic Letters associated with nickel alloy components, and (2) staff-accepted industry guidelines. Based on its review of the AMR items on SCC in LRA Tables 3.1.2-1, -2, -3, -4, -5, and -6, the staff has determined that: (1) the LRA is either lacking AMRs to manage SCC in some nickel-alloy components of the RCPB, including nickel-alloy pressure boundary welds (i.e., bimetallic welds), or (2) that the existing AMR items for nickel alloy components of the RCPB do not include the applicable FSAR statement to implement: (1) NRC Orders, Bulletins, and Generic Letters associated with nickel alloy components, and (2) staff-accepted industry guidelines. The staff requests that the actions of the applicant:

a. Identify all nickel-alloy component and weld locations in the RCPB that are exposed to the reactor coolant, and clarify whether the LRA includes applicable AMRs on management of SCC or any of its forms (such as primary water stress corrosion cracking, etc.) in the components.

b. If it is determined that the LRA has omitted any applicable AMR entries on management of SCC (or its forms) in specific nickel alloy components or weld, amend the LRA to include the applicable AMRs.

c. Amend all of the applicable AMRs on SCC of nickel-alloy components or welds to include the commitment statement that is referenced for nickel-alloy AMR items in the GALL Report.

HNP Response:

a. As described in Progress Energy's Alloy 600 Strategic Plan the components/welds fabricated from nickel alloy are as follows:

Component	Number Per Unit
Pressure Safety and Relief Nozzle weld	4
Surge Nozzle SE weld	1
Spray Nozzle Safe End	1
CRDM Nozzle/Head	65
CRDM Nozzle Weld	65
Head vent	1
Bottom head Inst. Penetration	52
Core Support Pads	4
Hot leg-to-RV weld	3
Cold leg-to-RV weld	3

b. LRA AMR table entries are required for the Pressurizer Spray Nozzle Safe End (page 3.1-121), Pressurizer Relief Nozzle Safe End (page 3.1-121), and Pressurizer Safety Nozzle Safe End (page 3.1-122). These AMR lines will include the cracking aging effect and refer to HNP's commitments to (1) NRC Orders, Bulletins, and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.

An LRA amendment is required.

c. The HNP LRA currently contains commitments to (1) NRC Orders, Bulletins, and GenericLetters associated with nickel alloys and (2) staff-accepted industry guidelines. Review of the Table 2 items that roll up to the following Table 1 item (3.1.1-31) demonstrate this. Table 1 Item 3.1.1-31 on page 3.1-23 states:

"Consistent with NUREG-1801 with exception. The aging effect is managed by a combination of the Water Chemistry Program and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. *The HNP commitment is described in the FSAR supplement.* Further evaluation is documented in Subsection 3.1.2.2.13. The exception involves differences from the NUREG-1801 recommendations for the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program implementation."

In Subsection 3.1.2.2.13 (page 3.1-13) it states:

"In addition, HNP provides in the FSAR Supplement a commitment to comply with applicable NRC Orders and to implement applicable (1) Bulletins and Generic Letters and (2) staff-accepted industry guidelines."

In Appendix A (FSAR supplement) on page A-5 it states:

"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."

In Item 2 of Enclosure 1 of Serial: HNP-06-136, *Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63, Application for Renewal of Operating License*, November 14, 2006 it states:

"In accordance with the guidance of NUREG-1801, Rev. 1, regarding aging management 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."

The associated commitment discussed in Table 1 item 3.1.1-31 is applicable to all the Table 2 AMR lines in Section 3.1 of the LRA that roll up to it.

Note:

No commitment is associated with Table 1 Item 3.1.1-34 because only stainless steel or stainless steel-clad components are associated with this item. See discussion in Subsection 3.1.2.2.16.1 on page 3.1-14 of the LRA.

No commitment is associated with Table 1 Items 3.1.1-35 and 3.1.1-36 as discussed on page 3.1-25 of the LRA.

A license renewal application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-32

NRC Request:
Volumes 1 and 2 of the GALL Report, Revision 1 include applicable AMR items to manage the aging effects that are applicable to reactor vessel internal (RVI) components made from nickel alloy and stainless steel materials. These aging effects include: (1) cracking due to stress corrosion cracking (SCC, including irradiation assisted stress corrosion cracking [IASCC]), (2) loss of fracture toughness due to neutron irradiation embrittlement or void swelling, (3) changes in dimension due to void swelling, and (4) loss of preload due to stress relaxation. To manage these aging effects, the GALL Report recommends that the FSAR supplement a commitment to: (1) participate in industry programs for investigating and managing the effects of aging on the RVI components, (2) evaluate and implement the results of the industry programs as applicable to the RVI components, and (3) upon completion of these program, but not less than 24 months prior to entering the period of extended, submit an inspection plan for the RVI components to the NRC for review and approval. Based on its review of the AMR items for the RVI components in LRA Table 3.1.2-1, the staff has determined that, while the AMR items do credit the appropriate aging management programs in the GALL Report, the AMR items do not include the applicable provision for the FSAR statement to include the applicable commitment. In order to ensure that aging management of the RVI components will be implemented in accordance with the recommendations of the applicable AMR items in the GALL Report, the staff requests that the applicant amend the applicable LRA AMR items appropriately to include the appropriate FSAR supplement commitment statement.

HNP Response:
The HNP LRA currently contains a commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation. Reviews of the Table 2 items that roll up to the following Table 1 items (3.1.1-22, 3.1.1-27, 3.1.1-30, 3.1.1-33, and 3.1.1-37) demonstrate this. For example, Table 1 Item 3.1.1-22 on page 3.1-20 states:

"The *HNP commitment is described in the FSAR supplement.* Further evaluation is documented in Subsection 3.1.2.2.6."

In Subsection 3.1.2.2.6 (page 3.1-10) it states:

"HNP provides in the FSAR Supplement a commitment to: (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (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 Appendix A (FSAR supplement) on page A-5 it states:

"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 Item 1 of Enclosure 1 of Serial: HNP-06-136, *Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63, Application for Renewal of Operating License*, November 14, 2006 it states:

"In accordance with the guidance of NUREG-1801, Rev. 1, regarding aging management of reactor vessel internals components, 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."

The associated commitment discussed in the Table 1 items (Table 3.1.1) is applicable to all the Table 2 AMR lines in Section 3.1 of the LRA that roll up to it.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-33

NRC Request:
LRA, Table 3.1.2-4, page 3.1-111 lists carbon steel RCP Oil Cooler/Heat Exchanger Components in treated water environment. LRA identifies loss of material due to crevice, pitting, and general corrosion and credits Closed-Cycle Cooling Water Program for managing this aging effect. LRA shows consistency with GALL IV.C2-14 (RP-10) and GALL Table 1 item 53. Note B is used in the LRA. Note B indicates that the HNP program has an exception to the GALL report program.

LRA B.2.11, "Closed-Cycle Cooling Water System Program," under program elements affected by the exception states that:

- Parameters Monitored/Inspected

  Some heat exchangers are not monitored for flow, inlet and outlet temperatures, and differential pressure. In these cases, either the functionality of these heat exchangers is verified by activities outside the Closed-Cycle Cooling Water Program or the specific operating conditions of the heat exchanger render performance testing unreliable.

- Detection of Aging Effects

  Some heat exchangers that are not normally in operation are not periodically tested to ensure operability. However, the functionality of these heat exchangers is verified by activities outside the Closed-Cycle Cooling Water Program.

Please clarify whether this exception is applicable to the RCP Oil Cooler/Heat Exchanger. If so, please explain how the functionality of these heat exchanged is verified.

HNP Response:
The RCP Oil Cooler/Heat Exchanger Components intended function is pressure boundary. These components serve to maintain pressure boundary integrity of the Component Cooling Water System. Therefore, verifying the functionality in relation to a heat transfer intended function is not required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-35

NRC Request:
LRA Table 3.1.2-6 (Page 3.1-136) identifies loss of material due to flow accelerated corrosion (FAC) as an aging effect for the internal surfaces of "Feedwater Nozzle" fabricated from carbon or low alloy steel in treated water. LRA uses Note A which indicates that HNP is consistent with the GALL Report Table 1 item 59 (LRA listed 3.3.1-59, which appears to be a typo) and GALL Report IV.D1-5 (R-37). GALL Report IV.D1-5 identifies wall thinning due FAC. Please explain how LRA identified aging effect is consistent with the GALL Report for this line item.

HNP Response:
HNP considers the aging effects "wall thinning" and "loss of material" to be equivalent with respect to flow-accelerated corrosion.

The LRA will be amended to correct the typographical error identified in the above question.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-39

NRC Request:
GALL Report, Volume 2, item IV.D1-22 (R-48) identifies cracking due to intergranular attack as an aging effect for nickel alloy Steam generator tubes and sleeves in Secondary feedwater/ steam as an aging effect for nickel alloy Steam generator tubes and sleeves in Secondary feedwater/ steam as an aging effect for nickel alloy Steam generator tubes and sleeves in Secondary feedwater/ steam environment. The GALL Report recommends XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water for managing this aging effect. This line rolls up to GALL, Volume 1, Table 1, Line 72. Please explain why comparable line item for this MEAP is not included in the LRA tables.

HNP Response:
For the purposes of AMR, the HNP AMR methodology for predicting the cracking aging effect does not distinguish between this intergranular attack and intergranular stress corrosion cracking. These AERMs are both captured as "cracking due to stress corrosion cracking. This AERM is managed by a combination of the Steam Generator Tube Integrity Program and the Water Chemistry Program (aligned to GALL Volume 2 Item IV.D1-20) as shown on page 3.1-145 of the LRA. This is the same aging management strategy recommended in GALL for Table 2 item IV.D1-22.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-43

NRC Request:
LRA, Table 3.1.2-4, page 3.1-113 lists RCP Oil Spill Protection System Piping fabricated from carbon or low alloy steel internally exposed to Lubricating Oil or Hydraulic Fluid environment. Notes H is used for this line item. Note H indicates that aging effect is not the GALL Report for this component, material and environment combination. LRA credits Lubricating Oil Analysis B.2.25) and One-Time Inspection (B.2.18) programs for managing Loss of Material due to Galvanic Corrosion aging effect. It also lists GALL Report VII.G-26 (A-83) for this item. The aging effect identified for this GALL Report is Loss of material due to general, pitting, and crevice corrosion. Please explain why GALL Report item VII.G-26 is referenced for this LRA line item. Please discuss how Lubricating Oil Analysis and One-Time Inspection programs manage loss of material due to galvanic corrosion (aging mechanism), since aging mechanisms are not defined in LRA B.2.18 and B.2.25.

HNP Response:
Please explain why GALL Report item VII.G 26 is referenced for this LRA line item.

GALL Report VII.G 26 (A 83) is identified for this item because the component has been identified to be subject to loss of material due to general, pitting, and crevice corrosion. Since the component, material, and aging effects are the same as GALL, the Standard Note "A" has been identified.

Please discuss how Lubricating Oil Analysis and One Time Inspection programs manage loss of material due to galvanic corrosion (aging mechanism), since aging mechanisms are not defined in LRA B.2.18 and B.2.25.

For this AMR line item, the environment is lubricating oil. The oil collection piping consists of both carbon steel and stainless steel sections of piping. Since carbon steel piping is connected to stainless steel piping and since the lubricating oil can potentially contain moisture, "galvanic corrosion" is identified as an aging mechanism. Consistent with GALL, the Lubricating Oil Analysis Program "maintains oil systems contaminants (primarily water and particulates) within acceptable limits." Therefore, since galvanic corrosion requires presence of an electrolyte for the mechanism to occur, the program is appropriate to manage the aging effect. No operating experience has been identified to suggest that loss of material has occurred for these components, therefore the One-Time Inspection Program is adequate to verify the aging effect is not occurring.

Question No: 3.1-FS-44

NRC Request:
The AMRs in the LRA include numerous AMRs that credit the TLAA on thermal fatigue with the management of "cracking due to thermal fatigue" in the components. The corresponding AMR items in the GALL Report refer to this aging effect as "cumulative fatigue damage," and recommends that the TLAA on Metal Fatigue be used to manage this aging effect. The TLAA on Metal Fatigue is not an acceptable means of aging management in a component If a fatigue crack has already initiated in the component. For these AMR items, clarify: (1) why the aging effect description (i.e, "cracking due to thermal fatigue") differs from that used in the GALL Report, and (2) why the TLAA on metal fatigue is considered to be capable of managing cracking due to thermal fatigue if fatigue-induced cracking has already initiated in the components.

HNP Response:
1. The terminology used in the HNP LRA is adopted from the EPRI Mechanical Tools. This methodology will identify this as a potential AERM under 2 conditions. First, if an explicit fatigue evaluation has been performed and is part of the current licensing basis. Second, when using the temperature screening criterion for piping and equipment designed to ASME Section III, Class 2 and 3 and ANSI B31.1 that account for fatigue through use of the stress range reduction factor, f. At this point in the AMR process, the AERM is used as a placeholder to indicate that further evaluation is required.

2. A TLAA on metal fatigue is not considered capable of managing cracking due to metal fatigue. After the process described in 1 above, the AMR process ends and the TLAA evaluation begins. Section 4.3 of the LRA documents the resolution of those AMR lines where the potential aging effect of cracking has been postulated.

3. This methodology was used for the Brunswick License Renewal project. The Safety Evaluation Report (page 3-185) addressed this issue as follows:

The applicant's supplemental response to RAI 3.1.2.3.1.1-1, Part B, clarified that the phrase "cracking due to thermal fatigue," as defined in the applicable AMR line items for "Table 2" in LRA Sections 3.1, 3.2, 3.3, 3.4, and 3.5, corresponds to the definition "cumulative fatigue damage" in the applicant AMR line items for "Table 1" in LRA Sections 3.1, 3.2, 3.3, 3.4, and 3.5. The applicant changed the terminology because it recognized that 10 CFR 54.21(a) requires that aging effects be managed for the period of extended operation and because the term "cumulative fatigue damage" referred to a parameter that is used to assess the aging effect of cracking due to thermal fatigue and was not referring to the aging effect itself. Based on this assessment, the change in the terminology from "cumulative fatigue damage" in the "Table 1" to "cracking due to thermal fatigue" in the "Table 2" was done to satisfy the provision and criteria of 10 CFR 54.21(a). This meets the provisions in SRP-LR Sections 3.1, 3.2, 3.3, 3.4, and 3.5 for assessing cracking due to thermal fatigue/cumulative fatigue damage in ASME Code Class 1, 2, and 3 components and any applicable NSR components that are required to have thermal fatigue assessments for license renewal and, therefore, is acceptable. Refer to SER Section 4.3 for the staff's assessment of those plant components that are required to have thermal fatigue analyses for the LRA.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-50

NRC Request:
The LRA includes several AMR items on loss of material due to pitting and crevice corrosion and on cracking due to stress corrosion cracking (SCC) in feedwater (FW) nozzle and auxiliary feedwater (AFW) nozzle thermal sleeves under exposure to treated water. The applicant credits the Water Chemistry Program and the One-time Inspection Program to manage these aging effects in the components. The staff=s has determined that the scope of AMP B.1.28, One-Time Inspection Program, as given in the LRA does not specifically identify the feedwater (FW) nozzle and auxiliary feedwater (AFW) nozzle thermal sleeves as being within the scope of the AMP. The staff requests that the actions:

A. Clarify whether on not there are any other AMPs credited for the LRA that provide for periodic examinations of these thermal sleeves. If there are alternate AMPs, provide your basis why it acceptable to credit the One-Time Inspection Program as the means of managing loss of material and cracking these thermal sleeves in lieu of the alternate AMPs. Amend AMP B.1.28, One-Time Inspection Program, to include the FW nozzle and AFW nozzle thermal sleeves within the scope of the AMP.

B. The staff is of the opinion that initiation of cracking or loss of material in the FW and AFW nozzle thermal sleeves may impact the ability of the thermal sleeves to protect the FW and AFW nozzles against the consequences of thermal cycling, and thus impact their M-6 thermal insulation function. Provide your technical basis for concluding that loss of material or cracking would not impact the M-6 thermal insulation function for these thermal sleeves.

HNP Response:

A. Loss of material from pitting and crevice corrosion and cracking from SCC of the feedwater nozzle thermal sleeves and auxiliary feedwater nozzle thermal sleeves are managed by a combination of the Water Chemistry Program and the One-Time Inspection Program. The Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the subject aging effects. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

The basis document for the One-Time Inspection Program includes the feedwater nozzle thermal sleeves and auxiliary feedwater nozzle thermal sleeves in the one-time inspections to verify effectiveness of the Water

Chemistry Program. This level of detail is not provided in the LRA AMP description.

B. The LRA and the bases documents for the Water Chemistry and One-Time Inspection Program will be amended/revised to include, for the feedwater and auxiliary feedwater nozzles' M-6 Function, the Water Chemistry and One-Time Inspection Programs to manage the aging effects.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.1-FS-51

NRC Request:
(Supplemental Question)

The applicant identifies that a one time inspection is credited to manage loss of material due to general, pitting, or crevice corrosion, and in some cases cracking due to SCC, in the following component commodity groups:

- steam generator feedwater impingement plate and support
- feedwater distribution ring and support
- feedwater distribution ring spray nozzles
- auxiliary feewater internal spray pipe
- moisture separator assembly
- miscellaneous non-pressure boundary steam generator internals

a. For the steam generator feedwater impingement plate and support, feedwater distribution ring and support, feedwater distribution ring spray nozzles, auxiliary feewater internal spray pipe, the commodity groups are within the scope of AMP B.2.18, One-Time Inspection Program. Clarify whether on not there are any other AMPs in the LRA that provide for periodic examinations of these commodity groups. If there are alternate AMPs, provide your basis why it acceptable to credit the One-Time Inspection Program as the means for managing loss of material (and in some cases cracking) in these commodity groups in lieu of the alternate AMPs.

b. AMP B.2.18, One-Time Inspection Program, does not specify that the steam generator moisture separator assembly is within the scope the AMP. Clarify whether on not there are any other AMPs credited for the LRA that provide for periodic examinations of the steam generator moisture separator assembly. If there are alternate AMPs, provide your basis why it acceptable to credit the One-Time Inspection Program as the means of managing loss of material in this component in lieu of crediting these others AMPs. Amend AMP B.2.18, One-Time Inspection appropriately to include the steam generator moisture separator assembly within the scope of the AMP if the component is not currently within the scope of the AMP.

c. Define the specific steam generator commodity groups are being referred under your term "Miscellaneous Non-Pressure Boundary Internals." Provide your basis why it acceptable to credit the One-Time Inspection Program as the means for managing loss of material and cracking in each of these steam generator non-pressure boundary internals and *amend* AMP *B.2.18, One-Time Inspection Program, to specifically place these non-pressure* boundary *internals as being* with the scope of AMP B.2.18.

HNP Response:
Background – One-Time Inspection Program

The One-Time Inspection Program basis document provides a description of Program Scope by tabulating for each material/environment combination: system number/system name, and component inspected/description. Each table also provides aging effects and component intended functions.

a. The steam generator feedwater impingement plate and support, feedwater distribution ring and support, feedwater distribution ring spray nozzles, auxiliary feedwater internal spray pipe commodity groups are managed by the Water Chemistry Program and the One-Time Inspection Program.

For those components that are carbon steel, the aging effects managed are loss of material from pitting, crevice and general corrosion. For those components that are nickel based alloys, the aging effects managed are loss of material from pitting and crevice corrosion and cracking due to SCC.

The basis for why it acceptable to credit the Water Chemistry Program and the One-Time Inspection Program as the means for managing the subject aging effects is as follows:

Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the subject aging effects. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

In addition to the prevention and mitigation of the aging effects provided by the Water Chemistry Program, the One Time Inspection Program will rely on established NDE techniques, including visual, and/or volumetric techniques that are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B. The inspection and test techniques will have a demonstrated history of effectiveness in detecting the aging effect of concern. Evidence of degradation will result in evaluation by Engineering for repair/replacement in accordance with the Corrective

Action Program. Acceptance criteria will be based on construction code, manufacturer's recommendations, engineering evaluation, or metallurgical examination, as appropriate.

b. The steam generator moisture separator assembly commodity group is managed by the Water Chemistry Program and the One-Time Inspection Program. For the carbon steel steam generator moisture separator assembly, the aging effects managed are loss of material from pitting, crevice and general corrosion.

The basis for why it acceptable to credit the Water Chemistry Program and the One-Time Inspection Program as the means for managing the subject aging effects is as follows:

Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the subject aging effects. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

In addition to the prevention and mitigation of the aging effects provided by the Water Chemistry Program, the One Time Inspection Program will rely on established NDE techniques, including visual, and/or volumetric techniques that are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B. The inspection and test techniques will have a demonstrated history of effectiveness in detecting the aging effect of concern. Evidence of degradation will result in evaluation by Engineering for repair/replacement in accordance with the Corrective Action Program. Acceptance criteria will be based on construction code, manufacturer's recommendations, engineering evaluation, or metallurgical examination, as appropriate.

The basis document for the One-Time Inspection Program includes the subject components in the one-time inspections to verify effectiveness of the Water Chemistry Program. This level of detail is not provided in the LRA AMP description.

c. The steam generator Miscellaneous Non-Pressure Boundary Internals commodity group is managed by the Water Chemistry Program and the One-Time Inspection Program. For those components that are carbon steel, the aging effects managed are loss of material from pitting, crevice and general corrosion. For those components that are nickel based alloys or stainless steel, the aging effects managed are loss of material from pitting and crevice corrosion and cracking due to SCC.

Examples of the steam generator Miscellaneous Non-Pressure Boundary Internals include, primary separators, secondary separator vanes, various plates, stay rods and spacer pipes. These components will be added to the basis document Evaluation Group Tables.

The basis for why it acceptable to credit the Water Chemistry Program and the One-Time Inspection Program as the means for managing the subject aging effects is as follows:

Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the subject aging effects. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

In addition to the prevention and mitigation of the aging effects provided by the Water Chemistry Program, the One Time Inspection Program will rely on established NDE techniques, including visual, and/or volumetric techniques that are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B. The inspection and test techniques will have a demonstrated history of effectiveness in detecting the aging effect of concern. Evidence of degradation will result in evaluation by Engineering for repair/replacement in accordance with the Corrective Action Program. Acceptance criteria will be based on construction code, manufacturer's recommendations, engineering evaluation, or metallurgical examination, as appropriate.

The basis document for the OneTime Inspection Program includes the subject components in the one-time inspections to verify effectiveness of the Water Chemistry Program. This level of detail is not provided in the LRA AMP description.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.2.1-07-SA-01

NRC Request:
Provide basis for crediting One-Time Inspection Program for aging management of stainless steel refuelling water storage tank exposed to raw water environment in containment spray system, (Table 3.2.2-1, page 3.2-29).

HNP Response:
LRA Table 3.2.2-1 for this line item refers to plant specific note 214. This note states:

> This line item represents corrosion resulting from water seepage underneath the Refueling Water Storage Tank. The tank area enclosure for the RWST does not drain automatically. Therefore standing rainwater may accumulate to levels above the tank pad elevation.

Chemistry procedures provide guidance on the sampling of drainage water before it is released from the Tank Area. Results of sampling for radioactive contamination will be reported to Operations. Operations will release the water to the storm drain system or return it for processing in the liquid radwaste system.

Due to the limited duration of accumulated water in this area, damage from this aging effect is not expected to be significant. There is no site operating experience identifying degradation on the external surface of this component. For this reason the One-Time Inspection Program was selected. If degradation is more than anticipated, then the item would be entered into the corrective actions program and additional activities, e.g. repair, replacement or additional inspections, would address those concerns.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.2.1-27-SA-02

NRC Request:
In Table 3.2.1, item 3.2.1-27, aging effect is listed as loss of material due to general, pitting, crevice, and galvanic corrosion; for the corresponding items in Table 3.2.2-3, page 3.2-35 aging effect requiring management is annotated as loss of material due to general, pitting, and crevice corrosion. Provide justification for not including galvanic corrosion under the aging effect requiring management in Table 3.2.2-3.

HNP Response:
HNP methodology predicts galvanic corrosion when a component/commodity is electrolytically connected to a dissimilar material. In the case of the RHR Heat Exchanger Components, this component/commodity is not electrolytically connected to a dissimilar material.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.2.2-01-SA-01

NRC Request:
Provide basis for including piping insulation as component requiring aging management in Tables 3.2.2-1, 3.2.2-2, and 3.2.2-3.

HNP Response:
The identification of piping insulation as a component requiring aging management has been addressed generically during the scoping and screening audit. The inclusion of the piping insulation is based on a whether insulation has been included in the current licensing bases (CLB). For example, the areas that warranted research in the HNP licensing basis are as follows:

i) Thermal insulation credited in room cooler evaluations and
ii) Thermal insulation required for environmental control these evaluations could be from statements made in the FSAR or as a basis for the safety related HVAC calculations.
iii) FSAR references to the use of insulation.
iv) Review Design Basis Documents on the Treatment of Insulation

The license renewal basis document for piping insulation will be available for review during the AMR audit.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.2.2-01-SA-02

NRC Request:
Provide basis for crediting One-Time Inspection Program for management of aging effects (loss of material due to crevice corrosion, general corrosion, and pitting corrosion) of carbon or low alloy steel piping, piping components, and piping elements exposed to air/gas (wetted) (inside) environment in containment spray system, (Table 3.2.2-1, page 3.2-28)

HNP Response:
This line item represents the internal surface of carbon steel nitrogen supply piping to the Containment Spray Additive Tank. Corrosion is not expected to occur since a nitrogen blanket is maintained which prevents degradation.

The LRA will be amended to identify that this line item (Table 3.2.2-1, page 3.2-28) will be managed by the One Time Inspection Program and the Water Chemistry Program.

The Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the loss of material aging effect. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-10-MK-01

NRC Request:
Please confirm that there is no high strength steel bolting in the auxiliary systems.

HNP Response:
High strength steel bolts were not identified during the HNP aging management review for auxiliary systems.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-13-MK-01

NRC Request:
On page 3.5-139, HNP is using Water Chemistry Program to manage the loss of material (boral) for the spent fuel storage rack component type. In the Water Chemistry Program, there is no detection program element that would detect the loss of material for this component type. Please describe how the loss of material will be detected.

HNP Response:
HNP agrees that in the Water Chemistry Program, there is no detection program element that would detect loss of material for boral. However, assignment of Standard Note I indicated that aging effects in NUREG-1801 are not applicable. An evaluation for boral with regard to operating experience has determined that there has been no adverse HNP operating experience recorded. Additionally, both the V.C. Summer Nuclear Plant and the Brunswick Steam Electric Plant have been evaluated by the staff for these aging effects, and the Safety Evaluation Reports for License Renewal for these plants has determined the aging effects to be insignificant.

An amendment to the LRA is required based on this response. The LRA and the basis document will be amended to show that the HNP evaluation concluded that boral has no aging effects and therefore requires no aging management. The LRA and basis document plant specific note for the boral line item will be revised to clarify that boral material has no aging effects and therefore requires no aging management.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-21-CM-01

NRC Request:
In Table 3.3.1 on page 3.3-88 of the LRA, Item 3.3.1-21 for AMR component steel heat exchanger components exposed to lubricating oil, identifies microbiologically-influenced corrosion (MIC) as an aging mechanism requiring aging management. LRA Table 3.3.2-17 on page 3.3-216 for diesel engine governor oil cooler components, material carbon steel, does not identify MIC as an aging mechanism. Further NUREG-1801, Volume 2, Table VII.H2-5, also identifies MIC as an aging mechanism for steel in lubricating oil. Explain how MIC is managed for steel heat exchanger components in lubricating oil. MIC is also absent for steel heat exchanger components in LRA Table 3.3.2-19 on page 3.3-229.

HNP Response:
HNP methodology does not assume lubricating oil contains a source of MIC. This is based on the discussion in Appendix C, Section 3.1.6 of EPRI Report TR-1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3, which states:

While MIC contamination is possible in lubricating oil applications, the likelihood of MIC causing extensive damage in lube oil systems is minimal. Even if contamination of the oil occurs, the relatively clean systems and addition of corrosion inhibitors to the lubrication oil does not provide an environment conducive to microorganism growth. The potential for MIC growth and subsequent corrosion effects in lube oil systems appears to be very small based on the addition of lube oil corrosion additives, oil purity testing programs and the extremely low likelihood of lube oil contamination. Even if MIC were to be introduced into these systems, the sampling programs are likely to detect and correct the situation prior to MIC causing any appreciable corrosion of lube oil system components.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-25-CM-01

NRC Request:
In Table 3.3.1 on page 3.3-88 of the LRA, Item 3.3.1-25 for AMR component copper alloy HVAC piping, piping components, and piping elements exposed to condensation (external), identifies that HNP AMR methodology does not predict aging effects in absence of contaminants. Explain why HNP does not expect contaminants to be present for this component type.

Following original response to the above question the following additional information is required:
During a conversation with a member of the Progress Energy License Renewal Team, a discussion about the copper alloy HVAC Components Exposed to Condensation took place. The staff asked which specific HNP HVAC components are applicable to Table 1 item 3.3.1-25. The applicant explained that the specific components that are applicable to Table 1 item 3.3.1-25 are the actual cooling coils within the HVAC unit. The License Renewal Team member also explained that because there are no flat areas where contaminants may collect no stagnant pool or puddle of condensation would raise the local concentration of contaminants which would cause loss of material.

Please answer the following about the Table 1 item 3.3.1-25:

a. Confirm that no HNP copper alloy HVAC components exposed to condensation other than the cooling coils exist, including cooling water piping and refrigerant piping that admits the coolant to the cooling coils.

b. Confirm that there are no flat areas on the HNP copper alloy HVAC components exposed to condensation including those areas created by finned tubes.

HNP Response:
In LRA Table 3.3.1 on page 3.3-89 for item 3.3.1-25, the referenced LRA Section 3.3.2.2.10.3 addresses this concern.

> 3.3.2.2.10.3 Copper Alloy HVAC Components Exposed to Condensation
>
> For copper alloy with a zinc content of less than 15%, the HNP AMR methodology does not predict aging effects in the absence of contaminants. In the HNP ventilation systems, condensation is present but is drained away as it is formed on the cooling coil. This inhibits the concentration of contaminants.

HNP methodology does not require the mere presence of contaminants, but that they are capable of concentrating. Since condensation on cooling coils is frequent, there is little chance for the contaminants to concentrate.

The above position is supported by site operating experience described in the basis documents. A further discussion with system engineers regarding cooling coil leakage experience at HNP reveals no leakage from cooling coils that was initiated by external degradation due to aging effects/mechanisms.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-32-CM-01

NRC Request:
In Table 3.3.2-22 on page 3.3-251 of the LRA for AMR component type piping, piping components, and piping elements, material Copper Alloy >15% Zn, environment Fuel oil (Inside), Note D is referenced. Note D states that component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. Explain what component within Piping, piping components, causes this note to be used instead of Note B.

HNP Response:
HNP methodology generally treats the consistency determination of any typically similar component group as being inconsistent in a system that is not described in NUREG-1801 or when aligning to a NUREG-1801, Volume 2 table line for a system in a different Chapter or Section. This is considered a conservative approach in making standard note determinations. In this case a pipe in the Security Power System is considered different from a pipe in Section VII.H2, which is for the Emergency Diesel Generator System. NUREG-1801 does not contain a system identified as the Security Power System. Clearly there are differences in the actual operation, testing requirements, and environments to which these systems are subjected. Consequently, operating experience may differ. In Table 3.3.2-22, fuel oil components are considered similar to Section VII.H1, Diesel Fuel Oil System, in which case notes A or B were considered acceptable choices.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-33-CM-01

NRC Request:
In Table 3.1.2-4 of the LRA for AMR component RCP oil spill protection system piping, material stainless steel in a lubricating or hydraulic fluid (inside) environment Note A is referenced which claims consistency with GALL VII.G-18. The GALL includes as Aging Effects Requiring Management (AERM) LOM due to crevice corrosion, LOM due to pitting, and LOM due to MIC. The aging effects in Table 3.1.2-4 do not include MIC. Additionally, further evaluation 3.3.2.2.12.2 refers to MIC as an AERM. Discuss whether MIC will be managed for this component, material, and environment in Table 3.1.2-4. MIC is also absent for stainless steel components in lubricating oil in Tables 3.3.2-1, 14, 19, 22, 27, and 32 on pages 3.3-119, -202, -231, -254, -287, respectively.

HNP Response:
The HNP methodology does not assume lubricating oil contains a source of MIC. See the detailed response provided for Question 3.3.1-21-CM-01 for additional information.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-34-CM-01

NRC Request:
In Tables 3.3.2-53, -56, -57, -58, -59, -60, -61, -62, -63, -64, -65, -66, -67, and -71 on pages 3.3-363, -371, -375, -381, -385, -388, -391, -397, -401, -403, -412, -416, -420, and -438, of the LRA for AMR component elastomer seals and components, material elastomer in air-indoor or air/gas environments, Note E is referenced. Note E states: Consistent with NUREG-1801 item for material, environment, and aging effect, but a different AMP is credited or NUREG-1801 identifies a plant-specific AMP. For each of the elastomer seals identified above, explain why Note E was used for the existing and new AMPs used to manage LOM due to wear.

HNP Response:
NUREG-1800, Rev. 1, Section 3.0.1, "Background on Types of Reviews," on page 3.0-3, states:

> A portion of the AMR includes the assessment of the AMPs in the GALL Report. The applicant may choose to use an AMP that is consistent with the GALL Report AMP, or may choose a plant-specific AMP.

When performing the AMR for elastomer seals and components in air-indoor or air/gas environments subject to wear, the HNP License Renewal project selected NUREG-1801 (i.e., GALL) aging management evaluation for auxiliary systems, Item 3.3.1-34. This item recommends that a plant-specific AMP be employed to manage loss of material due to wear of elastomer seals and components. As discussed in Appendix B, Subsection B.1.1, of the HNP LRA, the AMPs employed for License Renewal at HNP are GALL Report AMPs; HNP does not employ any plant-specific AMPs. Therefore, when aligning HNP AMPs with GALL Item 3.3.1-34, GALL AMPs were used to address wear of elastomer components. The selected GALL AMPs were the External Surfaces Monitoring Program and the Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Because this differed from the recommended plant-specific AMP, Note E was selected to signify that a different AMP was being employed.

Note that HNP is preparing a plant-specific AMP for high-voltage, oil-filled cables as a result of the findings from the recent NRC audit of AMPs at HNP.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-39-CM-01

NRC Request:
In Table 3.5.2-17 on page 3.5-139 of the LRA for AMR spent fuel storage rack, material stainless steel in a treated water environment, Note I and plant-specific Note 560 is referenced. Note 560 states that: cracking due to SCC is not an applicable aging effect due to temperature of the fuel pool water being maintained below 140°F. Note I states that: aging effects are not applicable. Provide the HNP AMR methodology where this conclusion is substantiated including provisions for maintaining and monitoring fuel pool water temperature.

HNP Response:
The HNP material/environment aging effect evaluation has determined that Stress Corrosion Cracking (SSC) for stainless steel located in a treated water environment is not an aging mechanism below 140°F, which is consistent with NUREG-1801, Volume 1 Table 3 Items 39 and 90.

The Spent Fuel Pool Cooling and Related Systems review has determined that the HNP normal operating spent pool temperature is limited to a maximum temperature of 125.7°F, as stated in Note 560 on LRA page 3.5-203. The basis document, with the HNP AMR methodology, which concluded that aging effects are not applicable, and the references which provide for maintaining and monitoring spent fuel pool water temperature are available for review at HNP.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-40-CM-01

NRC Request:
In Table 3.3.2-27 on page 3.3-270 of the LRA for AMR fuel oil tank flame arrestors, material carbon or low alloy steel in an air–outdoor (outside) environment, Note E is referenced. Note E states that: this AMP is consistent with NUREG-1801 item for material, environment, and aging effect, but a different AMP is credited or NUREG-1801 identifies a plant-specific AMP. Further, the NUREG-1801, Volume 2 Item is VII.H1-11 which is for the external surfaces of the component. The AMP identified for this component is Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components. Clarify whether this AMP is being applied to the external surfaces of this component.

HNP Response:
In this case the AMP is being applied to the external surface. Maintenance performed on relatively small components such as a flame arrestor is capable of adequately observing the condition of the external surfaces as well as the internal surface. Observation of the external surface would occur in the course of work to clean and inspect the internal components (e.g. during disassembly and reassembly). Consequently, the condition of the entire component both interior and exterior surfaces is considered as part of the maintenance activity.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-42-CM-01

NRC Request:
In Table 3.3.1 Item 3.3.1-42 on page 3.3-93 of the LRA for AMR steel closure bolting exposed to air with steam or water leakage discusses that the AMR methodology for steel (surface temperature <212°F) always predicts crevice and pitting corrosion in addition to general corrosion in accordance with Item 3.3.1-43. Item 3.3.1-42 is not used in the HNP LRA. Clarify how general corrosion is managed for carbon or low alloy steel closure bolting in an air with steam or water leakage environment.

HNP Response:
The aging effect is managed by the Bolting Integrity Program as noted in Item 3.3.1-43. HNP methodology conservatively identifies the bounding set of aging mechanisms because the determination of when general corrosion occurs without any crevice or pitting corrosion is beyond the fidelity in the underlying science. Therefore, Item 3.3.1-42 was not used and Item 3.3.1-43 was consistently referenced in lieu of 3.3.1-42. The description in NUREG-1801, Section XI.M18 supports the applicability of the program for managing the subject aging effects. It states:

The program generally includes periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material due to corrosion, rust, etc.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-50- MK-01

NRC Request:
This Table 3.3.1 item 3.3.1-50 is being used to manage loss of material due to crevice and pitting corrosion for stainless steel piping, pipe components, and pipe elements in the fire protection system using the Closed Cycle Cooling Water System Program. Please describe how this aging effect will be managed by this AMP for this component type.

HNP Response:
The piping components in this group are associated with the Diesel Driven Fire Pump engine coolant system. HNP's methodology considered treated water consistent with the closed-cycle cooling water environment in NUREG-1801 as described in LRA page 3.0-11. This is supported by the Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, EPRI TR-1003056, Revision 3, which states in Section 2.2 of Appendix A.

> "Some of the PWR systems that contain non-borated treated water include main feedwater, main steam, intermediate or closed cooling systems, makeup water, emergency feedwater, and diesel jacket cooling water (typically with an ethylene glycol mix)."

As noted NUREG-1801, Section VII.H2, the Closed Cycle Cooling Water System Program can adequately manage the aging effects for this commodity group.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-70-MK-01

NRC Request:
In LRA Table 3.3.2-27, copper alloy spray nozzles and sprinkler heads exposed to raw water (Inside) environment with an intended function of M-8 (Spray Pattern) do not have an aging effect of fouling listed as an aging effect requiring management. Please explain why fouling is not consider an aging effect requiring management for these component types exposed to raw water.

HNP Response:
The aging effect flow blockage was applied to the steel piping system and not the sprinkler heads themselves, which normally experiences no water while in standby and are free of debris based on the piping configuration. Flow Blockage due to fouling will be added to the external environment of Spray nozzles and Sprinkler heads based on Industry OE. The Inspection of External Surfaces Program will address the sprinkler heads and the Fire Water System Program will address spray nozzles. An enhancement will be added for in-scope spray nozzles to either 1) add a requirement to perform flow testing to ensure proper spray pattern or add a modification to prevent blockage from external sources.

The selection of aging effects was meant to represent the actual condition of the equipment so that program activity and resources are focused on where they are most needed. Partial or full flow blockage, if present, would be expected elsewhere in the system such in pumps, valves, strainers, and long piping headers that normally experience stagnant flow conditions. Most of the sprinkler systems that are in scope are dry sprinkler systems and only periodically wetted during testing or actuation of the preaction/deluge valves. Some sprinkler systems in the Waste Processing Building are wet systems.

The geometry of the copper alloy sprinkler heads are arranged so they are not likely to foul. The sprinkler heads or inlet piping configuration typically extends from the top of the headers so sediment or other debris will not settle in them. Similar arguments can be made for nozzles which are open at the end of the steel piping systems and typically arranged so that sediment or debris will not accumulate in them.

As noted above, Flow Blockage due to Fouling needs to be addressed in piping headers and pumps that supply water to sprinkler heads and nozzles. Flow Blockage due to Fouling is an aging effect/mechanism associated with the entire water supply system. For example see other the piping components subjected to a Raw Water (Inside) environment in LRA Table 3.3.2-27. Flow testing, periodically cleaning system strainers, and other periodic operation of the systems and / or inspections are included in the Fire Water System Program. These program activities ensure that the supply headers are capable of supplying water under adequate pressure to the hose stations, nozzles and sprinkler heads and will be free of debris that could cause flow blockage at the sprinkler heads and nozzles.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-76-MK-01

NRC Request:
In the following LRA Tables 3.3.2-06, 3.3.2-07, 3.3.2-08, 3.3.2-09, 3.3.2-10 and 3.3.2-34, Table 1 line item 3.3.1-76 is used to manage loss of material for the following carbon or low alloy steel component types: buried piping, piping components, and piping elements; normal service water pumps; normal service water seal and bearing water booster pumps; piping, piping components, and piping elements; and system strainers as well as gray cast iron fire service screen wash pumps; normal service water pumps; and piping, piping components, and piping elements exposed to raw water with the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components or the External Surfaces Monitoring Program. Please explain the basis for adequately managing this aging effect without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
Questions 3.3.1-76-MK-01, 3.3.1-79-MK-01, 3.3.1-80-MK-01, and 3.3.1-81-MK-01 have a common theme and will be answered in the response to 3.3.1-76-MK-01. The response to these questions requires the answer to "Please explain the basis for adequately managing this aging effect without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program." Except for the containment isolation components in the Normal Service Water System (NSW), the components identified in these questions are non-safety related, subjected to raw water from the Harris Lake or Cooling Tower basin and are outside the scope of the Open-Cycle Cooling Water Program. This program is based on GL 89-13 as implemented by HNP's GL 89-13 Program, which only applies to the safety related Emergency Service Water and Emergency Screen Wash Systems and not those non-safety related systems associated with the referenced LRA tables.

Although not officially part of the program, the components that are the subject of these questions are subjected to the preventive measures in the Open-Cycle Cooling Water Program. NUREG-1801, Volume 2, Section XI.M20 describes the preventive measures as:

> The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments. Implementation of NRC GL 89-13 includes a condition and performance monitoring program; control or preventive measures, such as chemical treatment, whenever the potential for biological fouling species exists; or flushing of infrequently used systems. Treatment with chemicals mitigates microbiologically influenced corrosion (MIC) and buildup of

macroscopic biological fouling species, such as blue mussels, oysters, or clams. Periodic flushing of the system removes accumulations of biofouling agents, corrosion products, and silt.

The preventive measures described above primarily include design features, periodic flushing activities and/or chemical treatments. These activities include actions taken to prevent degradation. Condition and Performance Monitoring does not prevent degradation and is not considered a preventive action. Nevertheless, the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program represents a collection of condition monitoring activities since it is done in conjunction with ongoing preventive maintenance activities.

LRA Tables:

The following systems are subjected to Raw Water that is circulated through the cooling tower basin and connected systems:

- 3.3.2-6 Circulating Water (CW) System; See LRA Subsection 2.3.3.6 for portion supporting Compliance with Fire Protection 10 CFR 50.48.
- 3.3.2-7 Cooling Tower (CT) System; See LRA Subsection 2.3.3.7 for portion supporting Compliance with Fire Protection 10 CFR 50.48.
- 3.3.2-10 Normal Service Water (NSW) System; See LRA Subsection 2.3.3.12 for portion supporting Compliance with Fire Protection 10 CFR 50.48, seismic continuity, and spatial interaction.
- 3.3.2-13 Waste Processing Building Cooling Water System; See LRA Subsection 2.3.3.15 for portion supporting spatial interaction (Non-essential component cooling water heat exchanger water box components are made of carbon steel clad with 90-10 Cu-Ni and Monel brass. These components are supplied by the NSW System connected at its flanges. Also included are small bore piping components (e.g., heat exchanger instrumentation, test connections, vents and drains).

The following systems are subjected to Raw Water directly from the Lake:

- 3.3.2-8 Cooling Tower Make-Up (CTMU) System; See LRA Subsection 2.3.3.8 for portion supporting Compliance with Fire Protection 10 CFR 50.48.
- 3.3.2-9 Screen Wash System; See LRA Subsection 2.3.3.9 for portion supporting Compliance with Fire Protection 10 CFR 50.48, seismic continuity, and spatial interaction.
- 3.3.2-34 Upflow Filter System; See LRA Subsection 2.3.3.38 for portion supporting spatial interaction.

The Open-Cycle Cooling Water Program supporting document describes the use of the cooling tower basin water as a means to chemically treat the NSW, Circulating Water and the safety-related Emergency Service (ESW) systems. Cooling tower water is the water source for the first group of systems above. Consequently they receive the same chemical treatment preventive measures as the safety related portions of the system, which are supplied by the NSW system. The first group of system component materials is similar to the corresponding components in the safety-related ESW system.

Flushing activities for small bore piping in the NSW are performed as need to support normal plant operation. The NSW system intended function required for supporting ESW system loads during shutdown in case of fire are normally in operation and therefore do not require periodic flushing. Flow through small bore piping in the NSW is needed for normal operation and has no bearing on spatial interactions or seismic continuity. Flushing of small bore piping is not required to support the NSW system intended functions.

Per FSAR Section 9.2.1.2, "When operable, the Cooling Tower can provide cooling water for Unit shutdown without reliance on the Main or Auxiliary Reservoirs. During shutdown, the cooling tower evaporative losses are sufficiently low so that makeup to the Cooling Tower will not be required." Consequently, the CTMU system is not relied upon to supply makeup water during shutdown in case of fire. As noted in LRA Subsection 2.3.3.8, the CTMU System discharge piping forms a pressure boundary with the concrete conduit (pipe) between the CT Basin and the ESW & CT Makeup Intake Structure. No additional preventive measures other than design are needed to support this intended function.

The in-scope portion of the non-safety related Screen Wash System serves Bays 1 and 8 in the Screening Structure. Except for inspections of Intake Bay 1, there are no different preventive measures as compared with those that address the Emergency Service Water Screen Wash System in Bays 6 and 8. The conditions in Screening Structure Bays 6 and 8 are considered representative of the other bays including Bay 1. Because of the higher flow rates in the ESW system, buildup of silt and debris would be worse in Bays 6 and 8 as compared to Bay 1. Findings in Bays 6 and 8 are entered into the corrective action program, which should address any required actions for Bay 1.

The portion of the Upflow Filter system that is subject to AMR is located in the Screening Structure as discussed in LRA Subsection 2.3.3.38. This is an alternate supply of Raw Water to the plant's water treatment facilities and not the normal water supply. This system is included due to the potential for spatial interaction with safety related equipment located in the Screening Structure. Other than design considerations, no additional preventive measures are considered necessary for aging management of this equipment as there are no flow requirements needed to support the license renewal system intended function.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-77-MK-01

NRC Request:
In LRA Table 3.3.2-02, loss of material for carbon or low alloy steel piping, piping components, and piping elements exposed to raw water is managed by the One-Time Inspection Program. Please explain why this program is more suitable to manage this aging effect than the Open Cycle Cooling Water Program.

HNP Response:
These line items in Table 3.3.2-2 reference plant specific notes 369 and 376. These notes state:

Note 376: "This line item represents components associated with the BTRS Chiller condenser wetted by service water (raw water). These components are non-safety related; and, therefore, cannot be managed using the Open-Cycle Cooling Water Program."

Note 369: "An aging effect is not expected to occur, but the data is insufficient to rule it out with reasonable confidence. Therefore, a one-time inspection will provide assurance the aging mechanism is not occurring."

The assignment of the One-Time Inspection Program is not appropriate. The basis documents and the LRA will be amended to reassign this LRA item (page 3.3-134) from the One-Time Inspection Program to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Plant specific Note 369 will be deleted from this line item. Additionally, in the Discussion column for LRA Table line 3.3.1-77, change the last sentence of the second paragraph to read:

"The aging effect is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-79-MK-01

NRC Request:
In the following LRA Tables 3.3.2-06, 3.3.2-09, 3.3.2-10 and 3.3.2-34, Table 1 line item 3.3.1-79 is used to manage loss of material due to crevice and pitting corrosion and flow blockage due to fouling for the following stainless steel component types: piping, piping components, and piping elements; system strainer screens/elements; and, system strainers exposed to raw water with the Inspection of Internal Surfaces in Miscellaneous Piping Program. Please explain the basis for adequately managing these aging effects without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
Questions 3.3.1-76-MK-01, 3.3.1-79-MK-01, 3.3.1-80-MK-01, and 3.3.1-81-MK-01 have a common theme. See the response to Question 3.3.1-76-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-80-MK-01

NRC Request:
In the following LRA Tables 3.3.2-06, 3.3.2-09, 3.3.2-10 and 3.3.2-34, Table 1 line item 3.3.1-80 is used to manage loss of material due to microbiologically influenced corrosion (MIC) for the following stainless steel component types: piping, piping components, and piping elements; system strainer screens/elements; and, system strainers exposed to raw water with the Inspection of Internal Surfaces in Miscellaneous Piping Program. Please explain the basis for adequately managing these aging effects without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
Questions 3.3.1-76-MK-01, 3.3.1-79-MK-01, 3.3.1-80-MK-01, and 3.3.1-81-MK-01 have a common theme. See the response to Question 3.3.1-76-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-81-MK-01

NRC Request:
In the following LRA Tables 3.3.2-09, 3.3.2-13 and 3.3.2-34, Table 1 line item 3.3.1-80 is used to manage loss of material due to crevice, MIC and pitting corrosion as well as flow blockage due to fouling for the following copper alloy component types: piping, piping components, and piping elements; and, system strainer screens/elements exposed to raw water with the Inspection of Internal Surfaces in Miscellaneous Piping Program. Please explain the basis for adequately managing these aging effects without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
Questions 3.3.1-76-MK-01, 3.3.1-79-MK-01, 3.3.1-80-MK-01, and 3.3.1-81-MK-01 have a common theme. See the response to Question 3.3.1-76-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-89-MK-01

NRC Request:
In LRA Table 3.1.2-4, this Table 1 item is being used to manage the loss of material due to boric acid corrosion for the carbon or low alloy RCP lube oil collection tank component type. Please confirm that a Note A was intended as opposed to a Note C for a different component type.

HNP Response:
You are correct; Note A was intended in all cases in Table 3.1.2-4 where this line item is referenced. Line item 3.3.1-89 refers to component groups "Steel bolting and external surfaces exposed to air with borated water leakage."
In the context of the Boric Acid Corrosion Program, the external surfaces of any component are treated the same way. This is supported by NUREG-1801, XI.M10. It states under Scope of the program, "The program covers any structures or components on which boric acid corrosion may occur (e.g., steel and aluminum), and electrical components on which borated reactor water may leak." There are no adverse impacts of assigning Notes A or C to this line item as all susceptible components are treated equally. In this case, HNP believes there is no value in amending the LRA and supporting documentation solely for the purposes of providing a consistent Standard Note.

Question No: 3.3.1-89-MK-02

NRC Request:
In LRA Table 3.3.2-14, this Table 1 item is being used to manage the loss of material due to boric acid corrosion for carbon or low alloy piping, piping components, piping elements, and tanks component type exposed to treated water (inside). Please justify the use of this Table 1 item to manage this aging effect for this component type exposed to this environment.

HNP Response:
There is a mistake in LRA Table 3.3.2-14 on page 3.3-201. Under Piping, Piping components, Piping elements, and tanks, M-1, Carbon or Low Alloy Steel, the Environment for line items 3.3.1-59 and 3.3.1-89 should read Air - Indoor (outside). The LRA will be amended to make this correction.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-90-MK-01

NRC Request:
On page 3.3-352 of the LRA, cracking due to stress corrosion cracking for stainless steel piping, piping components, and piping elements exposed to treated water is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Please explain the basis for adequately managing this aging effect without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
This line has been deleted. The supporting document has been revised and cracking due to stress corrosion cracking is no longer considered applicable. This conclusion was reached since the fluid temperature is not expected to exceed the temperature threshold for this mechanism.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-91-MK-01

NRC Request:
On page 3.3-352 of the LRA, loss of material due to crevice and pitting corrosion for stainless steel piping, piping components, and piping elements exposed to treated water is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Please explain the basis for adequately managing this aging effect without the benefit of the preventive measures provided by a program such as the Open Cycle Cooling Water Program.

HNP Response:
The Spent Resin Storage and Transfer System is a nonsafety-related system and its internal environment is subjected to a treated water environment. This system is not a cooling water system nor associated with one. Therefore, this system should not be subjected to the requirements of GL 89-13.

In NUREG-1801, Section XI.M38, Internal Surfaces in Miscellaneous Piping and Ducting Components Program, describes this system as:

> The program consists of inspections of the internal surfaces of steel piping, piping components, ducting, and other components that are not covered by other aging management programs. These internal inspections are performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. The program includes visual inspections to assure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions.

There are no preventive measures required by this program and none are required to manage this equipment. Because this system is intermittently used, HNP considers it appropriate to use a condition monitoring program based on maintenance activities for managing the aging effect of loss of material. As noted in NUREG-1801, XI.M38, under Scope of the Program, loss of material can be identified by visual techniques and is therefore appropriate for this program. For the foregoing reasons, the selection of the Internal Surfaces in Miscellaneous Piping and Ducting Components Program is appropriate.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.1-98-MK-01

NRC Request:
On page 3.3-364, -377, and -383, copper alloy >15% zn piping, piping components, and piping elements exposed to air/gas (Wetted) (Inside) references Table 1 item 3.3.1-98. Please explain the basis for assigning this Table 1 item which is for piping, piping components, and piping elements exposed to dried air and not a wetted environment.

HNP Response:
On page 3.3-364, -377, and -383, copper alloy >15% Zn piping, piping components, and piping elements exposed to air/gas (Wetted) (Inside) are assigned Plant Specific Note 394. This Note states as follows:

394 This environment represents indoor air for systems with temperatures higher than the dew point, i.e., condensation can occur but only rarely, equipment surfaces are normally dry.

The subject LRA component commodity refers to the inside surface of brass ventilation instrumentation valves, that are expected to be normally dry.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-01-MK-01

NRC Request:
On page 3.3-120, the loss of material due to galvanic corrosion in carbon steel CSIP lube oil piping components is managed by the Lubricating Oil Analysis Program and the One Time Inspection Program with a Note H. How do these programs manage this AE for this material and environment?

HNP Response:
As noted in LRA Section B.2.25,

> The Lubricating Oil Analysis Program maintains oil system contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material, cracking, or reduction of heat transfer. Lubricating oil testing activities include sampling and analysis of lubricating oil for detrimental contaminants.

The HNP basis document that addresses aging mechanism in for dissimilar metals in Lubricating Oil indicates that lubricating oil does not produce any potential aging effects unless there is water contamination and pooling in contact with these dissimilar metals. As noted above, the Lubricating Oil Analysis Program monitors the quality of the oil and identifies the presence of moisture in its samples thereby ensuring the environment is not conducive to galvanic corrosion. The results of the One Time Inspection Program for these components will confirm the acceptability of this approach throughout the period of extended operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-01-MK-02

NRC Request:
On page 3.3-129, the change in material properties due to various degradation mechanisms for elastomers exposed to treated water and air-indoor (Outside) is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. How does this program detect this aging effect through visual inspections?

HNP Response:
This item is the elastomer diaphragm in the Boric Acid Tank. Inspections of elastomeric components will include physical manipulation to detect aging effects, in addition to visual inspection.

LRA Section B.2.24 will be revised by adding:

> Inspections of elastomeric components will include physical manipulation to detect aging effects, in addition to visual inspection.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-02-MK-01

NRC Request:
On page 3.3-133, loss of material due to galvanic corrosion for carbon or low alloy steel piping, piping components, and piping elements exposed to lubricating oil or hydraulic fluid is managed by Lubricating Oil Analysis and One-Time Inspection Programs. How does the Lubricating Oil Analysis Program provide preventive measures for this aging effect?

HNP Response:
See Response to Question 3.3.2-01-MK-01 READS:

As noted in LRA Section B.2.25,

> The Lubricating Oil Analysis Program maintains oil system contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material, cracking, or reduction of heat transfer. Lubricating oil testing activities include sampling and analysis of lubricating oil for detrimental contaminants.

The HNP basis document that addresses aging mechanism in for dissimilar metals in Lubricating Oil indicates that lubricating oil does not produce any potential aging effects unless there is water contamination and pooling in contact with these dissimilar metals. As noted above, the Lubricating Oil Analysis Program monitors the quality of the oil and identifies the presence of moisture in its samples thereby ensuring the environment is not conducive to galvanic corrosion. The results of the One Time Inspection Program for these components will confirm the acceptability of this approach throughout the period of extended operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-04-MK-01

NRC Request:
On page 3.3-145, the reduction of heat transfer effectiveness due to fouling of heat transfer surfaces for stainless steel primary sampling cooler tubes exposed to treated water (Inside) is managed by the Water Chemistry Program. How is the effectiveness of the Water Chemistry Program in managing this aging effect verified for this component type?

HNP Response:
The treated water on the inside of the tubes is primary water, which is maintained at a very high quality by the Water Chemistry Program. The external surfaces of the tubes are managed by the Closed Cycle Cooling Water Program, which includes verification of the heat transfer surfaces. See the response to question B.2.11-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-09-MK-01

NRC Request:
On page 3.3-160, the loss of material due to selective leaching for gray cast iron Fire Service Screen Wash Pumps exposed to raw water is managed by the Selective Leaching of Materials Program. These items have a Note B. Please confirm that a Note D was intended because this is a different component type.

HNP Response:
The Suction Bell and Casing of the Fire Service Screen Wash Pumps are gray cast iron. HNP methodology treats these items as piping components. Therefore, Note B is appropriate.

The above conclusion is consistent with the definition of Piping, piping components, and piping elements as described in Volume 2 of NUREG-1801, Table IX.B, Selected Definitions & Use of Terms for Describing and Standardizing Structures and Components:

This general category includes various features of the piping system that are within the scope of license renewal. Examples include piping, fittings, tubing, flow elements/indicators, demineralizer, nozzles, orifices, flex hoses, pump casing and bowl, safe ends, sight glasses, spray head, strainers, thermowells, and valve body and bonnet. For reactor coolant pressure boundary components in Chapter IV that are subject to cumulative fatigue damage, this can also include flanges, nozzles and safe ends, penetrations, vessel head, shell, welds, stub tubes and miscellaneous Class 1 components, such as pressure housings.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-19-MK-01

NRC Request:
On page 3.3-230, carbon steel piping, piping components, and piping elements exposed to lubricating oil or hydraulic fluid (Inside) does not have an aging effect requiring management. Explain what specific carbon steel component does not have an aging effect requiring management in this environment.

HNP Response:
This item in LRA Table 3.3.2-19 refers to plant specific note 729. Note 729 states on page 3.3-448:

This line represents an immersion heater whose configuration in the tank is such it would not credibly come in contact with water in the event of contamination or pooling. The HNP methodology predicts no aging affects in lubricating oil without water contamination.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-19-MK-02

NRC Request:
On page 3.3-228, cracking due to SCC for stainless steel flow restricting elements exposed to lubricating oil or hydraulic fluid (inside) is being managed by the Lubricating Oil Analysis and One-Time Inspection Programs. What measures in these programs provide prevention for this aging effect?

HNP Response:
Note 397 states that water contamination is assumed as the basis for predicting the potential aging effect. Cracking due to SCC for stainless steel flow restricting elements exposed to lubricating oil or hydraulic fluid is managed by ensuring the amount of water content is within acceptable levels. See the response to question 3.3.2-01-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-22-MK-01

NRC Request:
On page 3.3-252, change in material properties and cracking due to various degradation mechanisms for elastomer piping, piping components and piping elements exposed to fuel oil (inside) are managed by External Surfaces Monitoring. Please explain how this program can manage these aging effects before the loss of intended function. Also please explain why this same item has no aging effect listed for this material and environment combination.

HNP Response:
The elastomer hoses are fuel oil manifold lines that run along and in close proximity to the Security Power System diesel engine. LRA Note 353 describes the rationale for why the elastomer piping, piping components and piping elements have the listed aging effects and why visual observation of the exterior surface is indicative of the internal surface. The note is applicable whether or not the fluid temperature on the inside of the hose is higher than the surrounding air. In this case it is envisioned that the fuel oil hose would be heated by the air surrounding the engine components. Consequently, the aging effects would likely appear on the outside surface before they would on the inside.

Note 353: The aging effects for elastomer hoses are driven by temperature (T>95°F) and not the chemistry of the fluid medium. This is a standby system and temperature is usually maintained above 95°F by the keep warm subsystems. Since the external heat transfer mechanism is natural convection and minimal, it is reasonable to conclude that the aging effects on the external surface are representative of those on the internal surface. Consequently, aging management can be done by external examination.

For the line items with no aging effects, LRA Note 702 identifies the component and its environment.

Note 702: This AMR line represents fuel oil hoses connecting sections of the fuel oil supply and return line that transfers oil between the buried, main storage tank and the fuel oil day tank. They are connected to the tank and protected by an access cover on the concrete slab above the storage tank. The environment selected to represent this area is a cool, damp air space. Cool temperatures (< 95°F) in this air space ensure no aging effects for the hoses.

Question No: 3.3.2-27-MK-01

NRC Request:
On page 3.3-287, for copper alloy >15% zinc spray nozzles, fouling is not considered as an AE than required to be managed for the spray pattern intended function. Please confirm that this is intended and then justify excluding this AE for this intended function.

HNP Response:
Yes, this is intended. See Response to Question 3.3.1-70-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-27-MK-02

NRC Request:
On page 3.3-288, for copper alloy >15% zinc sprinklers, fouling is not considered as an AE than required to be managed for the spray pattern intended function. Please confirm that this is intended and then justify excluding this AE for this intended function.

HNP Response:
Yes, this is intended. See Response to Question 3.3.1-70-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-46-MK-01

NRC Request:
On page 3.3-347, reduction of heat transfer effectiveness due to fouling of heat transfer surfaces for stainless steel fuel pools heat exchanger tubes exposed to treated water (Inside) is managed by the Water Chemistry Program. How does a chemistry only program adequately manage this aging effect? How is the effectiveness of this program verified?

HNP Response:
The treated water on the inside of the tubes is spent fuel pool water, which is maintained at a very high quality by the Water Chemistry Program. The external surfaces of the tubes are managed by the Closed Cycle Cooling Water Program, which includes verification of the heat transfer surfaces. See the response to question B.2.11-MK-01.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.3.2-70-MK-01

NRC Request:
On page 3.3-340, the loss of material due to crevice and pitting corrosion for stainless steel remote sample dilution panel sample cooler tubes exposed to treated water (Outside) is managed using the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Please explain how this program will manage these aging effects on the external surfaces of the tubes.

HNP Response:
Correction, LRA page 3.3-430 shows this item.

Note 382 referring to the treated water side states:

This environment represents the chilled water loop from the Remote Sample Dilution Panel Refrigeration Unit to the Remote Sample Dilution Panel Sample Cooler.

The remote sample dilution panel sample cooler tubes are located inside the cooler, which contains the treated water. The cooler must be disassembled in order to inspect the outside of the tubes.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-2-SA-01

NRC Request:
Provide basis for crediting One-Time Inspection Program for management of aging effects (loss of material due to crevice corrosion, general corrosion, and pitting corrosion) of carbon or low alloy steel piping, piping components, and piping elements exposed to treated water (inside) environment in steam generator chemical addition system, (Table 3.4.2-2, page 3.4-38).

HNP Response:
Using the Steam Generator Chemical Addition System, ammonia and hydrazine were formerly injected by valves supplying the Steam Generators through the Main Feedwater and Auxiliary Feedwater Systems. These valves have been locked closed as chemical requirements for the new Steam Generators have changed.

The LRA Table 3.4.2-2, page 3.4-38 line item piping, piping components, and piping elements exposed to treated water (inside) represents carbon steel piping components that were originally filled with treated water.

The LRA will be amended to identify that this line item (Table 3.4.2-2, page 3.4-38; and the list of aging management programs shown in Section 3.4.2.1.2) will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.

Also, Plant Specific Note 413 will be amended to state the Item represents piping components that are water-filled but not used on a regular basis. The water source is from treated water.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-12-SA-01

NRC Request:
Provide basis for crediting One-Time Inspection Program for management of aging effects (loss of material due to crevice corrosion, pitting corrosion, and cracking due to SCC of stainless steel piping, piping components, and piping elements exposed to treated water (inside) environment in steam generator wet lay up system, (Table 3.4.2-12, page 3.4-74).

HNP Response:
The LRA will be amended to identify that certain components represented by this line item (Table 3.4.2-12, page 3.4-74) will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. Other components represented by this line item will be managed by a combination of the Water Chemistry Program and the One Time Inspection Program. The list of aging management programs shown in Section 3.4.2.1.12 will be revised accordingly.

Also, Plant Specific Note 414 will be amended to remove justification for use of the One Time Inspection Program by itself.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program includes visual inspections to assure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions.

The Water Chemistry Program provides for monitoring and controlling of water chemistry using site procedures and processes for the prevention or mitigation of the loss of material aging effect. The One-Time Inspection Program provides an inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-12-SA-02

NRC Request:
Provide basis for crediting One-Time Inspection Program for management of aging effects (loss of material due to crevice corrosion, general corrosion, microbiologically influenced corrosion, and pitting corrosion) of carbon or low alloy steel piping, piping components, and piping elements exposed to raw water (inside) environment in steam generator wet lay up system, (Table 3.4.2-12, page 3.4-73).

HNP Response:
The Steam Generator Wet Lay Up System is used only to maintain chemistry conditions during wet lay up of the Steam Generators. This will reduce Steam Generator corrosion during inactive periods.

The LRA Table 3.4.2-12, page 3.4-73 line item piping, piping components, and piping elements exposed to raw water (inside) represents a sample cooler with cooling water supplied from the service water system.

The LRA will be amended to identify that this line item (Table 3.4.2-12, page 3.4-73; and the list of aging management programs shown in Section 3.4.2.1.12) will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.

Plant Specific Note 412 will be amended to state the Commodity and environment represent a sample cooler with cooling water supplied with service water. The item represents piping components that are water-filled but not used on a regular basis.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-12-SA-03

NRC Request:
Table 3.4.2-12, page 3.4-73: Provide basis for crediting One-Time Inspection Program for management of aging effects (crevice, general, and pitting corrosion) of carbon or low alloy steel piping, piping components, and piping elements exposed to treated water (inside) environment in steam generator wet lay up system.

HNP Response:
The Steam Generator Wet Lay Up System is used only to maintain chemistry conditions during wet lay up of the Steam Generators. This will reduce Steam Generator corrosion during inactive periods.

The LRA Table 3.4.2-12, page 3.4-73 line item piping, piping components, and piping elements exposed to treated water (inside) represents carbon steel piping components that were originally filled with treated water.

The LRA will be amended to identify that this line item (Table 3.4.2-12, page 3.4-73; and the list of aging management programs shown in Section 3.4.2.1.12) will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.

Also, Plant Specific Note 414 will be amended to state the Item represents piping components that are water-filled but not used on a regular basis. The water source is from treated water.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-3-SA-01

NRC Request:
For items on pages 3.4-41, 3.4-43, and 3.4-45 where components are exposed to hydraulic fluid (inside), applicant takes credit for Lubricating Oil Analysis Program and One-Time Inspection Program to manage loss of material due to crevice, general, and pitting corrosion. Provide difference in properties between lubricating oil and hydraulic fluid and basis for selecting Lubricating Oil Analysis Program.

HNP Response:
The Main Steam Power Operated Relief Valve (PORV) operators use a triaryl phosphate hydraulic fluid. Water contamination is indicated by operating experience.

Lubricating oils are low to medium viscosity hydrocarbons, with possibility of water contamination, used for bearing, gear, and engine lubrication. Piping, piping components, and piping elements (whether copper, stainless steel, or steel) when exposed to lubricating oil that does not have water pooling will not be subject to aging degradation because there are no relevant aging mechanisms. Water contamination is assumed for the HNP lubricating oils.

The basis for selecting the Lubricating Oil Analysis Program follows:

1) At HNP, the analyses of hydraulic fluids and lubricating oils are performed using similar predictive maintenance processes and procedures.

2) The same HNP engineering personnel review analyses of hydraulic fluid and lubricating oils, respond to the findings, and when necessary take the appropriate actions. Equipment assessments created in the plant database "Plantview" between 2002 and 2006 were reviewed for those items relevant to lubricating oil contamination events and/or contamination or changes in lubricating oil properties. The review found results of analyses and recommendations for both lubricating oils and hydraulic fluid.

These items form the basis for selecting the Lubricating Oil Analysis Program to manage the PORV operators' hydraulic fluid environment.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-3-SA-02

NRC Request:
Provide basis for including piping insulation as component requiring aging management in Table 3.4.2-3, page 3.4-40.

HNP Response:
The piping insulation commodity identified on page 3.4-40 includes insulation associated with Main Steam Isolation Valve (MSIV) solenoid valves in the main steam tunnel of the Reactor Auxiliary building (RAB). Please see Plant Specific Note 409. The FSAR states that the ASCO solenoid valves associated with the MSIVs require insulation to reduce the maximum surface temperature to remain below the solenoid valve qualification temperature.

In addition, a review of Engineered Safety features safety related air handling units in the RAB and their associated HVAC calculations was performed. The review concluded that Main Steam System piping located in the RAB could contribute to Post-LOCA heat load and therefore a thermal insulation commodity should be added to the Main Steam System.

Based upon this, an insulation commodity was added to the Main Steam System as shown in Table 3.4.2-3, page 3.4-40.

Question No: 3.4.2-3-SA-03

NRC Request:
Table 3.4.2-3 and 3.4.2-6: Applicant states that Elastomers, PVC or Thermoplastics material not found in GALL for these components (Notes F and J). Provide (a) properties of the specific materials for these components, (b) the basis for concluding that no other aging effects occur in these environments, (c) specific tests and inspection methods for these components including the frequency of inspections, and (d) acceptance criteria and their bases for determining loss of strength of the elastomers, PVC, and thermoplastics.

HNP Response:
The referenced Elastomers, PVC or Thermoplastics materials detailed in Tables 3.4.2-3 and 3.4.2-6 are not found in GALL Chapter VIII. Requested information follows:

LRA Table 3.4.2-3 Elastomers

(a) LRA Table 3.4.2-3 identifies elastomer components which include synthetic rubber hydraulic fluid hoses associated with the Power Operated Relief Valve (PORV) actuators.

(b) The HNP basis calculation "Material/Environment Aging Effect Tools For License Renewal" (HNP Tools calculation) identified the aging effects for elastomers as change in material properties due to various degradation mechanisms and cracking due to various degradation mechanisms. These aging effects for elastomers in hydraulic fluid are consistent with industry practice as detailed in Aging Effects for Structures and Structural Components (EPRI Structural Tools). HNP plant-specific OE did not identify any new or unique aging effects. The HNP Tools calculation predicts wear for elastomers if there is relative motion between surfaces not associated with a design deficiency. No operating experience was discovered to indicate that relative motion between surfaces had occurred, and so, wear was not predicted. Based on the above, no other aging effects were predicted.

(c)(1) The external surface of the subject components will be managed by the External Surfaces Monitoring AMP using visual observations during periodic system walkdowns. A quarterly system walkdown frequency is typically established, with all components of the system observed at least once per operating cycle.

(c)(2) No age-related degradation of PORV elastomer components was found during the system operating experience review and as also stated by the Lubricating Oil Analysis program manager at HNP. The internal surfaces will be managed by the One Time Inspection AMP, which is applicable for situations in which additional confirmation is appropriate because an aging effect is not expected to occur but the data is insufficient to rule it out with reasonable confidence. For these cases, there is to be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly so as not to affect the component or structure intended function during the period of extended operation.

(d) For the elastomers in a hydraulic fluid environment, testing acceptance criteria will be determined prior to the period of extended operation. Acceptance criteria for elastomeric components may include physical manipulation to detect aging effects, in addition to visual inspection. EPRI's Aging Assessment Field Guide and Expansion Joint Maintenance Guide provide further discussion of elastomeric material inspection techniques.

LRA Table 3.4.2-3 PVC or Thermoplastics

(a) LRA Table 3.4.2-3 identifies PVC or Thermoplastic components which include a plastic breather cap for the hydraulic fluid system associated with the PORV actuators.

(b) The HNP Tools calculation determined that the acceptability for use of thermoplastics within a hydraulic fluid environment is a design driven criteria and once the appropriate material has been selected, there should be no applicable aging effects caused by the working fluid. Also, the HNP Tools calculation states that vinyls are generally unaffected by continuous water exposure and that it is basically inert to almost all inorganic substances (e.g., acids, alkalies, and salts), as well as to water, oil grease, alcohols, and similar materials. Thus, no aging effects were predicted.

(c) The external surface of the plastic breather cap will be managed by the External Surfaces Monitoring AMP using visual observations during periodic system walkdowns. A quarterly system walkdown frequency is typically established, with all components of the system observed at least once per operating cycle.

(d)(1) Since there were no internal surface aging effects, the application of internal surface acceptance criteria is not applicable.

(d)(2) For the external surface of the plastic breather cap, acceptance criteria will be determined prior to the period of extended operation. Acceptance may be based upon negligible visual degradation or manufacturer's recommendations, as appropriate.

LRA 3.4.2-6 Elastomers
(a) LRA Table 3.4.2-6 identifies elastomer components which represent rubber instrument air hoses associated with valve operators in the Turbine Building.

(b) The air hose internal surfaces are in contact with dry instrument air. Per NEI 95-10, Revision 6, Appendix F, Section 5.2.2.1, internal surfaces subject to dry instrument air should not be subject to aging effects/mechanisms. Thus, no aging effects were predicted.

(c) The external surface of the air hoses will be managed by the External Surfaces Monitoring AMP using visual observations during periodic system walkdowns. A quarterly system walkdown frequency is typically established, with all components of the system observed at least once per operating cycle.

(d)(1) Since there were no internal surface aging effects, the application of internal surface acceptance criteria is not applicable.

(d)(2) For the external surface of the air hoses, testing acceptance criteria will be determined prior to the period of extended operation. Acceptance criteria for elastomeric components may include physical manipulation to detect aging effects, in addition to visual inspection. EPRI's Aging Assessment Field Guide and Expansion Joint Maintenance Guide provide further discussion of elastomeric material inspection techniques.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-6-SA-01

NRC Request:
Table 3.4.2-6, page 3.4-53, applicant takes credit for GALL items VIII.I-15, and VIII.H-7 for aging management of elastomers. GALL items VIII.I-15, and VIII.H-7 are applicable to steel material only. Justify crediting these GALL Report items for elastomers.

HNP Response:
NEI 95-10 Revision 6, page 33 states, "Each combination of component type, material, environment and aging effect requiring management should be compared with NUREG-1801 Volume 2 line items to identify consistencies. If there is no corresponding line item in NUREG-1801 Volume 2, the combination is a plant-specific aging evaluation result."

Based on the above, HNP is not taking credit for GALL items VIII.I-15, and VIII.H-7 for aging management of elastomers, since NUREG-1801, Revision 1, Section VIII.D1 Feedwater System (PWR) and Section VII.D Compressed Air System do not contain elastomer material.

LRA Table 3.4.2-6, page 3.4-53 contains a line item for piping components constructed of elastomer in a dry air/gas environment with no aging effects and no aging management program required. NUREG-1801 Volume 2 Item VIII.I-15 was referenced in LRA Table 3.4.2-6, page 3.4-53 because the component, environment, aging effect/mechanism and aging management program was the same as GALL line item VIII.I-15.

LRA Table 3.4.2-6, page 3.4-53 contains a line item for piping components constructed of elastomer in an indoor air environment with aging effects of cracking due to various degradation mechanisms and change in material properties due to various degradation maechanisms managed by the External Surfaces Monitoring program. NUREG-1801 Volume 2 Item VIII.H-7 was referenced in LRA Table 3.4.2-6, page 3.4-53 because the components' external surface was the same, the indoor air environment was the same, aging effects were attributed although not identical, and the aging management program was the same as GALL line item VIII.I-15.

In summary, since there is no corresponding line item in NUREG-1801 Volume 2, the referenced GALL line items were selected.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.4.2-8-SA-01

NRC Request:
Section 3.4.1.1, page 3.4-2, applicant states that plant-specific OE review identified instances of general corrosion of carbon steel piping normally operating at temperatures above 212°F in Auxiliary Steam Condensate System and this has been addressed in the AMR of this system. Table 3.4.2-8 for Auxiliary Steam Condensate System does not address any environment with temperatures above 212°F. Clarify this discrepancy and provide specific examples of OE for Auxiliary Steam Condensate System.

HNP Response:
LRA Table 3.4.2-8 for Auxiliary Steam Condensate, page 3.4-63 identifies Piping, piping components, and piping elements made of carbon or low alloy steel in an Air – Indoor environment with aging effects of loss of material due to general corrosion and managed by the External Surfaces Monitoring AMP. This line item references Plant Specific Note 418. On page 3.4-77 this note states: Although these carbon steel lines are normally above 212°F, plant-specific operating experience has indicated there have been incidences where external corrosion has been found.

As part of the operating experience review performed, system engineers were interviewed as to the condition of each system. The system engineer for the Auxiliary Steam Condensate System reported that plant personnel have identified locations on the external surface of carbon steel piping operating above 212°F where corrosion has been found.

AR 154890 - A thru wall pipe leak was found on a drain line in the Water Treatment Building. This through wall condition was the result of external corrosion. This degraded Auxiliary Condensate pipeline is insulated. It appears that water or condensation has gotten underneath the metallic insulation and over time corroded through the pipe wall. This condition has been periodically found on some pipelines that are not in continuous operation. A work order has been created to replace this pipe. Although this OE is for a portion of the system not in the scope of License Renewal, it is considered representative of conditions for the in-scope portion.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-10-JW-01

NRC Request:
In Table 3.5.2-10 on page 3.5-102 of the LRA for AMR component concrete exterior below grade, material reinforced concrete in a soil environment, aging effect cracking, Table 1 item 3.5.1-28, Note 537 is referenced. Note 537 states: HNP does not have a porous concrete subfoundation and does not implement a de-watering system; therefore this aging effect is not applicable and no aging management is required. Explain why Note 537 is associated with Table 1 item 3.5.1-28, which addresses cracks and distortion due to increased stress levels from settlement.

HNP Response:
Note 537 is incorrectly associated with concrete exterior below grade, material reinforced concrete in a soil environment in Table 3.5.2-10 on page 3.5-102 of the LRA.

On the basis of this response, the LRA will be amended to remove Note 537 from concrete exterior below grade, material reinforced concrete in a soil environment in Table 3.5.2-10 on page 3.5-102 of the LRA.

Note 537 is also incorrectly associated with concrete exterior below grade, material reinforced concrete in a soil environment in the following locations in the LRA:
Table 3.5.2-17 on page 3.5-131
Table 3.5.2-26 on page 3.5-173

On the basis of this response, the LRA will be amended to also remove Note 537 from concrete exterior below grade, material reinforced concrete in a soil environment in Table 3.5.2-17 on page 3.5-131, Table 3.5.2-26 on page 3.5-173.

Note 537 is incorrectly associated with concrete foundation, material reinforced concrete in a soil environment [Table 1 item 3.5.1-28 III.A8-2 (T-08)] in the following locations in the LRA:
Table 3.5.2-27 on page 3.5-181

On the basis of this response, the LRA will be amended to also remove Note 537 from concrete foundation, material reinforced concrete in a soil environment [Table 1 item 3.5.1-28, III.A8-2 (T-08)] in Table 3.5.2-27 on page 3.5-181.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-01

NRC Request:
In Table 3.5.2-17 on page 3.5-129 of the LRA for AMR component canal and pool gates, material stainless steel in an air-indoor environment, no aging effect, Note 545 is referenced. Note 545 discusses new fuel storage racks as being stainless steel. Explain why Note 545 is associated with the components canal and pool gates.

HNP Response:
Note 545 for AMR component canal and pool gates, material stainless steel in an air-indoor environment was incorrectly assigned and should be changed to Note 540. Note 540 should be revised to include Canal and Pool Gates as follows:

> The components "Steel Components: All structural steel," "Steel Components: Fuel Pool Liner," "Floor Drains," "Sump Screens" or "Canal and Pool Gates" are aligned with III.B5-5 and/or III.B5-6 as "Miscellaneous Structures" because they have the same material, environment, aging effect and aging management program although they are not the same NUREG-1801 component "Support members; welds, bolted connections, support anchorage to building structure."

The stainless steel canal and pool gates still have no aging effects but this change provides a more consistent use of the plant specific notes.

On the basis of this response, the LRA will be amended to revise Note 540 as stated above in the response and AMR component canal and pool gates, material stainless steel in an air-indoor environment will be revised to delete Note 545 and add Note 540.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-02

NRC Request:
In Table 3.5.2-17 on page 3.5-131 of the LRA for AMR component concrete exterior below grade, material reinforced concrete in a soil environment, aging effect cracking, Table 1 item 3.5.1-28, Note 537 is referenced. Note 537 states: HNP does not have a porous concrete subfoundation and does not implement a de-watering system; therefore this aging effect is not applicable and no aging management is required. Explain why Note 537 is associated with Table 1 item 3.5.1-28, which addresses cracks and distortion due to increased stress levels from settlement. (Reference question 3.5.2-10-JW-01)

HNP Response:
The response to NRC question 3.5.2-10-JW-01 adequately answers this question.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-03

NRC Request:
In Table 3.5.2-17 on page 3.5-132 of the LRA for AMR component concrete interior, material reinforced concrete in an air-indoor environment, aging effect change in material properties, Table 1 item 3.5.1-33, Notes I and 502 are referenced. Note I states: Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. Note 502 states: Change in material properties due to elevated temperature is not an aging effect because the concrete is not subjected to general area temperature >150°F or local area temperatures >200°F. Explain why Notes I and 502 state that there are no aging effects and yet there is the aging effect change in material properties with AMP Structures Monitoring shown for this Table 2 AMR line item.

HNP Response:
The temperature range for the Fuel Handling Building is 60°F to 115.5°F. Note 502 states that there are no aging effects due to elevated temperature for this building. Therefore, applying Notes I and 502 to this line item is correct. However, the aging effect requiring management and aging management programs should both show "None."

On the basis of this response, the LRA and the basis document will be amended to change the aging effect requiring management and the aging management program column items to "None" for Table 3.5.2-17 on page 3.5-132 of the LRA for AMR component concrete interior, material reinforced concrete in an air-indoor environment.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-04

NRC Request:
In Table 3.5.2-17 on page 3.5-134 of the LRA for AMR component fire hose stations, material carbon steel in a borated water leakage environment, aging effect loss of material, Table 1 item 3.5.1-55, Note 544 is referenced. Note 544 references GALL item III.B5-7 which has nothing to do with boric acid corrosion. Explain why Note 544 is referenced instead of Note 539 which makes reference to GALL item III.B5-8 that addresses boric acid corrosion.

HNP Response:
Note 539 should be referenced instead of Note 544 in Table 3.5.2-17 on page 3.5-134 of the LRA for AMR component fire hose stations, material carbon steel in a borated water leakage environment, aging effect loss of material.

On the basis of this response, Table 3.5.2-17 on page 3.5-134 of the LRA for AMR component fire hose stations, material carbon steel in a borated water leakage environment, aging effect loss of material will be amended to change Note 544 to Note 539.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-05

NRC Request:
In Table 3.5.2-17 on page 3.5-136 of the LRA for AMR component new fuel storage rack, material stainless steel in an air-indoor environment, aging effect None, Table 1 item 3.5.1-59, Note 545 is referenced. Note 545 references GALL item III.B5-5. Explain why GALL item VII.A1-1 is shown for this Table 2 AMR line item when neither Note 545 or Table 1 item 3.5.1-59 reference GALL item VII.A1-1.

HNP Response:
NUREG-1801 (GALL) assumed that the New Fuel Storage Racks would be carbon steel and aligned it to the carbon steel GALL item VII.A1-1. However, the HNP New Fuel Storage Racks are stainless steel and GALL Item VII.A1-1 does not apply. There is no stainless steel GALL item for the New Fuel Storage Racks. Note 545 clarifies that the New Fuel Storage Racks are stainless steel components in an Air-indoor environment and would be aligned to the more applicable GALL item III.B5-5, with the same material, environment, aging effect (none), and aging management program (none). However, for clarification, Table 3.5.2-17 on page 3.5-136 of the LRA for AMR component new fuel storage rack, material stainless steel in an air-indoor environment, aging effect None, Table 1 item 3.5.1-59 will be revised to replace VII.A1-1 (A-94) with III.B5-5. In addition, Note 545 will be revised as follows:

NUREG-1801 assumes New Fuel Storage Racks are carbon steel, in an Air-Indoor environment, with aging effects (NUREG-1801 item VII.A1-1). However, the HNP New Fuel Storage Racks are stainless steel. Stainless steel in an Air-Indoor environment has no aging effects. The New Fuel Storage Racks are aligned with NUREG-1801 Item III.B5-5 because the New Fuel Storage Racks have the same material, environment, aging effect (none) and aging management program (none) although they are not the same NUREG-1801 component "Support members; welds, bolted connections, support anchorage to building structure."

On the basis of this response, the LRA will be amended in Table 3.5.2-17 on page 3.5-136 of the LRA for AMR component new fuel storage rack, material stainless steel in an air-indoor environment, aging effect None, Table 1 item 3.5.1-59 to replace VII.A1-1 (A-94) with III.B5-5 (TP-5) and Note 545 will be amended as stated above in this response.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-17-JW-06

NRC Request:
In Table 3.5.2-17 on page 3.5-138 of the LRA for AMR components roof membrane/built-up and seals and gaskets, Note 553 is referenced. Note 553 states in its second sentence: However, these elastomers are in the Group 3 structures rather than a Group 6 structure (III.A6-12). Table 3.5.2-17 is for the fuel handling building. Explain why the note refers to GALL Group 3 structures instead of GALL Group 5 structures, fuel storage facility.

HNP Response:
The basis document dedicated Note 553 for the roof membrane/built-up and seals and gaskets in the Fuel Handling Building. The Fuel Handling Building is a GALL Group 5 structure. The basis document does not make reference to Group 5 structures in Note 553, which was omitted in the LRA. Therefore, Note 553 should also include reference to GALL Group 5 structures.

Additionally, the basis document dedicated Note 553 for the roof membrane/built-up and seals and gaskets in the Tank Area/Building (RAI 3.5.2-27-JW-04) and the Diesel Fuel Oil Storage Tank Building. The Tank Area/Building and the Diesel Fuel Oil Storage Tank Building are GALL Group 8 structures. The basis document does not make reference to Group 8 structures in Note 553, which was also omitted in the LRA. Therefore, Note 553 should also include reference to GALL Group 8 structures.

On the basis of this response, the LRA and the basis document will be amended to change Note 553 to include GALL Group 5 and Group 8 structures.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-21-JW-01

NRC Request:
In Table 3.5.2-21 on page 3.5-148 of the LRA for AMR component concrete: exterior below grade, aging effect cracking and Table 1 item 3.5.1-28 there is only Note A. All other Table 2 AMR line items associated with Table 1 item 3.5.1-28 have also a Note 530. Explain why Note 530 does not apply to this Table 2 AMR line item. Explanation should also apply to Table 3.5.2-21 on page 3.5-149 for the AMR component concrete: foundation.

HNP Response:
The basis document applies Note 530 to AMR line items to address aging effects for concrete due to settlement. However, the basis document did not apply Note 530 to (2) non-safety related structures; namely the Security Building (LRA Table 3.5.2-21) and the Switchyard Relay Building (LRA Table 3.5.2-24), which was also omitted from the LRA. Therefore, Note 530 should be included for the Security Building (Table 1 Item 3.5.1-28 on LRA Table 3.5.2-21) for AMR component Concrete: Exterior Below Grade (LRA Page 3.5.1-148) and for AMR component Concrete: Foundation (LRA Page 3.5-149). Note 530 should also be included for the Switchyard Relay Building (Table 1 Item 3.5.1-28 on Table 3.5.2-24) for AMR component Concrete: Foundation (LRA Page 3.5-164).

On the basis of this response, the LRA and the basis document will be amended to include Note 530 at two locations for the Security Building (3.5.1-28 for AMR components Concrete: Exterior Below Grade, and AMR component Concrete: Foundation) and one location for the Switchyard Relay Building (3.5.1-28 for AMR component Concrete: Foundation).

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-26-JW-01

NRC Request:
In Table 3.5.2-26 on page 3.5-173 of the LRA for AMR component concrete: exterior above grade, the aging effect cracking is associated with Table 1 item 3.5.1-32. Explain why the aging effect cracking is associated with Table 1 item 3.5.1-32 instead of change in material properties.

HNP Response:
Table 1 item 3.5.1-32 in Table 3.5.2-26 on page 3.5-173 of the LRA should be removed because the Turbine Building does not have
Exterior Above Grade Concrete in a Water-Flowing environment associated with GALL Item III.A3-7. This will be consistent with other Group III structures once this change is made (See LRA Table 3.5.2-2 and LRA Table 3.5.2-10 as examples for where Concrete: Exterior Above Grade is not associated with GALL Item III.A3-7.)

On the basis of this response, the LRA and the basis document will be amended to delete the line in Table 3.5.2-26 on page 3.5-173 for Table 1 item 3.5.1-32 for AMR component concrete: exterior above grade.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-26-JW-02

NRC Request:
In Table 3.5.2-26 on page 3.5-173 of the LRA for AMR component concrete exterior below grade, material reinforced concrete in a soil environment, aging effect cracking, Table 1 item 3.5.1-28, Note 537 is referenced. Note 537 states: HNP does not have a porous concrete subfoundation and does not implement a de-watering system; therefore this aging effect is not applicable and no aging management is required. Explain why Note 537 is associated with Table 1 item 3.5.1-28, which addresses cracks and distortion due to increased stress levels from settlement. (Reference question 3.5.2-10-JW-01 and 3.5.2-17-JW-02)

HNP Response:
The response to NRC question 3.5.2-10-JW-01 adequately answers this question.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-26-JW-03

NRC Request:
In Table 3.5.2-26 on page 3.5-177 of the LRA for AMR component phase bus enclosure assemblies, material aluminum in an air-indoor environment, aging effect None, Table 1 item 3.5.1-58, Note 572 is referenced. Note 572 references GALL item III.B2-7. Explain why GALL item III.B2-7 is discussed in the note since GALL item III.B3-2 is shown for this Table 2 AMR line item and Table 1 item 3.5.1-58 does not reference GALL item III.B2-7.

HNP Response:
This question is answered in response to 3.5.2-2-JW-01.

On the basis of this response, the LRA will be amended to revise Note 572 to agree with the basis document, as stated in response to NRC question 3.5.2-2-JW-01.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-26-JW-04

NRC Request:
In Table 3.5.2-26 on page 3.5-177 of the LRA for AMR component phase bus enclosure assemblies, material stainless steel in an air-indoor environment, aging effect None, Table 1 item 3.5.1-59, Note 572 is referenced. Note 572 references GALL item III.B2-7. Explain why GALL item III.B2-7 is discussed in the note since GALL item III.B3-5 is shown for this Table 2 AMR line item and Table 1 item 3.5.1-59 does not reference GALL item III.B2-7.

HNP Response:
This question is answered in the response to 3.5.2-2-JW-01 and 3.5.2-2-JW-02.

On the basis of this response, the LRA will be amended to revise Note 572 to agree with the basis document, as stated in response to NRC question 3.5.2-2-JW-01.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-27-JW-01

NRC Request:
In Table 3.5.2-27 on page 3.5-181 of the LRA for AMR component concrete foundation, material reinforced concrete in a soil environment, aging effect cracking, Table 1 item 3.5.1-28, Note 537 is referenced. Note 537 states: HNP does not have a porous concrete subfoundation and does not implement a de-watering system; therefore this aging effect is not applicable and no aging management is required. Explain why Note 537 is associated with Table 1 item 3.5.1-28, which addresses cracks and distortion due to increased stress levels from settlement. (Reference questions 3.5.2-10-JW-01, 3.5.2-17-JW-02, and 3.5.2-26-JW-02).

HNP Response:
The response to NRC question 3.5.2-10-JW-01 adequately answers this question.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-27-JW-02

NRC Request:
In Table 3.5.2-27 on page 3.5-182 of the LRA for AMR component concrete roof slab, material reinforced concrete in an air-outdoor environment, aging effect loss of material and cracking, Table 1 item 3.5.1-27 and GALL item III.A8-1 are shown. Table 1 item 3.5.1-27 and GALL item III.A8-1 are not associated with loss of material and cracking but Table 1 item 3.5.1-26 and GALL item III.A8-5 are. Explain why GALL item III.A8-1 and Table 1 item 3.5.1-27 are associated with the aging effects for this AMR line item.

HNP Response:
On Table 3.5.2-27, page 3.5-182 of the LRA, for AMR component concrete roof slab, material reinforced concrete in an air-outdoor environment, the first row should be revised as follows:

Loss of Material, Cracking - Structures Monitoring Program - III.A8-5, (T-01) - 3.5.1-26 – Note A

The second row should be revised as follows:

Cracking - Structures Monitoring Program - III.A8-1 (T-03) - 3.5.1-27 – Note A, 504

On the basis of this response, Table 3.5.2-27, page 3.5-182 of the LRA, for AMR component concrete roof slab, material reinforced concrete in an air-outdoor environment will be amended as follows:

The first row will be revised as follows:

Loss of Material, Cracking - Structures Monitoring Program - III.A8-5, (T-01) - 3.5.1-26 – Note A

The second row will be revised as follows:

Cracking - Structures Monitoring Program - III.A8-1 (T-03) - 3.5.1-27 – Note A, 504

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-27-JW-03

NRC Request:
In Table 3.5.2-27 on page 3.5-182 of the LRA for AMR component concrete roof slab, material reinforced concrete in an air-outdoor environment, aging effect cracking, Table 1 item 3.5.1-27 and GALL item III.A8-5 are shown. GALL item III.A8-5 is not associated with cracking alone but GALL item III.A8-1 is. Explain why GALL item III.A8-5 is associated with the aging effect for this AMR line item.

HNP Response:
Refer to Response 3.5.2-27-JW-02.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-27-JW-04

NRC Request:
In Table 3.5.2-27 on page 3.5-184 of the LRA for AMR components roof membrane/built-up and elastomers, Note 553 is referenced. Note 553 states in its second sentence: However, these elastomers are in the Group 3 structures rather than a Group 6 structure (III.A6-12). Table 3.5.2-27 is for the tank area/building. Explain why the note refers to GALL Group 3 structures instead of GALL Group 8 structures, steel tanks and missile barriers.

HNP Response:
The response to NRC question 3.5.2-17-JW-06 adequately answers this question.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-28-JW-01

NRC Request:
In Table 3.5.2-28 on page 3.5-191 of the LRA for AMR component seismic joint filler, material elastomer in an air-indoor environment, GALL items VII.G-1 and VII.G-2 are shown. GALL item VII.G-2 is not associated with an air-indoor environment but GALL item VII.G-1 is. Explain why GALL item VII.G-2 is associated with the environment for this AMR line item.

HNP Response:
GALL item VII.G-2 was associated with an air-indoor environment because NUREG-1800 Table 3.3-1 and NUREG-1801 Table 3, Summary of Aging Management Programs for the Auxiliary System Evaluated in Chapter VII of the GALL Report line item 3.3-61 included both air-outdoor and air-indoor in the component description. However, for clarity, GALL item VII.G-2 will be removed from Table 3.5.2-28 on page 3.5-191 of the LRA for AMR component seismic joint filler, material elastomer in an air-indoor environment.

On the basis of this response, the LRA will be amended to remove GALL item VII.G-2 from Table 3.5.2-28 on page 3.5-191 of the LRA for AMR component seismic joint filler, material elastomer in an air-indoor environment.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-29-JW-01

NRC Request:
In Table 3.5.2-29 on page 3.5-192 of the LRA for AMR component cable tray, conduit etc., material stainless steel and carbon steel, Note 555 is referenced. Note 555 states: Buried conduits that connect to Class1 manholes have a designed water tight clamping arrangement. Provide a drawing showing the water tight clamping arrangement with the components identified. In Table 3.5.2-29 on page 3.5-197 of the LRA for AMR component seals and gaskets, material elastomer in a soil environment, Note 555 is again referenced. Note 555 states: The HNP AMR methodology concluded that the neoprene boot material has no aging effect in soil and, etc. Provide the HNP AMR methodology where this conclusion is substantiated.

HNP Response:
The water tight clamping arrangement for the manholes is shown on FSAR Figure 3.8.4-23 and is available at HNP for review.
The HNP AMR methodology for concluding the neoprene boot material has no aging effect in soil is included in the HNP AMR basis calculation. The HNP methodology is based on industry aging effects tools for structural and mechanical component materials.

However, in Table 3.5.2-29 on page 3.5-192 of the LRA for AMR component cable tray, conduit etc., material stainless steel and carbon steel, the Structures Monitoring Aging Management Program will be deleted and "None" added because there is no direct visual inspection of the stainless steel clamp or carbon steel closure plate in the soil environment. In addition Note 555 will be revised to clarify that water intrusion through the area where the buried conduits connect to the Class I manholes will be detected from inspections inside the manholes under the commodity, Concrete: Interior, using the Structures Monitoring Program.

On the basis of this response, the LRA will be amended in Table 3.5.2-29 on page 3.5-192 of the LRA for AMR component cable tray, conduit etc., material stainless steel and carbon steel in a soil environment to delete the Structures Monitoring Program and add "None." In addition Note 555 will be revised as follows:

"Buried conduits that connect to Class 1 manholes have a designed water tight clamping arrangement. The clamping arrangement provided includes a carbon steel support structure, stainless steel clamps, and a neoprene boot. The purpose of the clamping arrangement is to prevent water intrusion into the manholes.

Due to the inaccessible location of the arrangement and potential damage to the safety related cable, no direct visual inspection is planned for the buried clamping arrangement in the soil. However, degradation of the clamping arrangement leading to water intrusion into the manholes can be determined from inspections performed from inside the manhole. The interior of the manholes (included with Commodity, Concrete: Interior) will continue to be inspected by the Structures Monitoring Program for water intrusion, including the area where the buried conduits connect to the Class I manholes.

The HNP AMR methodology concluded that the neoprene boot material has no aging effect in soil.

The HNP AMR methodology concluded that carbon steel and stainless steel in soil have the aging effect of loss of material."

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-29-JW-02

NRC Request:
In Table 3.5.2-29 on page 3.5-197 of the LRA for AMR component siding, Notes A and 544 are referenced. The component siding is different from the components associated with GALL Volume 2 item III.B.5-7 which is shown for this AMR line item. Explain why the note is an A, consistent component, versus a C, component is different. Note 544 discusses the components non-fire doors, floor drains and fire hose stations, but not siding. Explain why Note 544 is associated with this AMR line item for siding when it does not discuss siding.

HNP Response:
The GALL does not include a category for carbon steel siding. However, GALL Volume 2 item III.B.5-7 includes a category called miscellaneous structures that has similar material and environment as for carbon steel siding. HNP has included carbon steel siding within this category but omitted to include details of the addition for the component to explanation Note 544 as was similarly done for non-fire doors, floor drains and fire hose stations. Additionally, Standard Note C is more appropriate than Note A for this line item to explain that the component is different, but consistent with NUREG-1801 item for material, environment and aging effect.

On the basis of this response, the LRA and the basis document will be amended to change plant specific Note 544, LRA page 3.5-201, to include siding. Additionally, for the siding line item on LRA page 3.5-197, the standard Note A will be changed to standard Note C.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-29-JW-03

NRC Request:
In Table 3.5.2-29 on page 3.5-196 of the LRA for AMR component Pipe, material Reinforced Concrete, environment Soil, aging effects: Cracking, Loss of material and Change in material properties; the aging management program shown is Buried Piping and Tanks Inspection Program. A review of the Buried Piping and Tanks Inspection Program shows that the inspection of buried reinforced concrete pipe is not part of the program. Explain why the Buried Piping and Tanks Inspection Program is shown for this line item to manage the aging effects when the component reinforced concrete pipe is not within the scope of the program.

HNP Response:
The LRA inadvertently did not incorporate a basis document change into LRA Table 3.5.2-29. LRA Table 3.5.2-29 should be revised as follows:

Table 3.5.2-29 on page 3.5-196 of the LRA for AMR component pipe, material reinforced concrete, in a soil environment, the Buried Piping and Tanks Inspection Program should be deleted and replaced with the Structures Monitoring Program. In addition Note 547 should be revised as follows:

"The reinforced concrete pipe and asbestos cement manifold header are mechanical components utilizing civil materials which do not align with NUREG-1801. The HNP AMR methodology concluded that reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments and reinforced concrete pipe in a Soil environment have the same aging effects as structural concrete. The Structures Monitoring Program was selected to manage the aging effects of reinforced concrete pipe in a Soil environment and Mechanical aging management programs were selected to manage the aging effects of reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments."

On the basis of this response, the LRA will be amended in Table 3.5.2-29 on page 3.5-196 of the LRA for AMR component pipe, material reinforced concrete, in a soil environment to delete the Buried Piping and Tanks Inspection Program and add the Structures Monitoring Program. In addition Note 547 on page 3.5-202 of the LRA will be amended as follows:

"The reinforced concrete pipe and asbestos cement manifold header are mechanical components utilizing civil materials which do not align with NUREG-1801. The HNP AMR methodology concluded that reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments and reinforced concrete pipe in a Soil environment have the *same aging effects as structural concrete. The Structures Monitoring Program was selected to manage the aging effects* of reinforced concrete pipe in a Soil environment and Mechanical aging management programs were selected to manage the aging effects of reinforced concrete and asbestos cement pipe in Raw Water and Air-Outdoor environments."

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-2-JW-01

NRC Request:
In Table 3.5.2-2 on page 3.5-83 of the LRA for AMR component phase bus enclosure assemblies, material aluminum in an air-indoor environment, Note 572 is referenced. Note 572 states: The component phase bus assemblies is aligned with III.B2-7 because it has the same material, environment, aging effect, and aging management program; although it is not the same NUREG-1801 component support members; welds, bolted connections, support anchorage to building structure. However, this Table 2 AMR line item is aligned with GALL Volume 2 item III.B3-2. Explain the discrepancy between the GALL alignment referenced in Note 572 and the GALL alignment shown for this Table 2 AMR line item.

HNP Response:
The basis document has the following text for Note 572:

The components "Phase Bus Assemblies" are aligned with III.B2-7 or III.B3-2 or III.B3-5 because they have the same material, environment, aging effect and aging management program although they are not the same NUREG-1801 component "Support members; welds, bolted connections, support anchorage to building structure." The basis document is available for review at HNP.

This Note 572 change inadvertently did not get incorporated into the LRA before submittal to the NRC. III.B3-2 is correct for Table 3.5.2-2 AMR line item for phase bus enclosure assemblies, material aluminum in an air-indoor environment on page 3.5-83 of the LRA.

The revised Note 572 applies to other locations in the LRA as well as follows:

Table 3.5.2-25, page 3.5-169 - GALL Volume 2 Item B2-7 is correct for AMR component phase bus enclosure assemblies, material aluminum in an air-outdoor environment.

Table 3.5.2-26, page 3.5-177 - GALL Volume 2 Item B3-2 is correct for AMR component phase bus enclosure assemblies, material aluminum in an air-indoor environment.

On the basis of this response, the LRA will be amended to revise Note 572 to agree with the basis document.

A License Renewal Application amendment is required.

Question No: 3.5.2-2-JW-02

NRC Request:
In Table 3.5.2-2 on page 3.5-83 of the LRA for AMR component phase bus enclosure assemblies, material stainless steel in an air-indoor environment, Note 572 is referenced. Note 572 states: The component phase bus assemblies is aligned with III.B2-7 because it has the same material, environment, aging effect, and aging management program; although it is not the same NUREG-1801 component support members; welds, bolted connections, support anchorage to building structure. However, this Table 2 AMR line item is aligned with GALL Volume 2 item III.B3-5. Explain the discrepancy between the GALL alignment referenced in Note 572 and the GALL alignment shown for this Table 2 AMR line item.

HNP Response:
Same response as for 3.5.2-2-JW-01. Note 572 will be revised.

III.B3-5 is correct for Table 3.5.2-2 AMR line item for phase bus enclosure assemblies, material stainless steel in an air-indoor environment on page 3.5-83 of the LRA.

The revised Note 572 applies to one other location in the LRA as well as follows:

Table 3.5.2-26, page 3.5-177 - GALL Volume 2 Item III.B3-5 is correct for AMR component phase bus enclosure assemblies, material stainless steel in an air-indoor environment.

On the basis of this response, the LRA will be amended to revise Note 572 to agree with the basis document, as stated in response to NRC question 3.5.2-2-JW-01.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.5.2-7-JW-01

NRC Request:
In Table 3.5.2-7 on page 3.5-95 of the LRA for AMR component supports for non-ASME piping & components, material stainless steel in an air-outdoor environment, Note 573 is referenced. Note 573 states: The HNP AMR methodology concluded that stainless steel in the air-outdoor environment has no aging effect. Provide the HNP AMR methodology where this conclusion is substantiated.

HNP Response:
The HNP methodology is substantiated in the bases documents located at HNP. The HNP methodology is based on industry aging effects tools for structural and mechanical component materials. In summary, the Air-Outdoor environment at HNP is subject to normal periodic wetting but is not exposed to an aggressive environment from any nearby industrial facilities or to a salt water environment which could have the potential to concentrate contaminates and cause aging effects for stainless steel. In addition, there is no HNP operating experience which indicates aging effects for stainless steel in the Air-Outdoor environment has occurred.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.6-RM-01

NRC Request:

Please identify the changes required to the LRA Table 3.6.2.1 for component "high-voltage power cables" and note "602" to reflect the HNP's response to audit question LRA 3.6.2-1-RM-02.

HNP Response:

Revision to LRA Table 3.6.2-1 and note "602" is shown below:

Component Commodity	Intended Function	Material	Environment	AERM	AMP	NUREG-1801, Vol. 2 Item	Table 1 Item	Notes
High-Voltage Power Cables	Electrical continuity	Oil-impregnated paper (insulation) Lead (sheath) Organic polymers (outer jacket)	Air-Outdoor	Loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure	Oil-Filled Cable Testing Program			J, 602

Notes for Table 3.6.2-1

602 – The HNP cables are high-voltage, oil-filled, paper insulated, lead-sheathed cables. Lead sheath cables are designed for submergence for extended periods. The impregnation of the paper tape improves the insulation's electrical resistance and provides an extra layer of defense against moisture ingress. The highly refined oil used for the insulating medium also serves to dissipate heat from the conductors and cools the cable when operating under load. The HNP cables have an Okolene (black polyethylene) outer jacket. Site and industry operating experience has shown this design to be extremely reliable in underground applications. Periodic cable testing will be performed to assure that the effects of aging will be managed during the period of extended operation.

A License Renewal Application amendment is required

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.6-RM-02

NRC Request:
Please describe the maintenance and inspections performed on the switchyard bus and connections including their frequencies to address potential degradation of switchyard connections. Discuss the HNP operating experience.

HNP Response:
Currently, HNP performs thermographic surveys on the switchyard bus and connections on a quarterly basis. For the period of extended operation, thermographic surveys of switchyard connections within the scope of License Renewal will be performed under the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Subsection B.2.37.

The aluminum bolting hardware used for the connections to the switchyard bus was selected to be compatible with the aluminum connector/conductor coefficient of thermal expansion. This ensures that the contact pressure of the bolt and washer combination used in the connector is maintained to the initial vendor specified torque value. The inscope bolted connections are those used to electrically connect the transmission conductors to the switchyard bus and the main power transformers. HNP design incorporates the use of stainless steel "Belleville" washers on the bolted electrical connections to the main power transformers to compensate for temperature changes, maintain the proper torque and prevent loosening of dissimilar metal connection hardware. This method of assembly is consistent with the good bolting practices recommended in EPRI Technical Report 1003471. A review of site operating experience (OE) is documented in a plant evaluation. This evaluation revealed no switchyard bolted connection failures attributed to aging. This confirms the proper design and installation of HNP transmission conductor bolted connections and demonstrates their reliability.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: 3.6-RM-03

NRC Request:
In LRA, Table 3.6.2-1, the applicant states that uninsulated ground conductors and connectors have no aging effects requiring management and no AMP is required. Discuss why torque relaxation for bolted connection is not a concern at HNP.

HNP Response:
Torque relaxation of bolted connections on uninsulated ground conductors is not a concern at HNP because all connections are bonded together using the powder weld (i.e. CADWELD®) process. Operating experience has proven that this method of bonding produces a permanent exothermic connection that will not loosen. Therefore, torque relaxation of bolted connections on uninsulated ground conductors is not an aging effect requiring management for the period of extended operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: AMP-MH-01

NRC Request:
In Appendix B of the LRA, the new programs consistent with NUREG-1801 do not have any summary of the 10 elements associated with the Aging Management Program. For each of the new programs please provide a more detailed summary of the program's 10 elements of an Aging Management Program (addressing each of the 10 elements is not specifically required).

Following original response to the above question the following additional information is required:
1. In response to NRC question AMP-MH-01, in the operating experience element for AMP B.2.21 - One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program, the applicant stated that "Augmented volumetric examinations were performed on selected piping welds in the charging and safety injection lines considered potentially subject to thermal fatigue. No indications of cracking were found."

Applicant to provide copies of these examination documents for NRC review.

2. HNP claims consistency with the GALL Report XI.M32, One-time Inspection for program element Parameters Monitored/Inspected which states that Loss of Preload due to thermal effects, gasket creep and self-loosening will be monitored by Visual (VT-3 or equivalent) Inspection for loosening of components.

In response to NRC question AMP-MH-01, in the parameters monitored/inspected element for AMP B.2.18 - One-Time Inspection Program, there is no mention of Loss of Preload aging effect in the table provided. Explain what method will be used to manage Loss of Preload at HNP.

3. HNP claims consistency with the GALL Report XI.M35, One-time Inspection of ASME Code Class 1 Small-Bore Piping for program element Scope of Program which states that "Guidelines for identifying piping susceptible to potential effects of thermal stratification or turbulent penetration are provided in EPRI Report 1000701, "Interim Thermal Fatigue Management guideline (MRP-24)," January 2001.

In response to NRC question AMP-MH-01, in the scope of program element for AMP B.2.21 - One-Time Inspection of

ASME Code Class 1 Small-Bore Piping Program, the applicant states that "The program will detect cracking in small-bore piping regardless of the aging mechanism causing the cracking and will consider the guidance in EPRI Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)."

Describe the major differences between EPRI Report 1000701 and EPRI Report 1011955. Describe how these differences affect other elements of the program, i.e., parameters monitored, detection of aging effects, monitoring and trending, and acceptance criteria.

4. HNP claims consistency with the GALL Report XI.M35, One-time Inspection of ASME Code Class 1 Small-Bore Piping for program element Detection of Aging which states that "For ASME Code Class 1 small-bore piping, one-time inspections using volumetric examination are performed on selected weld locations to detect cracking."

In response to NRC question AMP-MH-01, in the detection of aging element for AMP B.2.21 - One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program, the applicant states that "HNP has not experienced cracking of small-bore piping due to stress corrosion or thermal/mechanical loading."

Describe inspection techniques, frequency, sample size, data collection, and timing of one-time inspection to ensure timely detection of aging effects.

A License Renewal Application amendment is required.

HNP Response:

Responses to the Supplemental Questions are provided at the end of the section addressing each program. The information in the supplemental responses will not be included in the LRA.

There are 11 new programs, and they are addressed in the following LRA Subsections. Amend the LRA by replacing the **Operating Experience** discussion in each of the LRA Subsections with the following discussion of program elements.

B.2.6 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

Discussion of Program Elements

Scope of Program

This aging management program (AMP) will manage loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement in CASS reactor vessel internals components within the scope of License Renewal. The program will consider potential synergistic effects from thermal aging and neutron embrittlement and will be implemented as a supplemental examination (augmented inspection) as part of an existing ASME Code 10-year inspection interval Inservice Inspection plan.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The components within the scope of this AMP have been evaluated and determined to be potentially susceptible to thermal aging and neutron irradiation embrittlement based on materials of construction and neutron fluence values.

Detection of Aging Effects

The program allows for a component-specific evaluation, including a mechanical loading assessment to determine if the loading is compressive or low enough to preclude fracture. The component inspection techniques have not been determined at this time; however, they will be selected based on the recommendations of NUREG-1801, Section XI.M13.

Monitoring and Trending

Supplemental inspections performed under the program will meet the requirements of IWB-2400 using reliable methods.

Acceptance Criteria

Flaw detection criteria for the CASS components will be in accordance with the recommendations of NUREG-1801, Section XI.M13.

Corrective Actions

The Corrective Action program element is addressed generically in Subsection B.1.3.

Currently, HNP uses the 1989 Edition of ASME Section XI (the Code) for determining repair and replacement requirements. Repairs are performed in accordance with Article IWA-4000 and the corresponding IWX-4000 of the IWB/C/D portions of the Code. Replacements are performed in accordance with Article IWA-7000 and the corresponding IWX-7000 of the IWB/C/D portions of the code.

HNP is currently updating the ISI Program to the ASME Section XI, 2001 Edition with addenda through 2003, per 10 CFR 50.55a. In this edition/addenda of the Code, all Repair/Replacement Activities are controlled under Article IWA-4000. Section XI states, "The term repair/replacement activity includes those activities previously known as repair, replacements, modification, or alteration." All of the IWX-4000 and IWX-7000 articles of Section XI have been incorporated into IWA-4000.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new AMP for thermal aging and neutron irradiation embrittlement of CASS. The recommendations of NUREG-1801, Section XI.M13, were developed using research data obtained on both laboratory-aged and service-aged materials. The program will be updated periodically to incorporate industry knowledge. NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.18 ONE-TIME INSPECTION PROGRAM

Discussion of Program Elements

Scope of Program

This AMP is used to determine the need for additional aging management for structures and components that are managed by other programs. This is done by confirmation that either 1) the aging effect is not occurring, or 2) the aging effect is occurring very slowly so as not to affect the intended function of the structures and components during the period of extended operation. The program includes the one-time inspections specified in NUREG-1801 as well as inspections identified during plant-specific aging management reviews. One-time inspections of small-bore Class 1 piping is addressed in a separate AMP.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

Program activities will be directed towards aging effects/mechanisms that were identified during aging management reviews. Each structure or component requiring a one-time inspection is evaluated to determine: sample size (refer to the program element Detection of Aging Effects), inspection locations, and examination techniques. Typically the method of inspection would include:

Aging Mechanism	Typical Inspection Method
Crevice/Pitting Corrosion	Visual (e.g., VT-1) and/or volumetric
General/Galvanic/MIC	Visual (e.g., VT-3) and/or volumetric
Erosion	Visual (e.g., VT-3) and/or volumetric
Fouling	Visual (e.g., VT-3) and/or volumetric
Cracking	Enhanced Visual and/or volumetric

Detection of Aging Effects

Sample size would be based on considerations, such as, accessibility, leading or bounding locations, safety significance, severity of operating conditions, and design margins. Progress Energy non-destructive examination (NDE) procedures and personnel qualifications meet the requirements of the ASME Code, where applicable. Administrative controls and quality assurance requirements for NDE activities are implemented in accordance with 10 CFR 50, Appendix B. Inspections may be performed together with ASME inservice inspection activities, and they will be designed to ascertain if detrimental aging effects are occurring. In general, inspections will be scheduled to be accomplished no earlier than 10 years prior to the period of extended operation.

Monitoring and Trending

The One-Time Inspection Program is not intended to be a monitoring or trending program; should degradation be encountered, it would be evaluated, and if required, monitored or trended, under the Corrective Action Program.

Acceptance Criteria

Any indication or relevant conditions of degradation detected will be evaluated using defined acceptance criteria, such as minimum wall thickness for pressure boundary piping.

Corrective Actions

Unacceptable inspection findings will be evaluated in accordance with the Corrective Action Program. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The One-Time Inspection Program is a new program. The HNP aging management review process ensures that one-time inspections have been prescribed and developed with consideration of plant and industry operating experience.

NUREG-1801 is based on industry operating experience through January 2005. This program applies to potential aging effects for which there are currently no operating experience indicating the need for an aging management program. Nevertheless, the elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

In response to the supplemental questions provided by the NRC subsequent to the original response, the following was provided:

<u>Supplemental Question 2 Response</u> There are no components that rely on the One-Time Inspection Program to manage Loss of Preload. Loss of Preload due to thermal effects, gasket creep and self-loosening will be managed by the Bolting Integrity Program.

B.2.19 SELECTIVE LEACHING OF MATERIALS PROGRAM

Discussion of Program Elements

Scope of Program

The program will perform examinations to determine whether loss of material due to selective leaching is occurring for those components determined to be potentially susceptible during aging management reviews.

Preventive Action

The program is an inspection/verification program; there is no preventive action.

Parameters Monitored/Inspected

HNP will establish a representative sample set of components to be managed following review of industry technology used to inspect and evaluate the identification and progression of the selective leaching aging mechanism. A qualitative determination of selective leaching will be used in lieu of Brinell hardness testing for components within the scope of this program. Follow-up of unacceptable findings will include engineering evaluation and sample expansion as appropriate. Inspection of components will be scheduled under the Work Management Process to occur just before the beginning of the license renewal period.

Detection of Aging Effects

The occurrence of selective leaching will be determined using an inspection procedure that includes detailed instructions for performing visual inspections and applying qualitative determinations of selective leaching.

Monitoring and Trending

Selective leaching will be managed by one-time inspections of a representative set of components. The program will not rely on monitoring and trending of results.

Acceptance Criteria

Evidence of selective leaching will result in an engineering evaluation and will be tracked as a corrective action. Engineering evaluations for unacceptable degradation will be addressed in accordance with the Corrective Action Program.

Corrective Actions

The Corrective Action Process provides the administrative controls for identifying, documenting, tracking, investigating, correcting, and trending significant conditions adverse to quality. The Corrective Action Program is discussed generically in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Selective Leaching of Materials Program is a new program; therefore, operating experience to verify the effectiveness of the program is not available. A review of plant-specific operating experience has identified one occurrence of selective leaching of materials at HNP involving a naval brass shear pin on a fuel transfer cart used during refueling operations. The pin material consisted of 39% zinc and approximately 60% copper.

NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.20 BURIED PIPING AND TANKS INSPECTION PROGRAM

Discussion of Program Elements

Scope of Program

A new inspection procedure will be developed and will include guidance for condition monitoring of buried piping components/commodity groups that have been identified during aging management reviews.

Preventive Action

HNP has used preventive measures, such as, protective coatings or wrappings on buried steel and cast iron piping applications. The soil environment at HNP is non-aggressive.

Parameters Monitored/Inspected

Detailed procedural requirements for the Program will be developed and will include provisions to perform inspections of components following excavation and to take actions to assess coating damage and potential corrosion, and to employ appropriate care when burying piping components.

Detection of Aging Effects

Inspections will be performed at frequencies consistent with those in NUREG-1801, Rev. 1.

Monitoring and Trending

Site experience with corrosion of system components will be used to plan additional inspections.

Acceptance Criteria

The new procedure governing inspection of buried components will ensure that a coating engineer or other qualified individual assists in the evaluation of coating degradation and will include inspection checklists that will include requirements to document the results of inspections. The extent of degradation found during inspection will determine the nature of the corrective actions taken and the need for and locations of future inspections.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Buried Piping and Tanks Inspection Program is a new program applicable to buried piping at HNP. There is no existing operating experience to validate the effectiveness of this program. 12
At HNP, leaks have occurred in the Fire Protection System and the Potable and Sanitary Water System. These were evaluated and determined not to be caused by age-related degradation. A leak occurred in a buried portion of the Diesel Generator Fuel Oil Storage and Transfer System; however, due to the inaccessible location of the leak, its cause could not be determined. These leaks were evaluated in accordance with the Corrective Action Program. Based on this site experience, it can be concluded that leaks in HNP buried piping have been detected and that appropriate corrective actions have been taken to ensure no loss of component intended function. This experience is not atypical and justifies the use of the 10-year inspection frequency for buried components endorsed by NUREG-1801.

NUREG-1801 is based on industry operating experience through January 2005. Recent industry operating experience has been reviewed for applicability. More recent operating experience is captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

B.2.21 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING PROGRAM

Discussion of Program Elements

Scope of Program

This program is a new program that will be administratively controlled as an augmented ASME Code inservice inspection program. The program will detect cracking in small-bore piping regardless of the aging mechanism causing the cracking and will consider the guidance in EPRI Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)."

Preventive Action

The program is an inspection/verification program; there is no preventive action.

Parameters Monitored/Inspected

The program is intended to detect cracking in small-bore, ASME Class 1 piping less than NPS 4.

Detection of Aging Effects

HNP has not experienced cracking of small-bore piping due to stress corrosion or thermal/mechanical loading.

Monitoring and Trending

Inspections will be performed at a sufficient number of locations to assure an adequate sample. The sample population for the plant-specific program will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore locations. The sample prioritization will consider the potential for mechanical loading as a result of thermal stratification, piping potentially susceptible to IGSCC (i.e., normally stagnant piping), and locations identified for inspection under the RI-ISI program which considers thermal loading from plant cycles and thermal stratification.

Acceptance Criteria

Inspection results will be evaluated in accordance with ASME Code Subsection IWB-3400 and IWB-3131. Additional examinations, if required, will be performed in accordance with IWB-2430.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The industry has experienced unexpected cracking which may have been caused by aging mechanisms not originally considered in the design analyses. Additional events caused by fatigue due to thermal stratification resulted in the issuance of IE Bulletin 88-08 as supplemented. In response to IE Bulletin 88-08, an augmented inspection program was implemented during the first HNP inspection interval. Augmented volumetric examinations were performed on selected piping welds in the charging and safety injection lines considered potentially subject to thermal fatigue. No indications of cracking were found.

This is a new aging management program for the One-Time Inspection of ASME Code Class 1 Small-Bore Piping. There is no existing operating experience to validate the effectiveness of this program. Any future operating experience which may impact the program will be captured through the normal operating experience review process where it is screened for applicability. This process will continue through the period of extended operation.

In response to the supplemental questions provided by the NRC subsequent to the original response, the following was provided:

Supplemental Question 1 Response The industry has experienced unexpected cracking which may have been caused by aging mechanisms not originally considered in the design analyses. Additional events caused by fatigue due to thermal stratification resulted in the issuance of IE Bulletin 88-08 as supplemented. In response to IE Bulletin 88-08, an augmented inspection program was implemented during the first HNP inspection interval.

The examination document file was included on CD provided on 6/25/2007. Refer to file ISI 1-1-1.pdf on the CD.

Supplemental Question 3 Response In the original program prepared by the MRP Thermal Fatigue Task Force identified tasks in the areas of data collection, screening and analysis, inspection, monitoring, maintenance, modifications and preparation of a final guideline for thermal fatigue management. Because of NRC concerns that there were no immediate licensee actions, an interim guideline document (MRP-24) was made available in January 2001. Following issuance of MRP-24, additional testing and evaluations were undertaken to better understand the thermal fatigue mechanisms. This allowed models to be developed to predict susceptibility of lines to thermal fatigue. This enhanced guidance was incorporated into MRP-146. It was the intention of the MRP Task Force to replace the interim guidance in MRP-24 with that in MRP-146 when better guidance became available. This is discussed in the HNP basis document for the One-time Inspection of ASME Code Class 1 Small-Bore Piping Program, which states:

> In June, 2005, during the final development of NUREG-1801, Rev. 1, EPRI Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)" was issued. MRP-146 provides "needed guidelines" and other good practice recommendations for evaluating and inspecting regions in normally stagnant PWR reactor coolant system branch lines. MRP-146 was prepared to meet the objectives of Nuclear Energy Institute (NEI) 03-08, the industry materials initiative promulgated through the auspices of NEI. The issue of thermal fatigue cracking in normally stagnant non-isolable RCS branch piping is managed at HNP as part of the NEI 03-08 Industry Initiative on the Management of Materials Issues. As described above, the MRP-146 guidelines are identified as "needed guidelines" and are therefore characterized under the "Needed Implementation Category." Therefore, in lieu of MRP-24, the HNP One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will consider the guidelines outlined in EPRI Report 1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)". Since MRP-146 replaced the interim guidelines identified in MRP-24, this is not considered an exception to GALL.

Based on the discussion above, the interim guidance from MRP-24 has been incorporated into the enhanced guidelines of MRP-146. In addition, it is concluded that use of the guidance in MRP-146 in lieu of that in MRP-24 does not affect the

NUREG-1801, Section XI.M35, elements of parameters monitored, detection of aging effects, monitoring and trending, and acceptance criteria.

Supplemental Question 4 Response The inspection techniques, frequency, sample size, data collection, and timing of one-time inspection to ensure timely detection of aging effects will be developed as part of the specific implementation details of the HNP One-time Inspection of ASME Code Class 1 Small-Bore Piping Program. The program implementation will include the following.

Inspection Techniques

The program will include one-time volumetric examinations of a sample of ASME Class 1 butt welds for pipe less than NPS 4 and additional inspections in accordance with the ASME Section XI and NRC requirements for socket welds in effect during the period of extended operation.

Frequency

Refer to the Timing discussion below.

Sample Size

The sample population for the volumetric inspections will be at least 10% or will otherwise be based on an inspection plan approved by the NRC. In addition, the program will include controls to ensure that ASME Class 1 socket welds receive an inspection in accordance with the approved ASME Section XI ISI program and NRC requirements during the period of extended operation.

Data Collection

The HNP One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be administratively controlled as an augmented ISI program; data collection will be accomplished in conjunction with the ISI Program.

Timing

The volumetric inspections will be performed as a one-time inspection to be completed prior to the end of, and within the last five years of, the current operating period to ensure timely detection of aging effects. The frequency requirements for socket welds will follow the ASME Section XI ISI and NRC requirements during the period of extended operation.

B.2.24 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS PROGRAM

Discussion of Program Elements

Scope of Program

The program does not inspect components within the scope of License Renewal that have aging effects associated with boric acid corrosion; however, should borated water leakage be identified during inspections under this program, they would be reported and resolved under the Boric Acid Corrosion Program.

Preventive Action

The program is an inspection program; there is no preventive action.

Parameters Monitored/Inspected

Parameters monitored are those that may be detected by visual inspections, such as, cleanliness and freedom from obstructions for flame arrestors.

Detection of Aging Effects

Inspection intervals will be selected to assure detection of degradation prior to loss of intended function using existing administrative controls for scheduling preventive maintenance and surveillance, trending performance of systems and components, and equipment reliability guidance. Identification of paint or coating degradation, such as, blistering, cracking, or the presence of rust or mechanical damage, is an effective method for detection of the occurrence of corrosion on a painted steel surface. Locations of inspections will be selected to be representative of the material/ environment combinations evaluated in the aging management review for the components/commodities. Selection of locations will consider such factors as access, equipment design, radiation levels, and safety of inspectors.

Monitoring and Trending

Administrative controls for the program will ensure: (1) that qualified personnel will perform inspections under work order tasks, (2) that System Engineers confirm the effectiveness of inspection activities by monitoring, reporting, and controlling trending activities, and (3) that System Engineers periodically review the plant-specific operating history and industry operating experience applicable to the system.

Acceptance Criteria

System maintenance history will be periodically reviewed by System Engineers. Material failures or degraded material conditions will require initiation of a report and further evaluation in accordance with the Corrective Action Program.

Corrective Actions

The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program. However, operating experience has been incorporated into License Renewal program activities, and also NUREG-1801 is based on industry operating experience through January 2005. The HNP aging management review process ensures that the program has been prescribed and will be developed with consideration of plant and industry operating experience.

Periodic surveillance and preventive maintenance activities have been in place at HNP since the plant began operation. These activities have proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. There is a demonstrated history of detecting damaged and degraded components and causing their repair or replacement in accordance with the site Corrective Action Program.

B.2.33 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Discussion of Program Elements

Scope of Program

This program includes insulated cables and connections installed in an adverse, localized environment caused by heat or radiation in the presence of oxygen and purposely includes other plant areas for conservatism.

Preventive Action

The program is an inspection program; there is no preventive action.

Parameters Monitored/Inspected

The AMP is a representative sampling program. Accessible electrical cables and connections installed in adverse localized environments are visually inspected for cable and connection jacket surface anomalies.

Detection of Aging Effects

A representative sample of accessible insulated cables and connections installed in adverse localized environments will be inspected at least once every 10 years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking, swelling or surface contamination. The first inspection for license renewal will be completed before the period of extended operation. The HNP program utilizes plant operating experience (OE) to determine the plant areas to be inspected. HNP OE is used to identify past cable failures, cables that exhibited the effects of aging, hot spots, and adverse localized environments. Part of this OE review includes conversations with maintenance personnel and the use of environmental surveys. Based on this review of OE, the plant areas to be inspected become localized in nature, consisting of a limited area (or subset) of a much larger plant area or zone. The sample selection of cables and connections inspected within the limited plant area bound all cables and connections in the area since the inspection focuses on the worst case environments.

Monitoring and Trending

Trending of discrepancies is performed as required in accordance with the HNP Corrective Action Program.

Acceptance Criteria

The program requires that accessible cables and connections be free from unacceptable, visual indications of jacket surface anomalies, which suggest that conductor insulation or connection degradation exists.

Corrective Actions

Unacceptable visual indications of cable and connection jacket surface anomalies will be subject to an engineering evaluation. This evaluation is to consider the age and operating environment of the component, as well as the severity of the anomaly, and whether such an anomaly has previously been correlated to degradation of conductor insulation or connections. Corrective actions as required will be implemented through the Corrective Action Program and may include, but are not limited to, testing, shielding, or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination will be made as to whether this same condition or situation could be applicable to other accessible or inaccessible insulated cables and connections
and whether the sample size should be expanded. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The program is a new program with no site-specific operating experience history. However, as noted in NUREG-1801, industry operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.34 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

Discussion of Program Elements

Scope of Program

This program applies to electrical cables and connections (i.e., cable system) used in circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation that are subject to aging management review.

Preventive Action

This is a surveillance testing program and no actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The parameters monitored are determined from the specific calibrations, surveillances, or testing performed and are based on the specific instrumentation circuit under surveillance or being calibrated as documented in HNP procedures.

Detection of Aging Effects

For radiation monitoring circuits, HNP will review the results or findings of calibration or surveillance tests in order to detect severe aging degradation prior to the loss of the cable and connection intended function. Calibration or surveillance results that fail to meet acceptance criteria will be reviewed for aging effects when the results are available. For nuclear instrumentation circuits, the cable system will be tested as an alternative to the review of calibration or surveillance results described above.

Monitoring and Trending

Trending actions are not part of this HNP AMP because the ability to trend test results is dependent on the specific type of test chosen. However, the trending of discrepancies is performed as required in accordance with the HNP Corrective

Action Program.

Acceptance Criteria

Acceptance criteria will be set out in the applicable plant procedures.

Corrective Actions

Corrective actions will be implemented when calibration or surveillance results or findings of surveillances do not meet the acceptance criteria through the Corrective Action Program and will include an engineering evaluation to ensure that the intended functions of the electrical cable system can be maintained consistent with the CLB. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new program; there is no existing operating experience to validate the effectiveness of this program. However, as noted in NUREG-1801, industry operating experience has shown that exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced IR. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.35 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Discussion of Program Elements

Scope of Program

The program applies to inaccessible (i.e., in conduit or direct buried) medium voltage cables.

Preventive Action

Periodic actions are taken to prevent non-EQ medium voltage cables from being exposed to significant moisture. This activity includes inspecting for water collection in cable manholes and draining water as needed.

Parameters Monitored/Inspected

In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation.

Detection of Aging Effects

The first testing for license renewal will be completed before the period of extended operation. The manhole inspection frequency will be based on actual field data and shall not exceed two years. The first inspection for license renewal will be completed before the period of extended operation.

Monitoring and Trending

Trending actions are not part of the program. However, trending of discrepancies is performed as required in accordance with the Corrective Action Program.

Acceptance Criteria

The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

Corrective Actions

Engineering will perform an evaluation when acceptance criteria are not met in order to ensure that the License Renewal intended functions will be maintained. Such an evaluation will consider the significance of the test results, the operability of the component, the potential root causes, the corrective actions required, and a determination whether the condition is applicable to other in-scope medium-voltage cables. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new Aging Management Program for non-EQ inaccessible medium-voltage cables. There is no existing operating experience to validate the effectiveness of this program. As noted in NUREG-1801, industry operating experience has shown that cross linked polyethylene or high molecular weight polyethylene insulation materials are most susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage; for example, treeing is much less prevalent in 4KV cables than those operated at 13KV or 33KV. Also, minimizing exposure to moisture minimizes the potential for the development of water treeing.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of

OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

B.2.36 METAL ENCLOSED BUS PROGRAM

Discussion of Program Elements

Scope of Program

The program applies to MEBs within the scope of License Renewal.

Preventive Action

This is an inspection program; no actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The program will check a sample of accessible bolted connections for loose connection. Alternatively, bolted connections covered with heat shrink tape, sleeving, insulating boots, etc., may be visually inspected for insulation material surface anomalies.

Detection of Aging Effects

The program will check a sample of accessible bolted connections for loose connection by using thermography or by measuring connection resistance using a low range ohmmeter. Thermography will be performed while the bus is energized and loaded. Connection resistance measurements will be performed while the bus is de-energized and will consist of a sampling of accessible bolted connections. MEB internal surfaces will be visually inspected for aging degradation of insulating material and for foreign debris and excessive dust buildup, and evidence of moisture intrusion. Bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. Internal bus supports will be visually inspected for structural integrity and signs of cracks. This program will be completed before the period of extended operation and every 10 years thereafter.

As an alternative to thermography or measuring connection resistance of bolted connections, for the accessible bolted connections that are covered with heat shrink tape, sleeving, insulating boots, etc., HNP may use visual inspection of insulation material to detect surface anomalies, such as discoloration, cracking, chipping or surface contamination. If this alternative visual inspection is used to check bolted connections, the first inspection will be completed before the period of extended operation and every five years thereafter.

Monitoring and Trending

Trending actions are not part of the program, because the ability to trend inspection results is limited. However, trending of discrepancies is performed as required in accordance with the HNP Corrective Action Program.

Acceptance Criteria

Bolted connections are to be below the maximum allowed temperature for the application when thermography is used. If measuring connection resistance using a low range ohmmeter, microhm values of accessible bolted connections shall not exceed the high levels of the normal range indicated in the manufacturer's published data. If manufacturer's data is not available, the accessible bolted connection must be within 25% of the lowest value of the other sample connections. MEBs are to be free from unacceptable visual indications of surface anomalies, which suggest that conductor insulation degradation exists. In addition, no unacceptable indication of corrosion, cracks, foreign debris, excessive dust buildup or evidence of moisture intrusion is to exist. When the visual inspection alternative for bolted connections is used, the absence of discoloration, cracking, chipping, or surface contamination will provide positive indication that the bolted connections are not loose.

Corrective Actions

Corrective actions may include but are not limited to cleaning, drying, increased inspection frequency, replacement, or repair of the affected MEB components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible MEBs. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

This is a new Aging Management Program for MEB. There is no existing site specific operating experience to validate the effectiveness of this program. Industry experience has shown that failures have occurred on MEBs caused by cracked insulation and moisture or debris buildup internal to the MEB. Industry experience has also shown that MEB exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connections related to the repeated cycling of connected loads or of the ambient temperature environment. This phenomenon can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating or ambient heating) that are routinely cycled.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

(The following text enhances new Subsection B.2.37 provided in conjunction with the response to Audit Question B.2.37-XI.E6-RM-02. Replace the **Operating Experience** section from the response to Audit Question B.2.37-XI.E6-RM-02 with the following.)

B.2.37 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Discussion of Program Elements

Scope of Program

The scope of this program includes only external cable connections terminating at an active device such as motor, motor control center, switchgear or of a passive device such as a fuse cabinet. Wiring connections internal to an active assembly installed by manufacturers are considered a part of the active assembly and therefore are not within the scope of this program.

Preventive Action

No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

The program will focus on the metallic portions of non-EQ cable connections exposed to inside and outside air environments. The monitoring includes loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation. A representative sample of electrical cable connections will be tested and will consider the following factors: application (high, medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selected will be provided.

Detection of Aging Effects

This program will be implemented as a one-time inspection on a representative sample of non-EQ cable connections within the scope of license renewal prior to the period of extended operation. Inspection methods may include thermography, contact resistance testing, bridge balance testing, or other appropriate testing methods. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Monitoring and Trending

Trending actions are not part of the program because this is a one-time inspection program. However, trending of discrepancies is performed as required in accordance with the Corrective Action Program.

Acceptance Criteria

The acceptance criteria for each test are defined by the specific type of test performed and the specific type of cable connections tested.

Corrective Actions

An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the cable connections can be maintained. Such an evaluation will consider the the operability of the component, the potential root causes for not meeting the test acceptance criteria, the corrective action necessary, and possible changes to the one-time inspection program such as increased frequency and sample size. When an unacceptable condition or situation is identified, a determination is made on whether the same condition or situation is applicable to other in-scope cable connections not tested. The Corrective Action Program is discussed further in Subsection B.1.3.

Confirmation Process

This program element is addressed in Subsection B.1.3.

Administrative Controls

This program element is addressed in Subsection B.1.3.

Operating Experience

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program. Industry operating experience has shown that loosening of connections and corrosion of connections are aging mechanisms that could lead to a loss of electrical continuity without proper installation and maintenance activities. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Plant-specific and industry-wide OE was considered in the development of the electrical programs in LRA Appendix B. Industry operating experience that forms the basis for these programs is included in the OE element of the corresponding NUREG-1801, Chapter XI, Programs. Plant-specific OE was reviewed to ensure that the NUREG-1801, Chapter XI, Programs will be effective aging management programs for the period of extended operation. This review confirms that the OE discussed in the NUREG-1801, Chapter XI, Programs is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

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Question No: B.2.1-JM-02

NRC Request:
The "parameter monitored/inspected" program attribute for HNP bases document HNP-P/LR-0606, "License Renewal Aging Management Program Description of the ASME Section XI, Subsection IWB, IWC and IWD, Inservice Inspection Program," does not identify which aging effects the program monitors for. Clarify which aging effects are within the scope of the "parameter monitored/inspected" program attribute for the ASME Section XI, Subsection IWB, IWC and IWD, Inservice Inspection Program.

HNP Response:
As shown on Attachment 1, page 3 of the basis document, the following aging effects and mechanisms managed by this program are:

Cracking due To SCC
Loss of Fracture Toughness due To Thermal Embrittlement
Loss of Material due To Crevice Corrosion
Loss of Material due To General Corrosion
Loss of Material due To Pitting Corrosion
Loss of Material due To Wear

Question No: B.2.1-JM-04

NRC Request:
The "detection of aging effects" program attribute for HNP bases document HNP-P/LR-0606, "License Renewal Aging Management Program Description of the ASME Section XI, Subsection IWB, IWC and IWD, Inservice Inspection Program," does not identify which Non-destructive examination (NDE) methods or visual examination methods are within the scope of the AMP. Clarify which NDE methods and visual examination methods are within the scope of the "detection of aging effects" program attribute for the ASME Section XI, Subsection IWB, IWC and IWD, Inservice Inspection Program. Clarify how the inspection frequencies and sample sizes for the applicable examinations are established.

HNP Response:
NDE methods and visual examination methods within the scope of the "detection of aging effects" program attribute are as specificed in the American Society of Mechanical Engineers (ASME) Code, Section XI Subsections IWB, IWC and IWD, Inservice Inspection Program, 1989 Edition. The inspection frequencies and sample sizes for the applicable examinations are established by the American Society of Mechanical Engineers (ASME) Code, Section XI Subsections IWB, IWC and IWD, Inservice Inspection Program, 1989 Edition.

As noted in the description of the NUREG-1801 Section XI.M1 program, 10 CFR 50.55a governs the application of Codes and Standards. In conformance with 10CFR50.55a(g)(4)(ii), the ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.

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Question No: B.2.3-JM-01

NRC Request:
The "parameter monitored/inspected" program attribute for HNP bases document HNP-P/LR-0619, "License Renewal Aging Management Program Description of the Reactor Head Closure Studs Program," does not identify which aging effects the program monitors for. Clarify which aging effects are within the scope of the "parameter monitored/inspected" program attribute for the Reactor Head Closure Studs Program.

HNP Response:
As shown on Attachment 1, page 1 of the basis document, the following aging effects and mechanisms managed by this program are:

Cracking due To SCC
Loss of Material due To Wear

Question No: B.2.4-ZL-01

NRC Request:
The program description section of the Boric Acid Corrosion Program states that this 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.

1. What were HNP's response to the reactor vessel head penetrations leakage and reactor head boric acid corrosion problems occurred at the Davis Besse?

2. Has HNP incorporated NRC communications such as Bulletins, orders, GLs, and INs that are related to the boric acid corrosion problem into the Boric Acid Corrosion Program?

Amend the response to incorporate Bulletins 2003-02 and 2004-01 and HNP's responses to these documents as applicable documents for the BAC program.

Address impact of HNP Correspondence HNP-07-015, January 31, 2007 and HNP-07-026, February 27, 2007 on HNP commitments with respect to Bulletin 2004-01 responses.

HNP Response:
Vessel Head Penetration and Davis-Besse Response

HNP responded to the actions required by NRC generic communications addressing reactor vessel head penetration leakage and issues resulting from the Davis-Besse Nuclear Power Station corrosion problems by providing inspection results, including the results of a bare metal visual inspection of the reactor vessel head via the following correspondence:

1. HNP letter, Serial HNP-97-087: "Response to NRC Generic Letter 97-01,'Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,'" dated April 25, 1997.

2. HNP letter, Serial HNP-97-152: "Response to NRC Generic Letter 97-01,'Degradation of Control Rod Drive

Mechanism Nozzle and Other Vessel Closure Head Penetrations,'" dated July 29, 1997.

3. HNP letter, Serial HNP-99-006: "120 Day Response to NRC Request for Additional Information Regarding Generic Letter 97-01, 'Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations,'" dated January 26, 1999.

4. HNP letter, Serial HNP-01-124: "Response to NRC Bulletin 2001-01, 'Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,'" dated September 4, 2001.

5. HNP letter, Serial HNP-02-009: "Supplemental Response to NRC Bulletin 2001-01, 'Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,'" dated January 28, 2002.

6. HNP letter, Serial HNP-02-052: "15-Day Response to NRC Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,'" dated April 2, 2002.

7. HNP letter, Serial HNP-02-063: "60-Day Response to NRC Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,'" dated May 15, 2002.

8. HNP letter, Serial HNP-02-164: "Request for Additional Information, Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,'" dated January 24, 2003.

9. HNP letter, Serial HNP-02-118: "30-Day Response to NRC Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated September 12, 2002.

10. HNP letter, Serial HNP-03-118: "90-Day Response to NRC Bulletin 2003-02, Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," dated November 13, 2003.

11. HNP letter, Serial HNP-03-070: "Sixty-Day Report in Accordance with NRC Order for Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, Inspection of RPV Head During Refueling Outage," dated July 16, 2003.

No boric acid deposits or head degradation have been found due to reactor pressure vessel head penetration nozzle leakage.

Update of Program Based on Generic Communications

The HNP Boric Acid Corrosion Program was implemented in response to Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." Currently, the program activities are governed by a Corporate, fleet-wide program, and the program manager is responsible for reviewing industry operating experience, such as NRC generic communications, and updating the program as necessary. Through its responses to NRC Bulletins and Orders, HNP has confirmed that the scope of the Boric Acid Corrosion Program, as implemented through plant procedures, has appropriately addressed boric acid leakage detection issues associated with the following:

1. GL 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations."
2. NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."
3. NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reatcor Coolant Pressure Boundary Integrity." (The response to this Bulletin included a review of the effectiveness of the Boric Acid Corrosion Program at HNP.)

Bulletins 2003-02 and 2004-01

Bulletin 2003-02

The NRC issued Bulletin 2003-02 requesting specific information concerning licensees' Reactor Pressure Vessel lower head penetration inspection program. Note: Degradation of nozzles associated with the reactor vessel lower head penetrations has the potential to result in boric acid leakage.

HNP responded to the subject Bulletin via the following correspondence:

Letter from J. Scarola (PE) to NRC, Serial HNP-03-118: "90-Day Response to NRC Bulletin 2003-02, for Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," dated November 13, 2003.

HNP letter, Serial HNP-04-154: 60-Day Summary Report

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NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, Request (2),
dated January 13, 2005.

Bulletin 2004-01

The NRC issued this bulletin to:

- Advise PWR licensees that current methods of inspecting Alloy 2/182/600 materials may need to be supplemented.
- Request PWR addressees to provide the NRC with information related to the pressurizer penetrations and steam space piping connections materials of fabrication.
- Request PWR licensees to provide the NRC with information related to the inspections performed to ensure that degradation of Alloy 82/182/600 materials will be identified, adequately characterized, and repaired.
- Require PWR addresses to provide a written response to the NRC in accordance with the provisions of Section 50.54(f) of Title 10 of the Code of Federal Regulations (10 CFR 50.54(f)).

HNP responded to the actions required by NRC Bulletin 2004-01 via the following correspondence:

1. HNP letter, Serial HNP-04-097: 60-Day Response to NRC Bulletin 2004-01 for the Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, July 27, 2004.
2. HNP letter, Serial HNP-04-134: Response to Request for Additional Information Regarding NRC Bulletin 2004-01 for Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, October 29, 2004.
3. HNP letter, Serial HNP-04-166: 60-Day Report
NRC Bulletin 2004-01, Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, January 14, 2005.

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Impact of HNP Corrspondence HNP-07-015, January 31, 2007 and HNP-07-026, February 27, 2007 on HNP commitments with respect to Bulletin 2004-01 responses

HNP commitments with respect to Bulletin 2004-01 responses

The actions committed to by Harris Nuclear Plant (HNP) in response to Bulletin 2004-01 are identified in HNP letter, Serial HNP-04-097, dated July 27, 2004. Any other actions discussed in the subject correspondence represent intended or planned actions by HNP. They are described for the NRC's information and are not regulatory commitments.

No.	Commitments	Scheduled Completion Dates
1	HNP will perform bare metal visual inspection exams on the pressurizer penetration and steam space piping connections listed in Table C of letter HNP-04-097 during the next refueling outage (RFO-12) scheduled for the Fall 2004 and during every refueling outage until mitigation is performed, additional guidance is provided by the Materials Reliability Program, or new Code or regulatory requirements are imposed.	RFO-12 (Fall 2004) and every refueling outage per the commitment description.

Impact of HNP Corrspondence HNP-07-015, January 31, 2007 and HNP-07-026, February 27, 2007

In October 2006, while performing inspections of pressurizer (PZR) Alloy 82/182 butt welds in accordance with MRP-139, a PWR licensee discovered several circumferential indications in the PZR surge, safety, and relief nozzles. Because of the potential importance of this issue, HNP commited to the following:

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No.	Commitments	Scheduled Completion Date
1.	HNP will mitigate the pressurizer Alloy 82/182 butt welds by installing full structural weld overlays on these welds and will inspect post-overlay during refueling outage 14 (RFO-14) in the Fall 2007.	End of RFO-14 (Fall 2007)

Pursuant to discussion with the NRC by phone on February 20, 2007, leakage monitoring described in HNP Correspondence HNP-07-015 is supplemented as described in HNP Correspondence, ML070650468, Serial: HNP-07-026, "Inspection and Mitigation of Alloy 82/182 Pressurizer Butt Welds." HNP Correspondence HNP-07-026 provides the following original, as well as new or revised commitments:

No.	Commitments	Scheduled Completion Dates
1	HNP will mitigate the pressurizer Alloy 82/182 butt welds by installing full structural weld overlays on these welds and will inspect post-overlay during refueling outage 14 (RFO-14) in the fall 2007.	End of RFO-14 (fall 2007)
2	HNP will monitor unidentified RCS leakage daily while the plant is in Modes 1-3 during stable plant conditions until mitigation of the pressurizer Alloy 82/182 butt welds scheduled in RFO-14 (fall 2007).	Beginning on March 6, 2007 until mitigation of the pressurizer Alloy 82/182 butt welds
3	If unidentified RCS leakage should increase	Beginning on

	by 0.1 gpm in the daily measurement to the mean, sustained for 72 hours with at least 0.1 gpm not confirmed from sources other than the pressurizer nozzle welds, then the unit will be placed in Mode 3 within six (6) hours and Mode 5 within the next 36 hours, and a bare metal visual inspection of the unmitigated pressurizer surge, spray, safety, and relief nozzle butt welds and safe end butt welds containing Alloy 82/182 material will be performed.	March 6, 2007 until mitigation of the pressurizer Alloy 82/182 butt welds
4	If unidentified RCS leakage should increase by 0.25 gpm above the baseline, sustained for 72 hours with at least 0.25 gpm not confirmed from sources other than the pressurizer nozzle welds, then the unit will be placed in Mode 3 within six (6) hours and Mode 5 within the next 36 hours, and a bare metal visual inspection of the unmitigated pressurizer surge, spray, safety, and relief nozzle butt welds and safe end butt welds containing Alloy 82/182 material will be performed.	Beginning on March 6, 2007 until mitigation of the pressurizer Alloy 82/182 butt welds
5	HNP will report information of any corrective or mitigative actions taken, and if HNP shuts down due to unidentified RCS leakage (i.e., the action of Response 4 of this letter), then HNP will report bare metal visual inspection results.	Within 60 days of unit restart

Discussion:

Based upon a review of the subject correspondence, no unidentified changes to commitments pursuant to the Bulletin 2004-01 responses were found. The commitments contained in the Bulletin 2004-01 responses are in effect "...until mitigation is performed, additional guidance is provided by the Materials Reliability Program, or new Code or regulatory requirements are imposed." Since the HNP-07-015, January 31, 2007 and HNP-07-026, February 27, 2007 commitments include the mitigation actions committed to in the Bulletin 2004-01 responses, the HNP-07-015, January 31, 2007 and HNP-07-026, February 27, 2007 commitments subsume the actions committed to in the Bulletin 2004-01 responses.

As discussed in NRC Correspondence, Confirmatory Action Letter - Shearon Harris Nuclear Power Plant, Unit No. 1, dated March 22, 2007 (ML070780413):

"In your letter dated February 27, 2007 (Agencywide Documents Access & Management System (ADAMS) Accession Number ML070650468), you described actions you will take at Shearon Harris Nuclear Power Plant, Unit 1 for the pressurizer dissimilar metal butt welds containing Alloy 82/182 material. These commitments address: 1) completion schedules for inspection/mitigation of the welds; 2) RCS leak monitoring frequency, action levels, and actions; and 3) reporting requirements.

The NRC staff has reviewed these actions and commitments and agrees the actions and commitments are appropriate to address the potential of PWSCC of the applicable pressurizer dissimilar metal butt welds containing Alloy 82/182 material."

Conclusion:

As indicated above, the NRC has reviewed the subject correspondence and agrees the actions and commitments are appropriate to address the potential of PWSCC of the applicable pressurizer dissimilar metal butt welds containing Alloy 82/182 material. Based upon the review of the subject correspondence herein, no unidentified changes to commitments were found.

Question No: B.2.4-ZL-10

NRC Request:
The HNP technical specifications establish the leakage limits for reactor coolant pressure boundary leakage, unidentified RCS leakage, and identified RCS leakage that is not RCPB leakage. Clarify what type of activities or actions are taken to distinguish between these types of leakage upon discovery of RCS leakage, and clarify whether these activities are incorporated into the implementation procedure for the Boric Acid Corrosion Program.

HNP Response:
Attachment 2 to HNP Correspondence, ML070370405, Serial: HNP-07-015, "Shearon Harris Nuclear Power Plant, Unit No. 1 Inspection and Mitigation of Alloy 82/182 Pressurizer Butt Welds," provides a discussion of reactor coolant system (RCS) leakage monitoring. Pursuant to discussion with the NRC by phone on February 20, 2007, leakage monitoring is supplemented as described in HNP Correspondence, ML070650468, Serial: HNP-07-026, "Inspection and Mitigation of Alloy 82/182 Pressurizer Butt Welds." Additionally, HNP FSAR, Section 5.2.5, "Detection of Leakage Through Reactor Coolant Pressure Boundary," provides a detailed discussion of this topic.

As discussed in NRC Correspondence, Confirmatory Action Letter - Shearon Harris Nuclear Power Plant, Unit No. 1, dated March 22, 2007 (ML070780413):

> "In your letter dated February 27, 2007 (Agencywide Documents Access & Management System (ADAMS) Accession Number ML070650468), you described actions you will take at Shearon Harris Nuclear Power Plant, Unit 1 for the pressurizer dissimilar metal butt welds containing Alloy 82/182 material. These commitments address: 1) completion schedules for inspection/mitigation of the welds; 2) RCS leak monitoring frequency, action levels, and actions; and 3) reporting requirements.
>
> The NRC staff has reviewed these actions and commitments and agrees the actions and commitments are appropriate to address the potential of PWSCC of the applicable pressurizer dissimilar metal butt welds containing Alloy 82/182 material."

Procedures that implement HNP Technical Specification requirements are not considered License Renewal implementing documents because the requirements to perform these activities are associated with the HNP Technical Specifications and will not undergo change without appropriate review.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.5-FS-01

NRC Request:

1. Revised Order EA-03-009, Section IV.C.(3), for those plants in the Low category, requires that the a bare metal visual examination, meeting the requirements of Revised Order EA-03-009, Paragraph IV.C.(5)(a), be performed every third refueling outage or every five years. The Order also requires that plant in the low category perform nonvisual NDE (Ultrasonic, Eddy Current/dye penetrant testing, or a combination these examinations) at least every refueling or every 7 years. The program description part of this AMP states that HNP has completed the BMV inspection during RFO 11 and the calculation currently projects low susceptibility ranking into the period of extended operations. The HNP Operating Experience discussion has not provided any additional information for the bare metal visual examination results. Also, it is not clear whether nonvisual NDE examination has been performed or is scheduled to be performed.

    a. Please briefly discuss the results of the BMV examination. Explain how these results meet the requirements of the Revised Order EA-03-009.

    b. What is the HNP's calculated effective degradation years (EDY) for the completed plant cycles. Also, what is EDY projected value into the period of extended operation?

c. Clarify whether nonvisual NDE has been performed during the previous refueling outages. If so, are the results meet the requirements of the revised Order EA-03-009?

HNP Response:

a. The Order requires that a 100% Bare Metal Visual (BMV) examination of the Reactor Pressure Vessel (RPV) head surface be performed. HNP performed a 100% BMV examination during RFO-11. The BMV examination revealed no evidence of leakage. The detailed inspection report was provided to the NRC within 60 days as required by the Order. This report is filed under ADAMS accession number ML032040301. The next BMV examination will be performed in RFO-14, which is scheduled to occur in the Fall 2007 in accordance with the Order.

b. The actual calculated EDY through Cycle 12 is 2.521. Although HNP has completed 13 plant cycles, the calculation of actual EDY through Cycle 13 has not yet been completed. However, the projected EDY through Cycle 13 is 2.76. The period of extended operation begins during Cycle 27. The projected EDY through Cycle 27 into the period of extended operation is 6.16 ("low" category). The category is projected to remain "low" through operating Cycle 34. Beginning with Cycle 35, the ranking is projected to be "moderate" (EDY > 8) through Cycle 40 (60-years of operation). The projected EDY through Cycle 40 is 9.34. HNP will re-characterize the susceptibility category to "moderate" or "high" as appropriate in accordance with the Order and inspections and examinations will be performed as required.

c. During RFO-13, the reactor vessel head penetrations were examined using nonvisual NDE to satisfy the requirements of the Order in Section IV.C.(5)(b) and to provide a baseline for future examinations. No evidence of PWSCC was identified by these examinations. The detailed inspection report was provided to the NRC within 60 days as required by the Order. This report is filed under ADAMS accession number ML062010187.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.5-FS-03

NRC Request:
LRA Section 2.5 stated that the Parameters Monitored/Inspected elements of the program will be enhanced by enhancing Inservice Inspection Program procedure to include augmented inspections required by NRC Order EA-03-009 (as amended). Please provide additional information on the augmented inspection that will be added to the ISI program procedures.

HNP Response:
The subject Inservice Inspection Program procedure provides administrative controls for the HNP Inservice Inspection (ISI) Program. This site level procedure identifies augmented inspection programs which will be done under the controls of the ISI program. All of the inspections to be performed as required by Revised NRC Order EA-03-009 will be done as "augmented inspections" under the administrative controls of the HNP ISI Program. This enhancement was to clarify in the program procedure that inspections required by NRC Order EA-03-009 will be done as "augmented inspections" under the ISI Program. Subsequent to submittal of the HNP License Renewal Application, the ISI program procedure has been revised to incorporate this enhancement.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.6-FS-01

NRC Request:
Please clarify whether any component/commodity has been screened out, from being managed by this program, based on the screening criteria in the Christopher Grimes letter dated May 19, 2000. If so, please explain and provide supporting documents.

HNP Response:
No components have been screened out from being managed by this program based on the screening criteria in the Christopher Grimes letter dated May 19, 2000.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.6-FS-02

NRC Request:
The □Corrective Actions□ element of the GALL XI.M13 states that □Repair is performed in conformance with IWA-4000 and IWB-4000, and replacement in accordance with IWA-7000 and IWB-7000.□ However, LRA bases document for B.2.6 AMP (HNP-P/LR-0622) states that □Repairs and replacements will be performed in accordance with Subsection IWA as required by the Code. This appears to be an exception to the GALL corrective actions elements. Please explain this inconsistency. Also, LRA B.2.6 claims consistency (without any exception) with GALL XI.M13, please clarify whether the LRA will be revised to address the above mentioned exception.

HNP Response:
Currently, HNP uses the 1989 Edition of ASME Section XI (the Code) for determining Repair/Replacement requirements. Repairs are performed in accordance with Article IWA-4000 and the corresponding IWX-4000 of the IWB/C/D portions of the code. Replacements are performed in accordance with Article IWA-7000 and the corresponding IWX-7000 of the IWB/C/D portions of the code.

HNP is currently updating the ISI Program to the ASME Section XI, 2001 edition with addenda through 2003, per 10CFR50.55a(g)(4)(ii). In this edition/addenda of the Code, Repair/Replacement Activities are controlled under Article IWA-4000. Section XI states, "The term repair/replacement activity includes those activities previously known as repair, replacement, modification, or, alteration." All of the IWX-4000 and IWX-7000 articles of Section XI have been incorporated into IWA-4000. The LRA will be amended to identify this as an exception.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.8-CM-01

NRC Request:
The Bolting Integrity Program identifies that structural and support bolting at HNP does not include any bolting that is considered to be high strength. Identify all materials that are to fabricate bolting at HNP and define the criteria that are used by CPL to screen a particular bolting material in as "high-strength"or "not high-strength."

HNP Response:
Structural and support bolting materials at HNP subject to License Renewal aging management review are available for review at HNP during the upcoming NRC site audits/inspections. As stated in Section B.2.8 of the LRA, the Structures Monitoring Program and the ASME Section XI Inservice Inspection, Subsection IWF Program are credited for aging management of structural bolting.

The criterion used by HNP to screen structural bolting material as "high-strength" is the criterion provided in Section XI.M18 (page XI M-65, element 3) of NUREG-1801 which discusses high strength bolts as having actual yield strength greater than or equal to 150 ksi.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.8-CM-02

NRC Request:

Provide additional examples of Bolting Integrity Program operating experience.

HNP Response:

AR 1151 - PLANT LEAK REDUCTION - 05/14/1997

1. Event Description

Technical Specification 6.8.4 requires that leakage on various systems such as Residual Heat Removal (RHR), Chemical Volume Control System (CVCS), Safety Injection, be reduced to levels which are as low as practical. Review of the work histories on the CVCS and RHR systems indicate that there are numerous bolted joint assemblies (valve to bonnet, flanged valves, flanged heat exchangers, etc.) which have reoccurring leaks. The general repair approach for such leaks is to replace the gasket. Generic guidance is needed to eliminate these repetitive leaks.

2. Action taken

The following improvements have been made related to leaks at HNP:

1. Maintenance maintains a leak list which tracks active leaks and develops a schedule for repair. This list is the accountability of the Maintenance Fix It Now (FIN) Team supervisor and FIN Team resources are used to make repairs where possible. As a result, most leak repairs bypass the 12-week scheduling process and are repaired in a relatively short period of time. An exception would be leaks that require an outage to repair or an LCO entry.
2. Maintenance requires an initiation of an Action Request (AR) for maintenance rework items. Typically rework is defined as re-performance of a corrective maintenance work order within one year of initial repair. The purpose of the AR is to identify corrective actions to prevent rework on the component in the future. This reduces the repetitiveness of repairs.
3. Engineering has reorganized such that the Rapid Response Team provides component specialists to work with

Maintenance to resolve long-standing Maintenance issues. Any leaks with frequent recurrence will be identified to the valve component engineer to identify actions needed to permanently resolve.

Based on the above, current site practices are satisfactory and no additional actions are required.

AR 25929 - Leak in Fire Protection Line near Turbine Building - 05/25/2001

1. Event Description

Fire pump start was noted with no alarms indicating flow in any building. Operations personnel began a walk down and water was noted coming from the ground on the east side of the turbine building near the deluge valve room. Subsequent to the identification of the leak, the area near the East side of the Turbine building was excavated. The leak was confirmed to be a dislocation of the 12" 90 degree elbow.

2. Cause

Disassembly of the elbow joint found several of the mechanical flange bolts failed allowing separation of the pipe to elbow joint. Information from the Harris Center Metallurgical Lab review indicated the bolts failed in tension. Engineering review describe the most likely cause being a minor leak at the elbow allowed the immediate area to become softened. Recent work in the area then drove equipment over the location and most probably caused some misalignment. Subsequent pump starts thrust the elbow sufficiently to overstress the bolting and cause a tension failure.

3. Action taken

Corrective actions include replacement of the failed components, realignment of the installed piping and installation of a thrust block to reduce the load on the elbow during pump starts.

AR 48782 - Three Flange Bolts Missing for Valve 1CW-28 - 09/21/2001

1. Event Description

Three 2-1/4" diameter bolts were found missing from the lower flange of expansion joint 7CW-J6-1. These bolts are used to attach the lower flange of the expansion joint to the upper body of valve 1CW-28. These three bolts were found missing when personnel were preparing to remove the existing expansion joint.

Harris Nuclear Plant License Renewal Audit Question and Response Database

2. Cause

These three bolts have evidently been missing since the expansion joints were originally installed. The expansion joint may have been slightly misaligned which caused the female threaded holes in the valve body to be partially obscured. This would have made it very difficult to install the bolts.

3. Action taken

The expansion joint adjacent to valve 1CW-28 was replaced in Refueling Outage 10. At this time the bolt holes were cleaned and found to be suitable for reuse. New bolts were inserted into these holes and torqued to fasten the expansion joint above. The expansion joint is now properly fastened.

AR 84089 - CORRODED FLANGE BOLTS ON UNDERGROUND FIRE SUPPRESSION PIPING - 02/07/2003

1. Event Description

While implementing a work order to excavate a suspected underground leak on the fire protection piping, the 3" piping off the jockey fire pump discharge was found to be leaking at a mechanical joint. Some of the carbon steel bolts used to connect the flanges together were found to be extremely corroded.

2. Cause

Several of the corroded bolts were sent to the Harris Center Metallurgical Lab for failure analysis. The findings were that the bolts had corroded primarily due to a lack of protective coatings. All buried mechanical joints are required to have a protective coating applied. These joints did not appear to have any substantial application of protective coating. The corroded flange bolts were caused by inadequate application of protective coatings during original construction.

3. Action taken

Additional mechanical joints were uncovered during the excavation. All the corroded bolts were replaced. Flanges and rubber seal gaskets were also replaced. After reassembly, all of the mechanical joints were coated prior to backfill of the excavation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.9-FS-01

NRC Request:
How many SG tubes in each replaced SG has been plugged or repaired? What are the SG tube repair methods (plugging, sleeving, kinetic expansion, etc.) that are reviewed and accepted by the NRC staff under this program?

HNP Response:
As discussed in page Page AI-3 of 3 of HNP correspondence "One-Year Special Report Steam Generator Tube Inservice Inspection Results," Serial HNP-07-039, April 20, 2007, the following numbers of SG tubes have been plugged to-date:

Steam Generator A: 3
Steam Generator B: 1
Steam Generator C: 3

As discussed in the following NRC correspondence, "SHEARON HARRIS NUCLEAR POWER PLANT, UNIT I - ISSUANCE OF AMENDMENT ON STEAM GENERATOR TUBE SURVEILLANCE PROGRAM," dated March 16, 2007, referring to Page 6-19e:

3. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.9-FS-02

NRC Request:
What are the major differences between NEI 97-06 Rev 1 and Rev 2. How these differences affect the following elements of the program: scope of program, preventive actions, detection of aging effects, and monitoring and trending elements.

HNP Response:
In the HNP response[1] to NRC Generic Letter 2006-01[2], HNP committed to submit a request to modify the Steam Generator portion of the Technical Specifications that would be consistent with Technical Specification Task Force, Improved Standard Technical Specifications Change Traveller, Steam Generator Tube Integrity (TSTF-449), Revision 4[3]. The NRC reviewed and approved Revision 4 to TSTF-449, as stated in Generic Letter 2006-01.

HNP submitted an application for Technical Specification improvement regarding steam generator tube integrity based on TSTF-449, Revision 4, dated May 23, 2006[4], as supplemented by letters dated October 3, 2006[5] and October 24, 2006[6]. The NRC approved this request via correspondence dated March 16, 2007[7]. NEI correspondence transmitting NEI 97-06, Revision 2[8] states that NEI 97-06, Revision 2 is consistent with TSTF-449, Revision 4 and also incorporates additional changes developed by the industry as part of the continuing effort to improve steam generator program guidance. NEI 97-06, Revision 2 summarizes changes between NEI 97-06, Revision 1 and NEI 97-06, Revision 2.

HNP's commitment to NEI 97-06 Revision 2, is consistent with NRC and industry adoption of improved steam generator Technical Specifications and is considered a change associated with the current licensing basis. The adoption of the improved steam generator Technical Specifications has been approved by the NRC, as discussed above.

1. "30-DAY RESPONSE TO NRC GENERIC LETTER 2006-01, 'STEAM GENERATOR TUBE INTEGRITY AND ASSOCIATED TECHNICAL SPECIFICATIONS'," Serial HNP-06-029, dated February 16, 2006

2. NRC GENERIC LETTER 2006-01, "STEAM GENERATOR TUBE INTEGRITY AND ASSOCIATED TECHNICAL SPECIFICATIONS," dated January 20, 2006

3. Technical Specification Task Force, Improved Standard Technical Specifications Change Traveller, Steam Generator Tube Integrity (TSTF-449), Revision 4, May 6, 2005 (70 FR 24126)

4. APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY, Serial HNP-06-060, May 23, 2006

5. RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE LICENSE AMENDMENT REQUEST APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY, Serial HNP-06-116, October 3, 2006

6. RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE LICENSE AMENDMENT REQUEST APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY, Serial HNP-06-126, October 24, 2006.

7. "SHEARON HARRIS NUCLEAR POWER PLANT, UNIT I - ISSUANCE OF AMENDMENT ON STEAM GENERATOR TUBE SURVEILLANCE PROGRAM," dated March 16, 2007.

8. NEI Correspondence "Revision 2 to NEI 97-06 Steam Generator Program Guidelines," dated September 2, 2005.

Question No: B.2.9-FS-03

NRC Request:
FOLLOW UP ON STEAM GENERATOR TUBE INTEGRITY PROGAM

I. Please provide additional information regarding to HNP Steam Generator Tube Integrity Program Operating Experience, specifically:

(1) Provide listing and a brief summary of the your industry operating experience review. Please discuss whether it is relevant to the HNP steam generator program. If so, did it introduce any new requirements for this program.

(2) Provide listing and a brief summary of the your plant specif operating experience history. Please discuss condition reports, corrective actions, and explain how the corrective actions were resolved. Specifically, explain whether these conditions introduced any new requirements to the steam generator program.

II. NRC Information Notice 97-88, □Experiences During Recent Steam Generator Inspections,□ dated December 16, 1997 stated that □In May 1997, the licensee for the Shearon Harris Nuclear Power Plant found that four perforated, carbon steel ribs in a steam generator had been extensively damaged. The ribs are welded to the feedwater impingement plate which shields the steam generator tubes from direct impact of the feedwater flow. The licensee concluded that the high flow velocities of the feedwater eroded the ligaments between the perforations on the ribs.□

Please discuss whether loss of material aging effect due to erosion is applicable to the secondary side components of the replaced steam generators. If not, please explain. If so, please discuss how it is managed (prevented, inspected, detected, monitored, etc.) by the steam generator tube integrity program.

HNP Response:

1.(1) Summary industry operating experience review:

The following industry OE was reviewed for the Steam Generator Tube Integrity Program:

Harris Nuclear Plant License Renewal Audit Question and Response Database

NRC Generic Letters

NRC Generic Letter 2004-01 "Requirements for Steam Generator (SG) Tube Inspections."

NRC Generic Letter 2006-01 "Steam Generator Tube Integrity and Associated Technical Specifications."

NRC Information Notices

NRC Information Notice 97-88, "Experiences During Recent Steam Generator Inspections."

NRC Information Notices issued since January 2005

NRC Information Notice 2005-02: "Pressure Boundary Leakage Identified on Steam Generator Bowl Drain Welds."

NRC Information Notice 2005-09: "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005.

NRC Information Notice 2005-29: "Steam Generator Tube and Support Configuration."

NRC Licensee Event Reports

LER 05-002-00, (Docket No. 05000482), Reactor Coolant System Pressure Boundary Leakage Due to Small Cracks in Steam Generator Lower Head Bowl Drain Lines (Wolf Creek).

LER 05-001-00, (Docket No. 05000275), Steam Generator Tube Plugging Due to Stress Corrosion Cracking (Diablo Canyon Unit 1).

INPO Operating Experience for the Industry

OE20448, Additional Indications Identified on Steam Generator Tubes Using Plus Point Probe (Saint Lucie Unit 2).

OE20053, "Unknown Extra Lower (Cold Leg) SG Tube-to-Tubesheet Rolled Joints Found with PWSCC (Davis-Besse)."

OE20324, Steam Generator Tube Leak Due to Foreign Material Intrusion (Arkansas Nuclear One Unit 2).

OE20361, Loose Parts Found on Secondary Side of Steam Generator (Calvert Cliffs Unit 2).

OE20410, Steam Generator U-Bend Fan Bar/Collector Bar Not Detected in Hot Leg Row 1 Tubes (Byron Unit 1).

OE20384, New Damage Mechanism Identified for the Plant's SG Tubes (Watts Bar).

OE20539, Foreign Material and Tube Wear Issues in Steam Generator (South Texas Project Unit 1).

OE20339, Steam Generator Tube Crack Indications (Vogtle Unit 1).

OE20655, Unexpected Tube Wear in Replacement Steam Generators (Oconee Unit 1).

OE20919, Minor Primary-to-Secondary Leakage Identified Shortly After Startup from Refueling Outage (Watts Bar Unit 1).

OE20932, Broken SG Diagonal Tube Supports (Batwings) (Waterford Unit 3).

OE20792, Tube Plug Breakaway Stems Found in Feedwater Heaters (Summer).

OE21907, Foreign Material Found Inside the Reactor Vessel Originated from a Rotating Eddy Current Probe (Comanche Peak Unit 1).

Summary

HNP has programs and processes in-place for incorporating lessons learned from industry events into the steam generator tube integrity program. The above industry operating experience items were reviewed for applicability to the HNP steam generator tube integrity program, as follows:

With respect to the NRC Generic Letters, HNP found that steam generator tube inspections are consistent with the NRC's position regarding tube inspections. Additionally, HNP has submitted an application for Technical Specification improvement regarding steam generator tube integrity consistent with NRC and industry adoption of improved steam generator Technical Specifications. The adoption of the improved steam generator Technical Specifications has been approved by the NRC.

The subject NRC Information Notices and Licensee Event Reports were reviewed and found not to be directly applicable to the present-day HNP Model Delta 75 steam generators. Although the operating experience was not directly applicable to the HNP steam generators, the underlying aging mechanisms were also reviewed. The aging mechanisms associated with the NRC Information Notices and Licensee Event Reports were found to be addressed by the HNP steam generator tube integrity program.

INPO Operating Experience was reviewed for applicability to the HNP steam generator tube integrity program. For those events that were directly related to the present-day HNP Model Delta 75 steam generators, it was found that the HNP steam generator tube integrity program addressed the concerns identified. For those events that were not directly related to the present-day HNP Model Delta 75 steam generators, the underlying aging mechanisms were also reviewed. The aging mechanisms associated with the INPO Operating Experience were found to be addressed by the HNP steam generator tube integrity program.

1.(2) Summary plant specific operating experience review:

<u>HNP Operating Experience Since Steam Generator Replacement in Fall 2001</u>

HNP NCR 92603 (INPO OE16308), Incorrect Diameter Probe Used During Eddy Current Inspection.

HNP NCR 125127 (INPO OE18651 and OE18385), Steam Generator Tube Leak.

<u>Self-Assessments</u>

INPO Steam Generator Program Assessment (2000).

HNP Steam Generator Life Cycle Management Program Self-Assessment (2004).

Summary

A review of plant-specific condition reports, internal and external assessments was conducted and showed the Steam Generator Integrity Program to be critically monitored, effective in maintaining tube integrity, and continually improving.

Corrective actions introducing new requirements to the Steam Generator Integrity Program associated with Nuclear Condition Reports (NCRs) consisted of the following:

> Revision to the Steam Generator Tube Integrity Surveillance Test Procedure to improve identification/storage of various eddy current probes brought on site.
>
> Revision to the HNP Steam Generator Program Procedure to incorporate an independent review of the foreign object search and retrieval data.
>
> Revision to the HNP procedures to require verification of the automated analysis parameters during the Site-Specific Performance Demonstration

Corrective actions introducing new requirements to the Steam Generator Integrity Program associated with external and internal assessments consisted of the following:

> Action items to improve the documentation for eddy current techniques, tube plug inspection acceptance criteria, documentation of deviations to EPRI documents used in the Steam Generator Integrity Program guidelines, long-range planning of inspection activities for the replacement steam generators, and improvements in implementation of chemistry and primary-to-secondary leakage procedures.
>
> Additional actions taken included: (1) improvements in the documentation of actions from tube leak events, (2) improvements in degradation assessment, condition monitoring, and operational assessment procedures, (3) improvements in Steam Generator in-service inspection procedures, (4) improvements in primary-to-secondary leak detection procedures, and (5) review of the In-Service Inspection vendor root cause analyses.

The details of the operating experience reviews will be available during the AMR audit.

II. Aging Effect Due to Erosion

The table below lists the Steam Generator secondary side components associated with the aging effect, Loss of Material due to Erosion. The aging management program associated with the component/aging effect is also listed, along with a brief discussion of how the aging effect is managed.

Component	Aging Management Program	Discussion
Steam Generator feedwater impingement plate and support	One Time Inspection Program	The One-Time Inspection Program will provide a visual and/or volumetric inspection that either verifies that unacceptable degradation is not occurring or triggers additional actions that assure the intended function of affected components will be maintained during the period of extended operation.
Steam Generator tube bundle wrapper	Steam Generator Tube Integrity Program and Water Chemistry Program	Consistent with NUREG 1801, as indicated in LRA Table 3.1.2-6 (Page 3.1-148), loss of material of the tube bundle wrapper is managed by the Steam Generator Tube Integrity and the Water Chemistry Programs.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.10-MK-01

NRC Request:
Explain why the Enhancement Section of the AMP stated "None" when the NUREG-1801 Consistency statement for the AMP indicated that the program was an existing program that, following an enhancement, will be consistent with GALL AMP XI.M20. Please provide the enhancement and included in an appropriate license renewal commitment.

HNP Response:
The consistency statement is incorrect. The LRA should read:

"The Open-Cycle Cooling Water System Program is an existing program that is consistent with NUREG-1801, Section XI.M20."

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.10-MK-02

NRC Request:
Clarify how injections of chemical additives into the Open Cycle Cooling Water System would provide for sufficient concentrations of additives and adequate aging management in low-flow or stagnant-flow regions of the system.

HNP Response:
The Open-Cycle Cooling Water System Program in LRA Section B.2.10 does not credit the use of chemical treatment alone to ensure proper aging management in stagnant and low flow lines. Per the program's processes and procedures, HNP performs periodic flushing of small bore vents and drains and system flow paths that are not periodically operated. The program also includes periodic inspection and cleaning of large bore system pipelines and intake bays. In addition to periodic flushing of small bore lines, several pipe lines are periodically replaced or will be replaced with stainless steel piping. Finally, as part of the OCCWS Program, HNP uses NDE to manage aging effects in some intermittent flow sections of various size lines in the open-cycle cooling water piping.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.11-MK-01

NRC Request:
LRA Section B.2.11 for the Closed-Cycle Cooling Water System Program describes that some heat exchangers are not monitored for flow, inlet and outlet temperatures, and differential pressure, or from the functionality testing of this program. Identify which closed-cycle cooling water heat exchangers are excluded from the performance-based testing or functionality testing of the AMP, and clarify how the performance and structural integrity of these heat exchangers will be monitored. If other AMPs are credited with these aging management functions, identify which AMPs are credited for aging management of these heat exchangers.

HNP Response:
Flow, temperature and pressure are not specifically monitored in the following Heat Exchangers. As noted in Section B.2.11, in these cases, either the functionality of these heat exchangers is verified by activities outside the Closed-Cycle Cooling Water Program or the specific operating conditions of the heat exchanger render performance testing unreliable.

Primary Sample Condenser and Cooler - The performance of the sample coolers and condensers is validated as the system is used by chemistry personnel. These components are not needed for safe shutdown and not required to mitigate the consequences of an accident.

Component Cooling Water Heat Exchangers - The Component Cooling Water Heat Exchangers are tested/inspected as part of HNP's commitments to Generic Letter 89-13 as described in the Open-Cycle-Cooling Water System Program in B.2.10. An engineering evaluation concluded that factors inherent in the testing process makes the test results too unreliable to be used for operability determinations or as a basis for an inspection program. In addition, temperature and pressures are indicated on the main control board and Operations monitors them to ensure they are performing as expected for the plant conditions.

Emergency Diesel Generator Oil and Jacket Water Coolers - The Emergency Diesel Generator jacket water coolers are tested/inspected as part of HNP's commitments to Generic Letter 89-13 as described in the Open-Cycle-Cooling Water System Program in B.2.10. Inspection and cleaning of the Emergency Diesel Generator Lube Oil Cooler is included as part of a maintenance periodic test. The degradation of heat exchanger performance can be identified through these inspections.

EDG Turbocharger Intercoolers - The combustion air intercoolers are inspected/cleaned as part of periodic diesel generator maintenance. The degradation of heat exchanger performance can be identified through this inspection.

Reactor Coolant Drain Tank Heat Exchanger - The RCDT Heat Exchanger performs no safety related heat transfer function. The heat exchanger tubes provide a pressure boundary function. Nevertheless, high reactor coolant drain tank heat exchanger high temperature is annunciated and the procedural response is to investigate temperature increases that would indicate heat exchanger fouling.

Fuel Pool Heat Exchangers - An engineering evaluation concluded that factors inherent in the testing process make the test results too unreliable to be used for functionality determinations. Degradation of heat exchanger performance can be identified through control room and local alarms. This is considered an exception because specific performance testing is not performed. Per FSAR Section 9.1.3:

> "Control Room and local alarms are provided to alert the operator of high and low pool water level, and high temperature in the fuel pool. A low flow alarm, based on measured flow to the fuel pool, is provided to warn of interruption of cooling flow."

Air Handling Unit Cooling Coils - The safety-related air handling units are periodically inspected and differential pressures recorded. The condition of heat exchanger performance can be identified through this inspection. This is considered an exception because specific performance testing is not performed. Per procedures, Operations performs periodic monitoring of the rooms cooled by these safety-related units.

The licensing renewal activities described above along with the activities described in the Closed-Cycle Cooling Water System Program ensure the performance and structural integrity of these heat exchangers will be maintained during the period of extended of operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.11-MK-02

NRC Request:
The Closed Cycle Cooling Water Program references EPRI TR-107396, Revision 1. The GALL Report recommends Revision 0 of this EPRI report. Describe the process used at HNP to evaluate the incorporation of new industry standards or guidelines into existing plant procedures. Use the incorporation of EPRI TR-107396, Revision 1, into the HNP Environmental and Chemistry Sampling and Analysis Program (CRC-001), as an example.

HNP Response:
HNP and other utilities provide input to as well as reviews of the recommendations of the changes made to EPRI guidelines. EPRI recommended changes are input to the industry operating experience review. As part of this process it is reviewed against the current applicable chemistry program. During this review, manufacturer recommendations or manuals and associated station documents are consulted. Following this review, appropriate changes are made to station chemistry controlling documents such as CRC-001. The change process for this controlling procedure is subject to the safety review process (10 CFR 50.59 process).

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.12-CM-01

NRC Request:
HNP claims consistency with the GALL Scope of Program which states that the HNP Boraflex Monitoring Program monitors the effects of aging on the Boraflex panels. This program supports the HNP response to NRC Generic Letter 96-04 (References 5.16 and 5.20) and is implemented by EPT-099, Boraflex Integrity Test. ETP-099 uses coupons that are subject to testing under an accelerated, and long-term frequency. Explain whether there are enough test coupons available to comply with this commitment for the period of extended operation.

HNP Response:
The frequency for test sample removal for long-term surveillance testing is once per four years. There are currently sufficient test coupons in the spent fuel pools to continue testing at this rate throughout the period of extended operation. There are also additional test coupons available in storage tracked through part number 729-786-95. These additional test coupons could be used for future integrity testing after an analysis is put in place to account for the lack of exposure to spent fuel pool conditions.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.12-CM-02

NRC Request:
HNP claims consistency with enhancement for Program Element Preventive Action which states that the EPRI Racklife Predictive code will be used. Explain the frequency for running the code. Explain the frequency that data from samples will be input into the racklife program.

HNP Response:
As stated in LRA Section B.2.12, the Boraflex Monitoring Program will be enhanced to incorporate use of the Racklife predictive code, or an equivalent program, into the Boraflex Monitoring Program. Racklife has been used to monitor Boraflex performance and was used to develop information presented in HNP letter to the NRC, dated April 25, 2005; Supplemental Response to NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks." Updates to the Racklife code are not performed on a specified frequency; rather, the Boraflex Monitoring Program will use spent fuel pool silica concentration to initiate an update to the Racklife model.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.12-CM-03

NRC Request:
HNP claims consistency with enhancement for Program Element Detection of aging effects which states that the amount of boron carbide released from the Boraflex panel is determined through direct measurement of boron areal density and correlated with the levels of silica present with a predictive code. This is supplemented with detection of gaps through blackness testing and periodic verification of boron loss through areal density measurement techniques such as the BADGER device.

There is no mention of BADGER type testing performed to periodically verify boron loss. Explain what method will be used to perform this.

HNP Response:
The HNP Boraflex Monitoring Program will employ equivalent methods of areal density measurement in addition to blackness testing. This is consistent with the NUREG-1801 program element for Detection of Aging Effects, because it allows "..periodic verification of boron loss through areal density measurement techniques such as the BADGER device." Use of the BADGER device is not mandatory.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.12-CM-04

NRC Request:
HNP claims consistency with enhancement for Program Element Monitoring and Trending which states that the periodic inspection measurements and analysis are to be compared to values of previous measurements and analysis to provide a continuing level of data for trend analysis. Explain whether the enhancement for use of Racklife will be included in the trend analysis.

HNP Response:
The data obtained using the Racklife code is reviewed, recorded, and predicts the degradation of Boraflex over time. The concentration of silica in the spent fuel pools is trended to determine the continued accuracy of the Racklife predictions and to update the Racklife model.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.13-JW-01

NRC Request:
In Harris commitment letter HNP 06-0136 enclosure 1, the commitment 9 enhancements are not consistent with LRA Appendix A.1.1.13 and Appendix B.2.13. Commitment 9 items (5) and (6) are skipped in the numerical sequence. Commitment 9 item (7) is not included in LRA Appendix A.1.1.13 and Appendix B.2.13. Commitment 9 item (4) is not phrased consistent with LRA Appendix A.1.1.13 and Appendix B.2.13. Commitment 9 item (1) is not consistent with the text in LRA Appendix B.2.13. Explain the reasons for these discrepancies.

HNP Response:
HNP LRA Commitment No.9 and Appendix B.2.13 were not consistent with LRA Appendix A.1.1.13 and the basis document. HNP LRA Commitment No.9 and Appendix B.2.13 should be made consistent with LRA Appendix A.1.1.13 and the basis document.

On the basis of this response, HNP-06-0136, Enclosure 1, Harris Nuclear Plant License Renewal Commitments, Commitment No. 9 will be amended to agree with LRA Appendix A.1.1.13 and the basis document, as follows:

Commitment No. 9, item (7) will be deleted. LRA Appendix B.2.13, Detection of Aging Effects, Item (1) will be changed to state: "to include all cranes that are within the scope of License Renewal." After these changes, there will be only four (4) enhancement items associated with Harris commitment letter HNP 06-0136, Enclosure 1.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.15-PB-01

NRC Request:
NUREG-1801 states that the AMP applies to water-based fire protection systems and components that are tested in accordance with the applicable NFPA codes and standards. Calculation No. HNP-P/LR-0611; program element 3-2 states that flow testing is not in compliance with NFPA25. What aspect of the flow test is not in compliance and why was this type of enhancement (pipe wall thickness) selected?.

HNP Response:
HNP's water-based fire protection system and components are not presently licensed to NFPA 25. Flow testing is performed in accordance with the applicable NFPA as described in the Fire Protection program. The first option to be consistent with NUREG-1801, Element 3-2, is to perform flow testing per NFPA 25. Reconciliation between the NFPA codes of record with NFPA 25 is needed to credit this option. HNP conservatively chose the second available option, which is to perform and trend non-intrusive wall thickness evaluations at various locations, in lieu of flow testing. This is acceptable per the description of this program element in NUREG-1801.

Nevertheless, HNP will modify the enhancements described in the LRA under Parameter Monitored Inspected, Detection of Aging Effects, Monitoring and Trending, and Acceptance Criteria and its associated commitment to say:

Revise the Program to 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.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.15-PB-02

NRC Request:
The NUREG 1801 in Program Element 1-1 (X1.M27) Firewater Systems that managing loss of material due to corrosion, MIC or biofouling of carbon steel components. The HNP Fire Protection Program does not specifically identify the actions to mitigate "Biological Growth" from the Fire Water System, although it states the element is consistent with the GALL with no exceptions. Please identify HNP enhancements that allow this element with no exceptions.

HNP Response:
HNP periodically performs flow tests and fire protection main line basket strainer inspections to ensure no significant corrosion, MIC or biofouling has occurred in water-based fire protection systems. Flow testing is described in the supporting comparisons with GALL under Preventive Actions of the basis document.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.16-MK-10

NRC Request:
In HNP LRA AMP B.2.20, the statement is made that there are no buried tanks within the scope of license renewal. Why is the buried main tank for the Security Power System diesel engine fuel oil not included within the scope of license renewal and the external surface managed with LRA AMP B.2.20 Buried Tank and Inspection Program?

HNP Response:
Correction, the statement in LRA AMP B.2.20 says, "There are no buried tanks in the program." The buried main tank for the Security Power System diesel engine fuel oil is in scope and listed in Tables 2.3.3-22 and 3.3.2-22 as Buried Tanks. In Table 3.3.2-22 of the LRA, the HNP methodology concluded that Fiber Glass or Fiber Reinforced Plastic Buried Tank subjected to Soil (Outside) has no external aging effects. Plant specific note 728 describes the details of this tank, it states:

728. The buried tank is composed of an inner and outer tank. The exterior surface in contact with soil is made of a self-reinforcing resin (FibreThane) specifically formulated for use in the manufacture of composite storage tanks. The inner shell is steel with no coating. This Air/Gas (Wetted) environment represents the air space inside the fuel oil tanks above the fuel oil level and the air space between the tanks, which is accessible for inspection. The Fuel Oil Chemistry Program and One-Time Inspection Program are appropriate because they include an inspection of the internal surfaces of the tank.

Question No: B.2.16-MK-11

NRC Request:
The calculation for the Fuel Oil Chemistry Program (HNP-P/LR-0631) has been revised. This revision eliminates one of the exceptions to the GALL Report that was identified in the HNP LRA under the Preventive Actions program element. Please confirm that the LRA will be amended to remove this exception from the application. In addition, provide the proposed revised text to make it clear that this exception has been deleted. Please note in your response that an LRA amendment is required.

HNP Response:
Section B.2.16 of the LRA will be amended to say:

Under the Program Description section, the sentence starting with "Exposure to fuel oil contaminants..." will be changed to state:

"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. Subsequently, periodic sampling is performed to assure that the tanks are free of water, particulates, and biological growth."

Under the Exceptions to NUREG-1801 section, for the Preventive Actions, Change item 1 to a Note as stated below and renumber the remaining items:

> "Note: A stabilizer containing a biocide and corrosion inhibitor is added to new fuel before it is added to the storage tanks in the Diesel Fuel Oil System and the Security Power System."

Clarify new Item 3 for the Diesel Driven Fire Pump fuel oil storage tank that since the stabilizer is not added, the concomitant biocide and corrosion inhibitors are also not added. The third exception will now read:

> "The use of stabilizers, biocides, and corrosion inhibitors in the diesel-driven fire pump fuel oil tank is not warranted, as fuel oil is frequently refreshed. The consumption of fuel oil is the result of the monthly requirement in the Fire Protection Program to run the pump for 30 minutes on relief flow. The frequent addition of diesel fuel oil obviates the need for these additives. Additionally, site operating experience does not show adverse trends in

corrosion in the fuel oil components."

Under the Exceptions to NUREG-1801 section, for the Monitoring and Trending element, in the Security Power System discussion in item 2, the paragraph will be revised to state:

"Security Power System: The Buried Tank is monitored semiannually not quarterly. This exception is acceptable because operating experience shows no evidence of corrosion or biological growth since the installation of the new tank, use of the Diesel Grade No 1-D and fuel oil stabilizer containing a biocide and corrosion inhibitors. This covers over ten years of operating experience. If Diesel No. 2-D is used in the future, the monitoring, except for biological growth, will be performed on a quarterly basis for the main storage tank only. (Note: Replacement of the tank was done to comply with more stringent state and federal codes for buried fuel oil tanks.)"

(The revised words for the Program Description above will also replace the second and third sentences in LRA Subsection A.1.1.16.) A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.16-MK-12

NRC Request:
For the Fuel Oil Chemistry Program (B.2.16), please confirm the sampling frequency for the emergency diesel generator and security building diesel generator fuel oil day tanks. Also please provide any necessary enhancements or exceptions related to the sampling of these tanks

HNP Response:
The exceptions and enhancements described in the LRA require revision in order to clarify HNP's position. The Fuel Oil Chemistry Program description in NUREG-1801 addresses the aging management of the Main Fuel Oil Storage Tank. It does not address components downstream of the tank as they are supplied by that fuel oil supply. Since HNP manages the components downstream of the fuel oil storage tank using this program, limited and periodic confirmatory testing is being performed by sampling the fuel oil in the day tanks. Since this testing is not addressed in NUREG-1801, the frequency and testing is not considered an exception.

The LRA will be amended by discussing the confirmatory sampling activity of Fuel Oil Day Tanks in the program description and in the enhancement section Under Monitoring and Trending as described below:

Fuel Oil System:
Periodic testing of fuel oil for water and sediment and particulate count is a confirmatory test in EDG day tank and is considered an enhancement. As noted above this sampling is not described in NUREG-1801 and is not considered an exception. Note that the periodic sampling frequency may change with operating experience.

Security Power System:
The testing for water and sediment and particulate count is a confirmatory test which is already being done semiannually for the Security Diesel day tank. Therefore, this is neither an enhancement nor an exception. Note that the periodic sampling frequency may change with operating experience.

In both of the above systems, the initial frequency will be semiannual. Reference to periodic sampling will be made in the LRA AMP description section and in the FSAR amendment (LRA Appendices A1.16 and B.2.16). The license requirement will take effect prior to the extended period of operation.

Note: There are no commitments being made regarding biological testing in the fuel oil day tanks. Biological testing of fuel oil in the day tank will be performed on a conditional basis only.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.19-CM-01

NRC Request:
For Program Element Parameter Monitored/Inspected, HNP describes that the one-time inspection for the initial components will be performed on a representative sampling of components. Describe the methodology used to choose the test population.

HNP Response:
At this time, the methodology for choosing a population has not been decided. HNP is reviewing the latest industry technology used to inspect and evaluate the identification and progression of the selective leaching aging mechanism. HNP will adjust its methods based on an assessment of the effectiveness of the suggested approach or approaches.

For example, development of some level of industry guidance regarding the criteria for visual identification and hardness testing, i.e. EPRI report TR 1013477, Nuclear plant License Renewal Commitments - Utility Implementation Guidance, is scheduled for 2008.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.20-MK-01

NRC Request:
Is all of the underground piping managed by the Buried Piping and Tank Inspection Program safety related? If not, how are program elements 7, 8, and 9 implemented by the 10 CFR 50 Appendix B Quality Assurance Program?

HNP Response:
No. As stated in LRA Section B.1.3, the elements of corrective action, confirmation process, and administrative controls are common to all AMPs. The program controls and requirements meet the 10 CFR 50, Appendix B requirements. However, they apply to any component within the program regardless of safety class.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.20-MK-02

NRC Request:
Clarify whether or not there is any buried cast iron piping within the scope of license renewal. Is so, identify the inspection method that will be employed to detect loss of material due to selective leaching in the buried cast iron piping and clarify whether AMP 2.2.19, "Selective Leaching of Materials Program," or AMP B.2.20, "Buried Piping and Tanks Inspection Program," will be the AMP that is credited for detection of this aging effect.

HNP Response:
Yes, in the Fire Protection System, the buried portion of the fire hydrants, post indicating valves and pipe fittings are made of gray cast iron and are included in the Buried piping, piping components and piping elements Component Commodity shown in LRA Table 3.3.2-27, page 3.3-262. Buried ductile iron piping is evaluated as carbon or low alloy steel as shown on page 3.3-261.

In the Oily Drains System, the portion outside the Diesel Fuel Oil building was evaluated as buried, cast iron piping components and included in the Piping, piping components and piping elements Component Commodity shown in LRA Table 3.3.2-29, page 3.3-297.

Both tables indicate that loss of material due to selective leaching is being managed by the Selective Leaching of Materials Program. In LRA Section B.2.19, the discussion of Scope of Program under Exceptions describes the inspection method that will be employed to detect loss of material due to selective leaching. As noted in this section, Item 2 states that "(2) *other mechanical means, i.e., scraping, or chipping, provide an equally valid method of identification*" to Brinell hardness testing. This exception has been accepted by the NRC as stated in the "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2," NUREG-1856. This report states on page 3-88:

"Exception: A qualitative determination of selective leaching will be used in lieu of Brinell hardness testing for components within the scope of this program. The exception involves the use of examinations, other than Brinell hardness testing, identified in GALL AMP XI.M33. The exception is justified, because (1) hardness testing may not be feasible for most components due to form and configuration (i.e., heat exchanger tubes); and, (2) other mechanical means, (i.e., scraping or chipping provide an equally valid method of identification).

The staff reviewed the applicant's exception and determined that it is justified on the following basis: (1) hardness testing is not feasible for most components due to form and configuration; (2) other mechanical means (i.e., resonance when struck by another object, scraping, or chipping) will be used and provide an equally valid method of identification; and, (3) the applicant's program will include one-time inspections and qualitative determinations of selected components that may be susceptible to selective leaching. The staff considered the applicant's justification to be reasonable and acceptable."

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.20-MK-03

NRC Request:
In the Buried Piping and Tank Inspection Program, identify the methodology and criteria that will be used to determine locations for inspection based on areas with the highest likelihood of corrosion problems.

HNP Response:
The specific locations and methodology have not been determined. HNP will remain abreast of the Industry with regard to technologies in use and use site and industry Operating Experience reviews and Benchmarking to assist in the selection of an appropriate approach. As described in B.2.20, detailed procedural requirements for the program will be developed. Areas with highest likelihood of corrosion may be identified based on review of site specific operating experience in which degradation has occurred.

HNP will consider using other technologies available to meet its commitments. For example, EPRI is working on a device that could be used for condition Assessment of Large-Diameter Buried Piping (Reference 1). It is envisioned that this technology will scan long sections of piping which provides the advantage of not relying on sampling method. However, the technology is not available and its effectiveness is not known. Structural Integrity Associates has developed a Buried Piping Assessment Program to determine locations that may be suspect (Reference 2). A third approach may involve the review of DC electric current information from HNPs cathodic protection system. It may be used to suggest areas where coating degradation may have occurred.

HNP will base its approach on the effectiveness and cost of the various technologies available.

References:

1) EPRI Report 1011829, "Condition Assessment of Large-Diameter Buried Piping, Phase 2: Vehicle Design and Construction," December 2005.

2) Structural Integrity Associates, Inc. website, <<http://structint.com/images/buriedpipingflyer.html>>.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.20-MK-04

NRC Request:
The Operating Experience program element for AMP B.2.20, "Buried Piping and Tanks Inspection Program," states that leaks have occurred in some of the buried piping at HNP. Identify the buried piping systems and locations which experienced the degradation, and identify the root cause of the leaks for the degraded buried piping locations. Clarify whether the affected locations were ASME Code Class and how the degraded locations were repaired.

HNP Response:
HNP operating experience reviews have identified that underground piping leaks have occurred.

For non-ASME Code Class pipe:

- An underground leak on the discharge line of the diesel driven fire pump. The one GPM leak originated from a 90 degree elbow mechanical joint. The cause of the leak appears to be differential settlement of the soil backfill supporting the fire line. This leak is not considered age-related degradation.

- The 3 inch piping of the jockey fire pump discharge was found to be leaking at a mechanical joint. Some of the carbon steel bolts used to connect the flanges together were found to be extremely corroded to the extent that the bolts were no longer structurally functional. All mechanical joints are required to have a protective coating applied (such as Flaketar coal tar epoxy). These joints did not appear to have any substantial application of protective coating. Flaketar coating was used on the joint prior to backfill.

  The Site Fire Water System contains piping components that are flanged to underground piping, e.g. hydrants, valves, pipe sections. Similar to other piping components, the bolting is required to have protective coatings, e.g. Flaketar coal tar epoxy. The lack of coating in this case was assumed to be an error of omission as no other failures of this nature have been identified in over 20 years of operation.

- A leak was traced to the 12" fire header on the discharge of the Motor Driven Fire Pump. The leaks were found at two adjacent mechanical joint flanged connections. This leakage at a buried joint was identified and attributed to soil settlement at a flanged connection and is not considered age-related degradation. A contributing factor is that the gasket loses some of its elasticity due to age and hardens. The leaking flanged connections were replaced using new gaskets and new flanges. Gaskets are considered to be subcomponents of the piping and not credited as pressure boundary components. For License Renewal, gaskets are considered to be consumables as discussed in NUREG-1800, Table 2.1-3.

- A potable water line was installed very close to the yard grade, about one foot below the yard surface north of Unit 2. A forklift carrying materials heavier than a normal forklift traveled over this underground piping. The action of the heavy load movements caused the line to break. This piping leak was due to localized heavy load movements and is not considered age-related degradation.

For ASME Code Class pipe:

- During the 10 year pressure testing of fuel oil system buried piping in Refueling Outage 13, a leak was identified in the diesel fuel oil piping from a main diesel fuel oil storage tank to the day tank. The "A" train piping was unable to hold the required pressure. The leakage was isolated to a section of pipe under the Diesel Generator Building. The section of pipe under the building was abandoned and the underground piping was brought above ground just outside the building. The new piping from the buried line enters the Diesel Generator Building above grade level.

The location of the piping leakage was abandoned in place. The investigation concluded that: "Due to the location of the leak underneath the EDG Building, the pipe section with the leak could not be visually inspected; the apparent cause is a piping through-wall leak caused by exterior corrosion at a location where the coating was either defective or damaged during installation." The subject section of diesel fuel oil piping is ASME Code Class 3.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.21-SA-01

NRC Request:
Are there any socket weld identified as high safety significant locations as part of risk-informed inservice inspection (RI-ISI) program? How does HNP intend to examine socket welds under RI-ISI program?

HNP Response:
There are socket welds currently identified as "high" safety significant locations in RI-ISI.

VT-2 examinations are required to be performed on all Class 1 systems every outage. Currently, HNP performs this examination twice each outage, once at the beginning so we can correct any problems during the scheduled outage, and again as the plant is starting up to meet our ISI commitments. These examinations will include High Safety Significant class 1 socket welds. HNP will follow Section XI and NRC requirements for socket welds during the period of extended operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.21-SA-02

NRC Request:
How does HNP select sample population and location of piping to be inspected under the One -Time Inspection of ASME Code Class 1 Small-Bore Piping Program?

HNP Response:
Consistent with GALL, inspections will be performed at a sufficient number of locations to assure an adequate sample. The sample size for the plant-specific program will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations. The sample prioritization will consider the potential for mechanical loading as a result of thermal stratification, piping potentially susceptible to IGSCC (normally stagnant piping), and locations identified for inspection under the RI-ISI program (which considers thermal loading from plant cycles and thermal stratification).

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.21-SA-03

NRC Request:
B.2.21 states that element Monitoring and Trending has an exception to Gall. Basis Document for B.2.21 states that this program element is consistent with GALL with no exceptions. Clarify the discrepancy.

HNP Response:
This discrepancy reflects a minor "cut and paste" error in the conclusion column for the "Monitoring and Trending" program element and the "Detection of Aging Effects" program element. The element evaluations clearly identify the exceptions to GALL which is reflected in the LRA. The basis document will be revised to correct this discrepancy.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.21-SA-05

NRC Request:
Basis Document for B.2.21, element 4-1 states that socket welds receive a VT-2 visual inspection in accordance with the approved ASME Section XI ISI program. ASME Section XI specifies surface examination of socket welds. Clarify the discrepancy.

HNP Response:
ASME Section XI currently requires a pressure test at the end of each refueling outage on all Class 1 socket welds. VT-2 visual examinations are performed at that time. Currently, Section XI requires a surface examination of selected Class 1 socket welds. HNP will follow Section XI and NRC requirements for socket welds during the period of extended operation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.22-SA-01

NRC Request:
Please provide operating experience information for the External Surfaces Program (B.2.22) highlighting the condition of a component when observed during system walkdowns or maintenance.

Following original response to the above quesion the following additional information is required:
1. In response to NRC question B.2.22-SA-01, the applicant references three work orders 992505, 992506, and 992513 and Action requests AR 1118 and 117120.

Applicant to provide copies of these documents for NRC review.

HNP Response:
On June 25, 2007, the information requested in the supplemental request (i.e., provide complete copies of the work orders and ARs) was provided to the NRC reviewer.

Work Orders

Description: Repair Corroded Pipe, Flanges and Bolting
Date: 02/16/2007

Work Order 992505
Component: Pipe 1-FP-310
Location: "C" main transformer pit

Work Order 992506
Component: Pipe 1-FP-313
Location: "B" main transformer pit

Work Order 992513
Component: Pipe 1-FP-315

Location: "A" main transformer pit

Work Order Description:

The insulated, heat traced fire protection pipe in this line has significant outside diameter corrosion. Replace the flange, bolting and pipe as necessary to restore to an acceptable condition.

Work Order Instructions:

Remove the 6" and the 3" flanges from the line. Inspect lines for corrosion. If needed replace the first section of 3" piping. Install new flanges onto piping, using approved pipe sealant. Fabricate new studs. Install studs into flanges and torque to 150 ft. Lbs.

Action Requests

AR 1118 - Chilled Water System Degradation - 03/12/1997

1. Event Description

The Chilled Water System piping and supports are exhibiting rust and corrosion. There have been no operability concerns identified as a result of these inspections. During ISI examinations on the component supports of the above system, moisture was observed under the insulation which is contributing to rusting of unpainted hanger parts, field welds, and welded pipe attachments. The systems operate at temperatures below ambient and water condenses onto surfaces that are exposed to air.

2. Cause

The piping was originally placed in service without an application of paint to protect the piping and associated components (valves, hangers, supports). This condition, along with the inadequate use of fiberglass insulation, creates a "sponge" effect which allows the corrosion of the carbon steel components. This "sponge" effect is created due to the formation of condensate droplets on the exposed portions of piping, hangers or valves and the associated saturation of the attached insulation.

3. Action Taken

The first action taken was to evaluate the effectiveness of the current method of repair. This method consisted of cleaning the component, applying a corrosion barrier (paint) and replacement of insulation. This was accomplished by inspecting several of the hanger supports identified during Refueling Outage 6 that required repair. Additionally, approximately 15 hangers were removed from the piping to inspect the condition of the piping itself. The material condition of the inspected chilled water piping was found to be in satisfactory condition.

Based upon the results of the inspections, the current method of repair was validated. A plan was developed to define the scope and implement the repair approach for the remaining piping system components.

AR 117120 - Corrosion of Traveling Screen Baskets - 02/04/2004

1. Event Description

The screen panels (or baskets) on the Emergency Service Water traveling screens are deteriorating due to corrosion. Although not a current concern for operability or function, this condition will eventually lead to loss of structural integrity.

2. Cause

The extent of corrosion is due to inadequate original design. The galvanized coating on the baskets has not lasted as long as expected. Without this coating, extensive corrosion has occurred in the underwater/spray/humid service environment for these screens.

3. Action Taken

The recommended action is to replace the panels with new or refurbished baskets that are coated with an approved coal tar epoxy. This provides an improved corrosion barrier, which is also easier to monitor for condition and repair.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.23-CM-01

NRC Request:
Program Element 5, Monitoring and Trending, states that "The wall thickness measurements will be trended and wear rates will be calculated."

The Monitoring and Trending program element indicates that plant-specific wear results should be used to establish the wear projections for the thimble tubes.

The WCAP-12866 methodology includes using a generic wear rate exponent for predicting wear projections in lieu of using actual plant-specific wear rate data. One of the enhancements for the HNP program calls for CPL to use the generic wear rate exponent in WCAP-12866 for the wear projections if proper justification is made. This not consistent with GALL's statement that wall-thickness measurements will be trended.

Explain how proposal to use the generic exponential wear value from WCAP-12866 would yield more conservative future wear projections than would be projected if the actual plant specific wear data were used.

HNP Response:
HNP agrees that allowing a provision to use a generic wear curve exponent is not consistent with GALL. Therefore, Section B.2.23 of the LRA will be changed to delete statement "If the generic value of 0.67 is used for "n", a basis must be provided for using the generic value in lieu of plant-specific data." Program basis documents will also be revised accordingly and program enhancements, when implemented, will not allow for use of the generic value in lieu of a plant-specific data.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.24-SA-02

NRC Request:
Please provide operating experience information for the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program (B.2.24) highlighting the condition of a component when opened for maintenance.

Following original response to the above question the following additional information is required:
1. In response to NRC question B.2.24-SA-02, the applicant references two Action Requests AR 18914 and 19169.

Applicant to provide copies of these documents for NRC review.

HNP Response:
In response to the supplemental request, on June 25, 2007 copies of the requested documents (complete copies of the ARs) were provided to the NRC reviewer.

AR 18914 - 'A' Condensate Pump Suction Expansion Joint Degraded - 04/24/2000

1. Event Description

During inspection of the "A" Condensate Pump suction 48" expansion joint, internal degradation was discovered. The rubber liner was found separated on both sides of the arch; the liner was completely pulled away in some locations exposing the fiber element of the joint to process flow. This caused softening of structural joint material, although no cracks were detected in the structural backing material. The "B" Condensate Pump suction expansion joint was inspected and also found to be degraded.

2. Cause

A 48" expansion joint with an oversized arch exposed to near full vacuum is a very demanding environment. Since the arch is the structural weak link of the joint, in this application the oversized arc is prone to inward deflection. Inward deflection will stress the liner to structural fiber interface and promote separation of the two.

The existing joints protrude into the flow stream. They should not. This condition indicates that the arch was drawn into the flow stream by the vacuum forces and not enough structural support existed in the arch for the application. In addition, once separation occurs, the flow will tend to further separate the liner and joint fabric.

A combination of flexing induced liner separation from the structural fabric and flow over the separated liner is the likely cause of expansion joint failure.

3. Action Taken

It was recommended that the replacement expansion joints provide additional structural support to the arch to prevent it from being drawn into the pipe to the extent liner separation occurs. This is best accomplished through a filled arch and upgraded structural fabric. Upgrading from polyester to nylon tire cord and filling the arch will provide the desired structural support to the arch. This change was accomplished via plant design change.

AR 19169 - Foreign Material in "B" Moisture Separator Reheater - 04/30/2000

1. Event Description

A fragment of steel plate was found in the "B" Moisture Separator Reheater during Refueling Outage 9. The plate was found in the south end elliptical head between the external manway and the internal doors.

2. Cause

The origin of the plate steel was not found. All accessible regions of the Moisture Separator Reheater were inspected. No missing material was found. It is possible this was left in the Moisture Separator Reheater after some rework was performed during Refueling Outage 8. Some of the rework performed during Refueling Outage 8 involved welding patches on the distribution header. The material thickness of the header is about 1/8" which is close to the material thickness of the loose plate steel.

3. Action Taken

A crawl through found no other loose material. No similar patches were installed in Refueling Outage 9. No further actions were taken.

AR 29988 - High Pressure Seal Oil Backup Pump Oil Leak - 03/25/2001

1. Event Description

During the performance of pump operability test oil leak developed on casing of high pressure seal oil backup pump. A little over a gallon of oil accumulated on top of and along the sides and below lube oil reservoir.

2. Cause

The High Pressure Seal Oil Backup Pump was started for a routine operability test. During the test, an oil leak developed on the outboard back plate. The pump was disassembled and inspected during Refueling Outage 10 to determine the cause of the leak. The cause of the pump housing leak was due to the upper inner bearing race rubbing against the back plate. The inner race is shrunk fit onto the pump impeller shaft. The apparent cause of the inner race becoming loose and machining a hole through the back plate is inadequate interference fit between the race and shaft. This is a manufacturing defect.

3. Action Taken

A new pump was installed and tested.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.26-JW-01

NRC Request:

1) Explain the need for the program enhancements to address surface irregularities, moisture barriers, pressure retaining bolting and augmented examinations.

2) Explain how these four items are addressed or inspected under the current IWE program.

3) Explain if this program has been in compliance with ASME Section XI, Subsection IWE since the final rulemaking to require IWE inspections was made by the NRC in 1996.

4) Explain historically what inspection findings have lead to the need for augmented inspections?

5) Explain if any augmented inspections are currently being performed?

HNP Response:

1) The HNP administrative engineering surveillance test procedure which provides instructions for the general visual examination for ASME Section XI, Subsection IWE does not specifically discuss items such as surface irregularities (for metallic surfaces without coatings), moisture barriers, pressure retaining bolting and augmented examinations. The inspections of these items are however included within the First Containment Inspection Interval Containment Inspection Program document and specific QA inspection documents. The enhancement only improves the administrative procedure by including the instructions for all the IWE inspection requirements into one administrative procedure.

2) The four items are addressed as follows:
Surface irregularities – The HNP administrative engineering surveillance test procedure for the ASME Section XI, Subsection IWE Category E-A, Containment surfaces inspections does not currently list surface irregularities as a specific recordable condition. However, gouges, dents, bulges, and other damage, deformation, or degradation are listed as recordable conditions in the HNP administrative engineering surveillance test procedure and envelopes surface irregularities. The enhancement adds the specific term of "surface irregularities" to the HNP administrative engineering surveillance test procedure. It should also be noted that a QA visual examination form is utilized for inspection of various MC surfaces and it does include "other signs of irregularities" as a specific recordable condition.

Moisture barriers - The inspections of the Category E-D, moisture barrier is performed using a QA visual examination form with the appropriate inspection attributes (wear, damage, erosion, tear, cracks, or other defects). The completed QA visual examination form for the moisture barrier inspections is attached to the administrative engineering surveillance test procedure for the ASME Section XI, Subsection IWE Program as a QA record.

Pressure retaining bolting - The inspections of the Category E-G, Pressure Retaining Bolting is performed using the First Containment Inspection Interval Containment Inspection Program document and a QA visual examination form.

Augmented examinations – An evaluation of the potential Category E-C, Containment Surfaces requiring augmented examination are included as an Appendix to the First Containment Inspection Interval Containment Inspection Program document. However, no areas have been identified as surface areas requiring augmented examination.

3) The program has been in compliance with ASME Section XI, Subsection IWE since the final rulemaking to require IWE inspections was made by the NRC in 1996. The First Containment Inspection Interval for Subsection IWE is defined from September 9, 1998 to September 8, 2008 as described in the HNP Containment Inspection Program Document.

4) The HNP Containment Inspection Program Document provides an evaluation of surfaces likely to experience accelerated degradation and aging. Four areas were identified with the potential for augmented inspections. However, after evaluation, none of the four areas were identified as surface areas requiring augmented examination. The enhancement was provided as a clarification to the administrative procedure, not because there was a need for augmented inspections.

5) No augmented inspections are currently being performed.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.26-JW-02

NRC Request:
HNP lists no actual Containment IWE Inservice Inspection findings under operating experience for AMP B.2.26 in the LRA. Provide documentation from discovery to resolution for any historical Containment IWE Inservice Inspection findings.

HNP Response:
A detailed operating experience review was performed and documented in the basis document for the ASME Section XI, Subsection IWE Program and is available for review at HNP. Specific examination reports are also available at HNP for review. The following provides a summary of the findings.

The First Containment Inspection Interval Containment Inspection Program document provides a historical record of Containment inspections prior to the implementation of the IWE Program as discussed below.
Vertical liner corrosion was identified at the interface between the base slab and liner in Refueling Outage 7 (1997). This required partial removal/replacement of the Moisture Barrier. HNP Engineering determined the liner thickness met design requirements and that the deteriorated Moisture Barrier was the root cause. The entire Moisture Barrier was removed during Refueling Outage 8 (1998); the liner was cleaned, the thickness was confirmed to meet design requirements, was coated, and a high density silicone seal Moisture Barrier was installed. The vertical and horizontal liner at the base slab was examined during Refueling Outage 8 and 9 and only minor corrosion was identified. No further actions were required. In addition, the liner plate below the top of the base slab was examined in Refueling Outage 7 after the Moisture Barrier was removed and only minor corrosion was identified. A sample section of liner under the sump topping slab was also examined and no corrosion was identified. In addition, corrosion of the exterior surface of the "A" Containment Spray Valve Chamber due to persistent groundwater intrusion was identified in 1993. However, only minor corrosion was recorded and Ultrasonic Testing (UT) was subsequently performed.

Docketed Letter HNP-00-122, Inservice Inspection Summary Report, To the USNRC from James Scarola, dated October 18, 2000 documents the IWE inspections performed in Refueling Outage 9 (completed 05/12/00). Some recordable indications were observed during the examination but they were determined to be non-relevant by the Responsible Engineer and Program Manager. Conditions observed included coating blisters, mechanical damage to coatings, and discolored coatings on the liner. No significant metal loss was identified in the areas. Some rust and pitting was identified inside the "A" Containment Spray Valve Chamber. The metal thickness however was above nominal thickness as determined by UT. The liner under the transfer canal was identified as bulged but was found acceptable by HNP Engineering with no further action needed. A complete examination of the Containment liner and penetrations, Moisture Barrier, gaskets on applicable penetrations, and penetration bolting was performed.

Docketed Letter HNP-05-018, Inservice Inspection Summary Report, To USNRC from DH Corlett, dated February 15, 2005 documents the IWE inspections performed in RFO-12 (completed 11/15/04). No recordable conditions were identified on the Containment liner from the Moisture Barrier to the center of the dome. A number of non-recordable conditions on the Containment liner were observed such as scattered mechanical damage, blisters with no resulting material loss, and small areas with flaking coatings. A recordable indication of blistering was observed on the protective coating inside the lower regions of each of the valve chambers. No significant material reduction was identified as determined by UT, and the surfaces were recoated. A complete examination of the Containment liner and penetrations, Moisture Barrier, Valve Chamber internals and bolting, Equipment Hatch and the Refueling Access Sleeve was performed.

Docketed Letter HNP-06-081, 90 day Inservice Inspection (ISI) Summary Report, To USNRC from DH Corlett, dated August 10, 2006 documents the IWE inspections performed in RFO-13 (completed 05/16/06). The report states that no examinations of ASME Class MC Components were required or scheduled, but as a prudent measure, examinations of the Moisture Barrier and approximately 12" up from the Moisture Barrier on the liner was performed with no recordable indications observed. The report also states a visual inspection inside the "A" Containment Spray valve chamber including the bolts and nuts on the manway was performed but no recordable conditions were observed. In addition to the report, a visual examination inside the three (3) remaining valve chambers was performed with no recordable conditions observed. One small damaged coating area was repaired in the "A" Containment Spray valve chamber.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.26-JW-03

NRC Request:
The HNP LRA Appendix B does not have a Protective Coating Monitoring and Maintenance Program section. Please explain how HNP met the intend of GL 98-04, GSI 191, and GL 2004-02?

HNP Response:
Actions taken related to GL 98-04, GSI 191, and GL 2004-02 are part of the current licensing basis; some of these actions remain ongoing.

The NRC issued Generic letter 98-04[1] to:

1. Alert addressees that foreign material continues to be found inside operating nuclear power plant containments.
2. Alert addressees to problems associated with the material condition of Service Level 1 protective coatings inside the containment.
3. Request information to evaluate the addressees' programs for ensuring that Service Level 1 protective coatings inside containment do not detach from their substrate during a DB LOCA and interfere with the operation of the ECCS and the safety-related CSS.

The HNP November 09, 1998[2] response to Generic letter 98-04 provided the requested information and this issue was closed out via NRC correspondence dated November 16, 1999[3].

As discussed in Generic Letter 2004-02[4], BWR research findings indicated that fibrous material plus particulate material could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance."[5] The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the emergency core cooling system (ECCS) and containment spray system (CSS) in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required.

In resolution of these issues, Generic Letter 2004-02 requested that addressees take the following actions:

1. Using an NRC-approved methodology, perform a mechanistic evaluation of the potential for the adverse effects of

post-accident debris blockage.
2. Implement any plant modifications that the above evaluation identifies as being necessary to ensure system functionality.

Generic Letter 2004-02 requested that addressees provide the following information within 90 days of the date of the safety evaluation report providing the guidance for performing the requested evaluation:

1. Information regarding planned actions and schedule to complete the requested evaluation for the adverse effects of post-accident debris blockage.
2. A statement of intent to perform a containment walkdown surveillance in support of the analysis of the susceptibility to the adverse effects of post-accident debris blockage.

This information was provided by HNP correspondence dated March 4, 2005[6].

Generic Letter 2004-02 further requested that addressees provide the following information by September 1, 2005:

1. Confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in Generic Letter 2004-02.
2. A general description of and implementation schedule for all corrective actions, including any plant modifications.
3. A description of the methodology that was used to perform the analysis for the adverse effects of post-accident debris blockage.
4. A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications.
5. A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment will be assessed for potential adverse effects of post-accident debris blockage.

This information was provided by HNP correspondence dated September 01, 2005[7]. Furthermore the September 01, 2005 letter makes the following commitment:

"Complete the corrective actions of this response letter (HNP-05-101) to Generic Letter (GL) 2004-02 by the GL requested due date of December 31, 2007."

As discussed above, activities related to GL 98-04, GSI 191, and GL 2004-02 are part of the current licensing basis.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Corrective actions as described in HNP correspondence dated September 01, 2005 are committed to be complete by December 31, 2007.

1. Generic letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, " July 14, 1998.

2. 120-Day Response to NRC Generic Letter 98-04. "Potential For Degradation of the Emergency Core Cooling System and the Containment Spray System After A Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies And Foreign Material In Containment", Serial HNP-98-155, November 09, 1998.

3. Completion of Licensing Action For Generic Letter 98-04, - "Potential For Degradation Of The Emergency Core Cooling System and the Containment Spray System After A Loss-Of Coolant Accident Because Of Construction And Protective Coating Deficiencies and Foreign Material In Containment" Shearon Harris Nuclear Power Plant, Unit 1, (TAC NO. MA4053), November 16, 1999.

4. Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004.

5. Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance."

6. Response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," PE&RAS-05-008, March 4, 2005.

7. Response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," HNP-05-101, September 01, 2005.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.26-JW-04

NRC Request:
In Basis document HNP-P/LR-0616 an exception is taken to the GALL AMP XI.S1 Program element Scope of Program. In section 7.3.1 of HNP-P/LR-0616 the following statement is made in discussing the exception to the AMP Scope of Program element: In conformance with 10 CFR 50.55a(g)(4)(ii), the ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval. This statement has been omitted from the HNP LRA on page B-76 where the exception to the Scope of Program element is discussed for the ASME Section XI, Subsection IWE Program. Explain why this statement was omitted from the LRA exception discussion and if it is applicable.

HNP Response:
The requirement of 10 CFR 50.55a to update the ISI Program during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval was inadvertently not repeated in the LRA, because of an oversight. This update is required by NRC regulatory requirements and is applicable to the ASME Section XI, Subsection IWE Program.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.27-SA-01

NRC Request:
The applicant states that the program is credited for the aging management of accessible and inaccessible pressure retaining primary containment. Are coated areas included under this program? If yes, please describe which coated areas are examined.

HNP Response:
The accessible exterior surfaces of the Containment cylinder wall and base mat where it is enclosed within the Reactor Auxiliary Building and the Fuel Handling Building are generally coated with Service Level II coatings with a few exceptions (such as at high or locked high radiation areas, and at the Main Steam Tunnel). These accessible coated surface areas are examined by the IWL Program. Coating degradation on concrete surfaces is not one of the required inspection attributes in the administrative procedure for the IWL Program because coating degradation does not affect the structural integrity or leak tightness of the Containment. However, according to the administrative procedure for the IWL Program, the VT examiners are encouraged to note all degraded or unusual conditions even if they do not affect the structural integrity or leak tightness of the Containment. The basis document for the IWL Program development stated the condition of the coating can be an indicator of distress and/or degradation of the concrete and that coating degradation should be investigated to assist in the determination of whether distress and/or degradation of the concrete is present. A review of the actual IWL program inspection results from 2001 show coating degradation is identified and evaluated by HNP Engineering.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.27-SA-02

NRC Request:
In Basis document HNP-P/LR-0617, an exception is taken to the GALL AMP XI.S2 Program element Scope of Program. In section 7.3.1 of HNP-P/LR-0617 the following statement is made in discussing the exception to the AMP Scope of Program element: In conformance with 10 CFR 50.55a(g)(4)(ii), the ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval. This statement has been omitted from the HNP LRA on page B-79 where the exception to the Scope of Program element is discussed for the ASME Section XI, Subsection IWL Program. Explain why this statement was omitted from the LRA exception discussion and if it is applicable.

HNP Response:
The requirement of 10 CFR 50.55a to update the ISI Program during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval was inadvertently not repeated in the LRA, because of an oversight. This update is required by NRC regulatory requirements and is applicable to the ASME Section XI, Subsection IWL Program. License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.28-SA-01

NRC Request:
In program description of LRA, applicant states that program procedures provide for visual examinations of ISI Class 1, 2, and 3 supports. Please justify for not including ASME Class MC supports under this program.

HNP Response:
There are no ASME Class MC supports at HNP as discussed in the First Containment Inspection Interval Containment Inspection Program document. The document states "The welded attachments to the metallic liner (e.g., floor beams, seismic restraints, leak channels, equipment/pipe supports, etc.) do not perform a pressure retaining function associated with the containment support load path. For this reason, the welded attachments are classified as nonstructural and are not subject to inspection."

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.28-SA-02

NRC Request:
Justify use of ASME OM Code and ASME OM Code Case OMN-13 for Snubbers and attachments.

HNP Response:
The discussion of the ASME OM Code and ASME OM Code Case OMN-13 will be removed from the LRA Appendix B.2.28 ASME Section XI, Subsection IWF Program, "Program Description" and "Scope of Program" because ASME OM Code and ASME OM Code Case OMN-13 applies only to snubbers, and snubbers are not in the scope of License Renewal . The discussion of the ASME OM Code and ASME OM Code Case OMN-13 will be removed from the LRA Appendix A.1.1.2.8 ASME Section XI, Subsection IWF Program Description for the same reason. The component and piping supports (including any snubber attachments) will meet the requirements of ASME Section XI, Subsection IWF, 1989 Edition, and in accordance with ASME Code Case N-491-2.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.29-CM-01

NRC Request:
GALL Report Section XI.S4 for the Appendix J Program describes that corrective actions are required for unacceptable performance. 1) Provide a comparison between the enhancement identified in LRA B.2.29 for corrective actions and the existing method of performing corrective actions. 2) Provide a list of Appendix J test failures due to causes other than valve seat leakage. Include the corrective actions taken to mitigate the failed components and restore Containment Integrity.

HNP Response:
Nuclear Condition Reports (NCRs) are required to be initiated by all employees according to the Corporate Corrective Action Program for unacceptable conditions such as a deficiencies or deviations in an item or activity that has affected or reasonably could affect nuclear safety or quality. NCRs are initiated when leakage rates do not meet the acceptance criteria and require investigation and appropriate corrective actions. However, an enhancement is needed to improve the site administrative procedures to describe the evaluation and corrective actions to be taken when leakage rates do not meet their specified acceptance criteria.

A record of the Appendix J test failures for Appendix J Type B and C testing is maintained by the Appendix J Program Engineer and is available for review at HNP. A review of failed tests through year 2000 (through RFO-9) was performed. Only two test failures due to causes other than valve seat leakage were identified, both on Type C testing. These were both related to flange leakage. NCRs were appropriately initiated. A leaking flange was identified in 2000 during the Local Leak Rate Test (LLRT) which was above the acceptance criteria. A Maintenance Work Request was initiated and the inboard valve flange was determined to be leaking. The inboard valve was disassembled from the penetration, the seating surfaces were cleaned, and the inboard O-ring and the outboard gasket were replaced. Another LLRT test was then performed with satisfactory results. A leaking flange was identified in 2004 during a LLRT which was above the acceptance criteria. A Maintenance Work Request was initiated and the gasket material on the inside face of blind flange was replaced. Another LLRT test was then performed with satisfactory results.

Additionally, a review of the failures due to valve seat leakage was performed through year 2000 and all had NCRs appropriately initiated.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.29-CM-02

NRC Request:
Program Element 1 describes the scope of the 10 CFR 50 Appendix J program.
Calculation No. HNP-P/LR-0615 describes that AMP B.2.29 is consistent with GALL. It was noted however that the calculation states that Containment Structural Integrity, which requires a visual inspection of the structure surfaces prior to the Type A Containment leak rate test (ILRT) to verify no apparent changes in appearance or other abnormal degradation, using structural inspection attributes for Containment as shown in EGR-NGGC-0351 (Reference 5.28).

There is no mention of Surface Inspection in GALL. Explain how this is consistent with GALL, and whether the IWE Program credits B.2.29.

HNP Response:
The 10 CFR 50 Appendix J Aging Management Program (B.2.29) will be consistent with GALL following enhancement. The requirement to perform a visual examination of the exposed accessible interior and exterior surfaces of the Containment, including the liner plate during the shutdown for each Type A containment leakage rate test is included based on Technical Specification surveillance requirement 4.6.1.6.1. This inspection is plant specific and is above the requirements identified in GALL Appendix J Program. HNP considered performing plant specific requirements over and above the requirements of GALL as still consistent with GALL.

The IWE Program does not credit Appendix J Program. In fact, the IWE Program basis document states that the Appendix J leak rate testing is evaluated as a separate AMP for License Renewal.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.30-JW-01

NRC Request:
Have any masonry walls been added to the program due to license renewal?

HNP Response:
Masonry Walls in two structures were added to the Masonry Wall Program as a result of License Renewal. The two structures are the Security Building and the HVAC Equipment Room.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.30-JW-02

NRC Request:
Explain how often masonry walls are inspected for cracking. Explain if the inspection frequency varies from wall to wall. If the frequency does vary, explain the basis for the differences in frequency.

HNP Response:
The inspection interval for inspection of masonry walls for cracking varies from structure to structure but shall not exceed ten (10) years as established in a corporate level inspection procedure used for HNP, Robinson Nuclear Plant, Brunswick Nuclear Plant and Crystal River 3 Nuclear Plant. The basis for various frequencies at HNP was established based on the safety significance of the structure (based on Probabilistic Safety Analysis rated systems, structures, and components), the condition of the wall based on the results from previous structural inspections, and to accommodate work load management of the HNP engineering personnel. This results in a frequency of inspection which ensures there is no loss of intended function between inspections as described in GALL Program XI.S5. For example, the masonry walls in the Reactor Containment Building are examined at 5 year intervals, the Fuel Handling Building at 7 year intervals, the Turbine Building at 8 year intervals, and several non-safety related structures at 9 year intervals, etc. An inspection frequency has not typically been established from wall to wall within a specific structure, however the Responsible Engineer has the responsibility to establish the inspection frequency based on the previous inspections. Since 1996, when the inspections were initiated, there have not been any unacceptable conditions identified on masonry walls from cracking. Therefore, there has not been a need to change the inspection interval for masonry walls. If unacceptable conditions are identified in the future, initiation of a Nuclear Condition Report and corrective actions are required. This corrective action could result in increasing the inspection interval for a specific masonry wall, based on the Responsible Engineer's disposition. The same corporate procedure is used for inspecting building concrete/grout. A recent example involved increasing the inspection interval for a foundation in the Diesel Generator to a yearly frequency based on the condition of the grout. It should also be noted that there are no unreinforced masonry walls located in safety related areas at HNP.
HNP does not consider the methodology for selecting the interval for inspection of masonry walls an exception to GALL Program XI.S5 Program Attribute 4 based on the following: IE Bulletin (IEB) 80-11 was issued to HNP for information while HNP was under construction. HNP designed and constructed Category I masonry walls as described in the HNP FSAR (Section 3.8.4.8). In order to preclude problems of the type addressed by IEB 80-11, HNP designed all masonry walls in the proximity of safety-related equipment to meet seismic design criteria. The walls were inspected by QA/QC inspectors in accordance with implementation procedures. In addition , attachments of equipment to masonry block walls was approved on a case by case basis. Safety related masonry walls at HNP are analyzed in a structural calculation.

Several NRC IE Construction Assessment Teams examined HNP's costruction activities of masonry walls in 1984 and 1986 and validated IEB 80-11 requirements were met. The following NRC Letters document that HNP designed and constructed masonry walls to IEB 80-11 requirements: NRC IR 50-400/84-41, 50-400/84-48, 50-400/86-03, 50-400/86-06 and 50-400/87/32.

The masonry walls at HNP were constructed to the requirements of IEB 80-11 and without the design and construction problems typical of earlier plants. The HNP masonry walls at HNP have proven to be designed, constructed, and verified to QA requirements and no unacceptable conditions have been identified in over twenty (20) years after installation. HNP considers the methodology utilized by the Responsible Engineer for selecting the inspection intervals for masonry walls at HNP to meet the GALL Program XI.S5 Program Attribute 4 attributes. In conclusion, at HNP there is no need to inspect non-reinforced masonry walls more frequent than reinforced masonry walls, unless unacceptable conditions are identified.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.31-JW-01

NRC Request:
For concrete structures below grade, 1) provide the dates and results (at specific locations, not average or ranges) of the two most recent tests and the scheduled frequency of groundwater monitoring. 2) Clarify if the Structures Monitoring Program will continue to perform the groundwater monitoring and inspect all inaccessible areas that may be exposed by excavation, whether the environment is considered aggressive or not.

HNP Response:
As stated in LRA Section 3.5.2.2.1, site groundwater was sampled for License Renewal in August 2005 from two wells (Well 57 – pH 7.6, chlorides 290 mg/l, sulfate 2.4 mg/l; Well 59 - pH 7.9, chlorides 42 mg/l, sulfate 2.1 mg/l). Prior to this, groundwater was sampled in 1973 from three site wells as recorded in FSAR Table 2.4.13-8 (Well 2 - pH 7.3, chlorides 23 mg/l, no sulfate reading; Well 4A - pH 7.9, chlorides 22 mg/l, no sulfate reading; Well 7A - pH 7.9, chlorides 21 mg/l, no sulfate reading). The original 1973 wells are no longer active. The Structures Monitoring Program will add a groundwater implementing procedure to require periodic groundwater chemistry monitoring including consideration for potential seasonal variations (as stated in LRA Appendix B Section B.2.31). The monitoring will begin on five (5) year intervals from 2005 until the extended period of operation (for trending prior to the extended operation period) and then on a yearly interval thereafter. This enhancement is being implemented even though the groundwater is currently non-aggressive. In addition, a Structures Monitoring Program implementing procedure is being enhanced to require inspection of inaccessible below-grade concrete when exposed by excavation prior to backfilling. The enhancement for inspecting inaccessible below-grade concrete when exposed by excavation will also be continued during the period of extended operation even though the groundwater is non-aggressive.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.31-JW-02

NRC Request:
Explain how the frequency of inspection for the structures, buildings and components within the scope of this program are affected when aging effects are discovered.

HNP Response:
The administrative procedure for the Structures Monitoring Program requires a reassessment and documented justification for an appropriate periodic inspection interval for each License Renewal structural system (structure) based on the results of the inspection. The administrative procedure for the Structures Monitoring Program also states the inspection interval shall be commensurate with the safety significance of the structure and its condition but shall not exceed ten (10) years. Based on this, the inspection intervals for License Renewal structures varies from structure to structure. For example, the Reactor Containment Building internal concrete is examined at 5 year intervals, the Fuel Handling Building at 7 year intervals, the Turbine Building at 8 year intervals, and several non-safety related structures at 9 year intervals.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.31-JW-03

NRC Request:
Has HNP ever discovered ground water infiltration through underground exterior walls of buildings within the Structures Monitoring Program? If yes provide the documentation for these issues showing when, where and how they were discovered. Also, how these issues were evaluated and resolved with a discussion on the need for any follow up inspections.

HNP Response:
The Structures Monitoring Program inspects the building structures for groundwater or other water seepage as one of the inspection attributes. During the baseline inspections (per the Maintenance Rule) of the structures in 1996, groundwater intrusion and/or seepage was recorded on a number of structures. The structures with groundwater intrusion and/or seepage included the Reactor Auxiliary Building, the Fuel Handling Building, and the Waste Processing Building. The groundwater intrusion and/or seepage was evaluated as acceptable by HNP Engineering and a comment was added that monitoring should be continued. However, at one location, the Reactor Auxiliary Building was recorded as unacceptable due to water seepage into the 216' elevation pipe tunnel and from water seepage from seismic gaps and penetrations. This was addressed with a plant modification. Other actions taken to remove/direct water in-leakage included the establishment of additional measures; such as caulking concrete cracks, using tygon tubing for drainage, and cutting channels into concrete floors, to direct water in-leakage towards floor drains. During the structure inspections in 2006, groundwater intrusion and/or seepage was recorded again on a number of structures, including the Reactor Auxiliary Building, the Fuel Handling Building, the Waste Processing Building, and the Turbine Building. However, all of the locations were recorded as acceptable by HNP Engineering and a comment was added that monitoring should be continued. The groundwater intrusion and/or seepage in some of the lower elevations is nearly continuous but is minimal and is adequately removed with the floor drains system. HNP does consider the minimal intrusion and/or seepage an internal flooding concern. In general, after identifying the specific groundwater intrusion and/or seepage, the groundwater intrusion and/or seepage is identified to the Maintenance organization as a housekeeping condition through the work management process. The specific areas are cleaned up and groundwater may be redirected as appropriate. Corrosion to any support steel in the areas due to the groundwater intrusion is removed and recoated as identified through system and structure walkdowns. If the groundwater intrusion and/or seepage is recorded as unacceptable by HNP Engineering, then the condition is entered into the corrective action program for resolution. There are currently no long term plans identified from Structures Monitoring Program inspections to eliminate the groundwater intrusion and/or seepage. However, HNP Engineering will continue to monitor the groundwater intrusion and/or seepage as part of the Structures

Monitoring Program and has decreased the inspection intervals for the affected structures from ten (10) years [Reactor Auxiliary Building (6 years), Fuel Handling Building (7 years), Turbine and Waste Processing Building (8 years)]. HNP Maintenance will continue to maintain the water control measures installed to route water in-leakage towards plant floor drains.

Historically, groundwater intrusion into the Reactor Auxiliary Building lower elevations was identified as early as 1980. Pressure grouting in 1984, and sealant injection, and other water control techniques since then have been utilized to help control the groundwater intrusion but some groundwater intrusion is still ongoing as identified by the structure inspection reports in 1996 and 2006. Inspection reports are available for review at HNP.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.32-JW-01

NRC Request:

Prior to the period of extended operation one enhancement to the B.2.32 Program is to revise the administrative controls that implement the program to 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. Since NCRs are not currently used, explain how unacceptable conditions are now documented under the program and processed for engineering evaluation or corrective action.

HNP Response:

Non Conformance Reports (NCRs) are required to be initiated by all employees according to the Corporate Corrective Action Program for unacceptable conditions such as a deficiencies or deviations in an item or activity that has affected or reasonably could affect nuclear safety or quality. The enhancement to the administrative procedure for the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants Program improves the administrative procedure by clarifying the corporate requirement. This enhancement also makes this administrative procedure consistent with the corporate level procedure used for the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants Program.

Question No: B.2.33-RM-01

NRC Request:
In LRA, Appendix B, Sections B.2.33, B.2.34, B.2.35, B.2.36, and B.2.37, the applicant states that these programs are new with no site-specific operating experience history. The SRP-LR, Revision 1, Appendix A, Branch Technical Position RLSB-1 states that an applicant may have to commit to providing operating experience in the future for new program to confirm their effectiveness. Describe how operating experience will be captured to confirm the program effectiveness and the process to be used to adjust the program as needed.

HNP Response:
Plant specific and industry wide operating experience (OE) was considered in the development of the Appendix B electrical programs. Industry operating experience that forms the basis for these Appendix B electrical programs is included in the operating experience element of the corresponding NUREG-1801 Chapter XI Programs. Plant specific operating experience was reviewed to ensure that the NUREG-1801 Chapter XI Programs will be effective aging management programs for the period of extended operation (PEO). This review is discussed in a plant evaluation. This review confirms that the operating experience discussed in the NUREG-1801 Chapter XI Programs is bounding. Operating experience going forward will be captured through the HNP Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of operating experience will continue throughout the PEO and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the HNP QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the Appendix B electrical programs continue to be effective in the management of aging effects.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.33-RM-02

NRC Request:
In LRA AMP Section B.2.33, the applicant states that the technical basis for selecting the sample of cables and connections to be inspected is defined in the implementing HNP program document. The staff requests the applicant to discuss the following: (1) Explain the sample selection method used for cables and connections from accessible areas such that they are inspected and represent, with reasonable assurance, all cables and connections, and (2) If an unacceptable condition or situation is identified for a cable or connection in the inspection sample, explain the inspection sample expansion and corrective actions.

HNP Response:
(1) The sample selection method used in the implementing HNP program document follows the guidance of NUREG-1801, Section XI.E1, whereby a representative sample of accessible electrical cables and connections installed in adverse localized environments are visually inspected and represent, with reasonable assurance, all cables and connections in that area. 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. The HNP program utilizes plant operating experience (OE) to determine the plant areas to be inspected. HNP OE is used to identify past cable failures, cables that exhibited the effects of aging, hot spots, and adverse localized environments. Part of this OE review includes conversations with maintenance personnel and the use of environmental surveys. Based on this review of OE, the plant areas to be inspected become localized in nature, consisting of a limited area (or subset) of a much larger plant area or zone. The sample selection of cables and connections inspected within the limited plant area bound all cables and connections in the area since the inspection focuses on the worst case environments.

(2) Corrective actions will be implemented through the HNP Corrective Action Program and may include, but are not limited to, testing, shielding or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination will be made as to whether this same condition or situation could be applicable to other accessible or inaccessible insulated cables and connections. The Corrective Action Program is implemented by the HNP QA Program in accordance with 10 CFR 50, Appendix B.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.34-RM-01

NRC Request:
In LRA, Appendix B, Section B.2.34, the applicant describes the AMP for Non-EQ Instrumentation Circuits Program. Clarify whether the tests proposed in the program include both cables and connections. If not, address how the cables and connections are tested.

HNP Response:
The tests proposed in the program include both cables and connections used in radiation monitoring and nuclear instrumentation circuits. The term "cable systems" used in the LRA refers to the combination of cables and connections.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.34-RM-02

NRC Request:
In LRA, Appendix B, Section B.2.34, the applicant states that for radiation monitoring circuits and the RG 1.97 wide range neutron flux 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. Clarify if radiation monitoring and wide range neutron monitoring cable systems are disconnected during calibration or surveillance testing.

HNP Response:
The radiation monitoring cable systems are initially disconnected at the channel drawers for calibration of individual circuit electronics. Loop calibration or surveillance testing is performed utilizing a check source with the cable systems connected. 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.

The RG 1.97 wide range neutron flux monitoring cable systems are also disconnected at the channel drawers for calibration of individual circuit electronics but are not equipped with a check source. Therefore, the cable systems used in the RG 1.97 wide range neutron flux monitoring circuits require testing to identify the potential existence of cable system aging degradation. The RG 1.97 wide range neutron flux monitoring circuits are part of the Excore Nuclear Instrumentation System. Similar to the cable systems used in the excore source, intermediate, and power range nuclear instrumentation circuits, the RG 1.97 wide range neutron flux monitoring circuits 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.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.34-RM-03

NRC Request:
In LRA Section B.2.34, the applicant states that exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced insulation resistance. Why is moisture not specified as a cause for reduced insulation resistance as specified in GALL Report AMP XI.E2? Also, clarify whether all instrumentation circuits susceptible to moisture sensitive to signal inaccuracies are included in the EQ.

HNP Response:
Moisture is specified as a cause for reduced insulation resistance as shown in LRA Table 3.6.2-1 (page 3.6-19). LRA Section B.2.34 is meant to provide summary level program information and not intended to exclude moisture. Note that the conclusion for LRA Section B.2.34 (page B-94) includes moisture as well as heat and radiation. Also, moisture is included as a stressor under the environments shown in LRA subsection 3.6.2.1.1 (page 3.6-4).

Not all instrumentation circuits susceptible to moisture, sensitive to signal inaccuracies are included in the HNP EQ Program. To discover that population of circuits not included in the HNP EQ Program, all impedance sensitive circuits within the scope of License Renewal that may experience a reduction in insulation resistance (IR) due to heat, radiation or moisture were screened against the criteria given in NUREG-1801, Section XI.E2. The resultant list of impedance sensitive neutron and radiation monitoring signal cables that may experience a reduction in IR are included in LRA, Appendix B, Section B.2.34.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.35-RM-01

NRC Request:
Are all medium voltage cables within the scope of license renewal included in HNP AMP B.2.35 (GALL Report AMP XI.E3). If not, provide a listing of the Medium Voltage cables installed at HNP and show how they were screened out for this program.

HNP Response:
No. Medium voltage cables within the scope of License Renewal that did not meet the criteria specified in NUREG 1801 Section XI.E3 are not included in HNP AMP B.2.35. All medium voltage cables within the scope of License Renewal were screened against the criteria given in NUREG 1801 Section XI.E3. Consistent with NUREG-1801, Section XI.E3, medium voltage cables included in HNP AMP B.2.35 meet the following criteria: (1) they are located underground and assumed wet, and (2) they must be energized at least 25% of the time. HNP medium voltage cables within the scope of License Renewal that did not meet these criteria were screened out and are not included in HNP AMP B.2.35.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.37-XI.E6-RM-01

NRC Request:
GALL Report AMP XI.E6 states that if an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested. Please clarify whether this recommendation will be implemented for HNP LRA Section B.2.37. If not, explain why this recommendation is not applicable to HNP.

HNP Response:
As stated in HNP LRA Section B.2.37, the HNP Connections Program is consistent with GALL Report AMP XI.E6. As specified in the program basis document, if an unacceptable condition or situation is identified in the selected sample, a determination will be made as to whether the same condition or situation is applicable to other connections not tested.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.2.37-XI.E6-RM-02

NRC Request:
In calculation No.HNP-P/LR-0668 (AMP XI.E6 evaluation), the applicant states that the program elements "Scope of Program," and "Detection of Aging Effects," program elements are not consistent with GALL AMP XI.E6 program elements "Scope of Program," and "Detection of Aging Effects." However, LRA Section B.2.37 states that all elements of this program are consistent with GALL AMP XI.E6. Please identify these exceptions in LRA Section B.2.37 and its technical justifications.

HNP Response:
This program was revised after the original submittal of the HNP LRA in 2006. The basis for this revision was the NRC Letter dated March 16, 2007, "Staff Response to the Nuclear Energy Institute (NEI) White Paper on Generic Aging Lessons Learned (GALL) Report Aging Management Program (AMP) XI.E6, 'Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirement'". The exceptions to LRA Section B.2.37 are described below. Under program element "Scope of Program" GALL AMP XI.E6 states "Connections associated with cables in scope of license renewal are part of this program, regardless of their association with active or passive components." Consistent with the clarification provided in the NRC letter, this element of calculation No.HNP-P/LR-0668 was revised to read "The HNP AMP applies to cable connections within the scope of license renewal not covered under the existing EQ program. The scope of this program includes only external cable connections terminating at an active device such as motor, motor control center, switchgear or of a passive device such as a fuse cabinet. Wiring connections internal to an active assembly installed by manufacturers are considered a part of the active assembly and therefore are not within the scope of this program." Under program element "Detection of Aging Effects" GALL AMP XI.E6 states ""Electrical connections within the scope of license renewal will be tested at least once every 10 years. Testing may include thermography, contact resistance testing, or other appropriate testing methods. This is an adequate period to preclude failures of the electrical connections since experience has shown that aging degradation is a slow process. A 10-year testing interval will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed before the period of extended operation." Consistent with the test frequency flexibility provided in the NRC letter, this program element was revised to read "This program will be implemented as a one-time inspection on a representative sample of non-EQ cable connections within the scope of license renewal prior to the period of extended operation. Inspection methods may include thermography, contact resistance testing, bridge balance testing, or other appropriate testing methods. This one-time inspection verifies that the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an

aging effect that requires a periodic aging management program.

GALL AMP XI.E6 along with the clarification provided in the NRC letter forms the technical basis and justification for the HNP Program described in LRA Section B.2.37. These exceptions to GALL AMP XI.E6 will be noted in an amendment to LRA Section B.2.37.

A License Renewal Application amendment is required.

Question No: B.3.2-RM-01

NRC Request:
In LRA, Appendix B, Section 3.2, the applicant states that the HNP EQ program has been effective at managing aging effects and the overall effectiveness of the program is demonstrated by the excellent operating experience for the systems and components in the program. Discuss the details of operating experience that led to this conclusion. Show where an existing program has succeeded and where it has failed in identifying aging degradation in a timely manner with the present program.

HNP Response:
The HNP EQ Program is continuously monitoring the qualification basis for all equipment in the EQ Program, including aging effects and their impact on the qualified lives of EQ equipment. Specific examples of activities that have taken place based on these monitoring activities are provided below:

Plant Change Request (PCR) 5809 (completed in March of 1991) was developed in order to evaluate the EQ impact of containment temperature data that had been obtained from temporary modification PCR 3315. PCR 3315 installed 11 RTD's to determine the actual temperatures in Containment. The results of PCR 5809 included the re-calculation of the qualified lives of 12 EQ documentation package (EQDPs) in order to assure that qualified lives of components were met.

PCR 6406 (completed in March of 1994) evaluated the Main Steam Tunnel (MST) qualified life calculations based on an evaluation of outdoor temperatures. This increase in outdoor temperature ultimately resulted in a Technical Specification/FSAR change to increase the MST ambient temperature. The PCR revised all EQ documentation impacted by the Technical Specification/FSAR change.

The information regarding PCR 5809 and PCR 6406 are examples of actions taken under the EQ Program to assure that environmental qualification of components was maintained.

NCR 133798 (completed in August 2004) was written because a slow stroke time on a Service Water solenoid operated valve (SOV) indicated a problem with the valve's solenoid. An incorrect assumption on the energization time of the solenoid led to the initiation of Engineering Change (EC) 59305. This EC changed the EQ documentation package for two SOVs to reflect the accurate service life energization time of these EQ components. NCR 133798 is an example of the EQ Program reacting to operating experience data to assure the continued environmental qualification of equipment.

The HNP EQ Program is required to be assessed by knowledgeable personnel from outside of the site EQ group at an interval of no more than 4 calendar years. Self Assessment (SA) 80126 (conducted in August 2003) was the latest formal assessment of the EQ program. Although this SA discovered a variety of improvement opportunities, there were no issues or findings which impact program effectiveness. This SA is an example of continuous self improvement.

The HNP EQ Program System Health Report is a web based document used to indicate the overall heath of the EQ Program and to proactively identify declining trends in the program.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.3.2-RM-02

NRC Request:
GALL Report AMP X.E1 states that important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). Explain why these attributes are not discussed in HNP LRA Section B.3.2 as recommended in GALL Report AMP X.E1.

HNP Response:
These attributes are discussed in Subsections 4.4.1 and 4.4.2 of the HNP LRA.

HNP LRA Section B.3.2 will include a statement that refers to LRA Section 4.4 for a discussion of EQ Program reanalysis attributes.

A License Renewal Application amendment is required.

Question No: B.3.2-X.E1-RM-02

NRC Request:
In calculation No.HNP-P/LR-0650 (AMP X.E1 evaluation), the applicant states that the HNP EQ program conforms to Regulatory Guide 1.89, Rev.0, not Rev. 1. However, GALL AMP X.E1 recommends Regulatory Guide 1.89, Rev.1, as the regulatory guidance for complying with 10 CFR 50.49. Explain why LRA Section B 3.2 program elements "Parameters Monitored/Inspected," and "Scope of Program," program elements did not identify this exception. Provide technical justification for this exception.

HNP Response:
The HNP EQ programs licensing basis is Regulatory Guide 1.89, Rev. 0. This is an exception to GALL AMP X.E1 which references Regulatory Guide 1.89, Rev. 1. The original licensing basis of the HNP EQ Program is not Regulatory Guide (RG) 1.89, Rev. 1, as specified in NUREG-1801, Section X.E1. HNP was originally licensed as a NUREG-0588, Category II plant, and IEEE Std 323-1971 was the original EQ Program basis. RG 1.89, Rev. 1 had not been issued when the HNP construction permit SER was issued. RG 1.89, Rev. 1 describes a method acceptable to the NRC staff for complying with § 50.49 of 10 CFR Part 50. The acceptable method follows the procedures described by IEEE Std 323-1974. Currently, the HNP EQ Program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. Under 10 CFR §54.21(c)(1)(iii), the HNP EQ Program, which implements the requirements of 10 CFR 50.49, is viewed as an aging management program for License Renewal. Section 4.4.2.1.3 of the "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR) states that the staff evaluated the EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii).

This exception will be identified in an amendment to the LRA.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: GALL XI-RM-01

NRC Request:
Please identify all non-safety electrical/I&C containment penetration assemblies at HNP? Identify the aging effects and explain how these penetration assemblies are managed.

HNP Response:
All electrical/I&C containment penetration assemblies at HNP are in the EQ Program. Their EQ documentation package (EQDP) is considered a TLAA.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.5.1-06-JW-01

NRC Request:
LRA Table 3.5.1, Item 3.5.1-06, refers to LRA Subsection 3.5.2.2.1.4 in the discussion column. In Subsection 3.5.2.2.1.4, the following statement is made: ACI 201.2R was not used as guidance for concrete mix proportions, but ACI 211.1-74 was used. ACI 211.1-74 provides guidance for producing high-density, low permeability concrete mix designs similar to ACI 201.2R. Provide a comparison of the similarities and differences between ACI 201.2R and ACI 211.1-74 for concrete mix proportion designs as they relate to HNP concrete specifications.

HNP Response:
The design of concrete mix in contact with the Containment liner (LRA Table 3.5.1, Item 3.5.1 06) at HNP was in accordance with ACI 211.1-74, "Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete," and also in accordance with Article CC-2232 of the ASME Code Section III, Division 2/ACI 359 Code [FSAR 3.8.1.6.1(f)]. LRA Section 3.5.2.2.1.4 discusses loss of material due to corrosion for the Containment liner, liner anchors, and integral attachments. HNP FSAR Section 3.8.1.5.4 states "The alkaline environment of the concrete adequately protects embedded steel parts from corrosion." ACI 201.2R (Section 4.5.1.1) states "Low water-cement ratios produce less permeable concrete and thus provide greater assurance against corrosion." Therefore, water-cement ratio is of primary importance in the discussion provided in the LRA Section 3.5.2.2.1.4.

Selection of the water-cement methodology is the same between the ACI 211.1-74 [Table 5.3.4(b)] and ACI 201.2R, "Guide to Durable Concrete." ACI 211-74 specifies a maximum water-cement ratio of 0.50 for "All other structures" with a footnote that it is based on ACI 201. ACI 201.2R (Section 1.4.2) also specifies a maximum water-cement ratio of 0.50 for "All other structures." The Containment concrete should be included in the "All other structures" category. The actual concrete mix designs at HNP for the Containment concrete were within the water-cement ratios specified in both ACI Codes.

Air entrainment is also an important element in designing a durable, low permeable concrete. Selection of the air content is similar between the two ACI codes. ACI 211-74 (Table 5.3.3) specifies an approximate average air content of 6% for ¾" aggregate and 4 ½ % for 1 ½" aggregate and adds a statement in Section 5.3.3 to see ACI 201 on air content recommendations. ACI 201.2R (Table 1.4.3) recommends an average air content of 5% for ¾" aggregate and 4 ½ % for 1 ½" aggregate with a 1 ½ % tolerance (or 6 ½ % and 6 % respectively). The actual mix designs at HNP for the Containment allowed up to 8% air entrainment for two of the three mixes for the Containment (using ≤ ¾" maximum size aggregate). This is slightly higher than the 6 ½ % and 6 %. However, the concrete mix designs used at HNP allowed the higher air content while still exceeding the concrete design strength requirements

ACI 201.2R (Sections 1.4 and 1.4.4) recommends suitable materials for producing durable, low permeable concrete. While not specifically discussed in ACI 211-74, the HNP FSAR (Section 3.8.1.6.1) and the original concrete specification identify the concrete materials specifications used at HNP, which are consistent with ACI 201.2R.

ACI 201.2R (Section 4.5.1.1) recommends use of lower water cement ratios for concrete in sea or brackish water (0.40) but this is not in ACI 211-74. However, the sea or brackish water is not applicable to HNP.

The original HNP concrete specification specified a water-cement ratio between 0.44 and 0.60 and the air content *specified as 4-8% for maximum aggregate size ¾" and 3-6% for maximum aggregate size 1 1/2". The actual mix design* for the Containment concrete as discussed above in this response was within the water-cement ratio and air content limits in the original HNP concrete specification.

Based on a review of OE and discussions with engineering, no aging effects have been identified for Containment concrete related to mix designs including loss of material due to corrosion.

Details are available at HNP for review in the bases and other reference documents.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.5.1-10-JW-01

NRC Request:
LRA Table 3.5.1, Item 3.5.1-10, refers to LRA Subsection 3.5.2.2.1.7 in the discussion column. In Subsection 3.5.2.2.1.7, the following statement is made: (2) to be susceptible to SCC, stainless steel must be subjected to both high temperature (>140 degrees F) and an aggressive chemical environment. Provide the maximum operating temperatures for the HNP components associated with item 3.5.1-10.

HNP Response:
The maximum operating temperature which the applicable stainless steel penetrations are subject to is 668 °F for several sampling lines. However, none of the applicable stainless steel components associated with LRA Table 3.5.1, Item 3.5.1-10 are subject to an aggressive chemical environment. According to HNP aging management review methodology, both high temperature (>140 °F) and an aggressive chemical environment are required for the SCC aging effect to be applicable. The basis document for determining aging effects for stainless steel material is available for review at HNP.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.5.1-26-JW-01

NRC Request:
LRA Table 3.5.1, Item 3.5.1-26, is associated with GALL item III.A3-5. Provide HNP original concrete specifications to confirm that existing concrete had air content of 3% to 6% and water to cement ratio of 0.35-0.45 when poured.

HNP Response:
LRA Table 3.5.1, Item 3.5.1-26, is associated with GALL Item III.A3-6, not GALL Item III.A3-5. LRA Table 3.5.1, Item 3.5.1-26 has further information provided in LRA Section 3.5.2.2.2.2.1.

The actual concrete mix design for the Class I structures monitored by the Structures Monitoring Program had air content ranging from 3% to 8% and water-cement ratios up to 0.50. The actual concrete mix design for the non-Class I structures monitored by the Structures Monitoring Program had air content ranging from 3% to 8% and water-cement ratios up to 0.592.

Based on the actual mix designs exceeding the limits in NUREG-1801, LRA Section 3.5.2.2.2.2.1 states that HNP will examine inaccessible non-Class I concrete used for the structures in scope for License Renewal when excavated for any reason. LRA Table Item 3.5.1-26 states the Structures Monitoring Program is used to manage aging effects of loss of material and cracking due to freeze thaw for accessible concrete for the safety related and non-safety related structures. In addition, while not currently stated in LRA Section 3.5.2.2.2.2.1 or Table Item 3.5.1-26, all inaccessible concrete (non-Class I and Class I) will be examined for loss of material and cracking when below-grade concrete is exposed for any reason prior to backfilling. This is stated in LRA Program B.2.31, in the Enhancements for the Scope of Program and for the Parameters Monitored/Inspected.

Details are available at HNP for review in the bases and other reference documents.

For clarification, LRA Section 3.5.2.2.2.2.1 will be revised to state inaccessible Class I concrete used for the structures in scope of License Renewal will be examined for loss of material and cracking when below-grade concrete is excavated for any reason. On the basis of this response, the LRA will be amended to incorporate this clarification to LRA Section 3.5.2.2.2.2.1.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.5.1-32-JW-01

NRC Request:
LRA Table 3.5.1, Item 3.5.1-32, refers to LRA Subsections 3.5.2.2.2.1 and 3.5.2.2.2.2.5 in the discussion column. In Subsection 3.5.2.2.2.2.5, the following statement is made: For inaccessible areas in structures outside the Containment Building, safety related, Class 1 concrete was constructed to ACI 211.1-74, which provides guidance for producing high-density, low permeability concrete similar to ACI 201.2R for concrete mix designs. Therefore, no aging management program is required for inaccessible areas in safety related structures outside the Containment Building. However, Table 1 line 3.5.1-32 is referenced in AMR Tables 3.5.2-2, 3.5.2-10, 3.5.2-12, 3.5.2-17, 3.5.2-27 and 3.5.2-28 for concrete exterior below grade and concrete foundation for managing the aging effect of change in material properties with the Structures Monitoring Program. Explain the contradiction since these six AMR tables are for safety related structures and based on the statement above no aging management program is required.

HNP Response:
HNP License Renewal inadvertently included Table 1 line item 3.5.1-32 on AMR Tables 3.5.2-2, 3.5.2-10, 3.5.2-12, 3.5.2-17, 3.5.2-27 and 3.5.2-28 for the concrete exterior below grade and concrete foundation component/commodity groups. Table 1 line item 3.5.1-32 should be removed from AMR Tables 3.5.2-2, 3.5.2-10, 3.5.2-12, 3.5.2-17, 3.5.2-27 and 3.5.2-28 for the concrete exterior below grade and concrete foundation component/commodity groups.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.6.2-1-RM-01

NRC Request:
In LRA Table 3.6.2-1, plant-specific Note 604, the applicant states that the fuse holders within the scope of the aging management review are used only in radiation monitoring I&C circuits. Provide details about the review criteria used for this determination and also provide details regarding the results of the plant walkdowns performed (number and condition of fuse holders inspected, etc.).

HNP Response:
HNP fuses were screened against the criteria given in NUREG 1801 Section XI.E5. The following is a summary of the HNP screening process.

The vast majority of fuse holders at HNP are located in active devices, such as control panels, switchgear, MCCs and termination cabinets. To discover the population of fuse holders located outside of these active components, a query was developed showing all HNP fuses within the scope of License Renewal. This produced a list of approximately 2600 items. Then, control wiring diagrams, plant engineering expertise and the equipment database (EDB) were used to determine which of these in-scope fuses were located within an active device, so that they could be eliminated from the process. This reduced the original list down to less than 40 fuses, all of which were installed only in radiation monitoring I&C circuits. A walkdown of the remaining fuses found them to be in an air-conditioned environment with no external signs of aging degradation.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 3.6.2-1-RM-02

NRC Request:
In LRA Table 3.6.2-1, the applicant states that high-voltage power cables have no aging effects requiring management. The plant-specific Note 602 for these cables states that the HNP PILC cables are similar in design to the Turkey-Point medium voltage cables and were evaluated and accepted by the staff. The staff notes that HNP cables are not the same as Turkey Point cables. HNP cables are oil filled high-voltage cables operated at 230kV. The HNP cable operating characteristic and its life depends on its dielectric properties and the applicant needs to address how it plans to address the aging effects on its technical merit.

The staff requests the applicant to discuss the following: (1) Which AMP addresses periodic testing of insulating oil in the cable system to prevent degradation of its dielectric properties, (2) Which AMP addresses the vendor recommended maintenance for oil-filled cable system during the period of extended operation (PEO), (3) Provide details of periodic visual inspections and walkdowns performed to date and proposed for the PEO to monitor for oil leakage including checking the torque of the pothead bolts, and (4) Explain the instrumentation including any alarms provided to monitor the oil levels for the cable system. Also, provide details regarding existing surveillance and calibration for these instruments and during the PEO.

HNP Response:
The HNP cables are high-voltage, oil-filled, paper insulated, lead-sheathed cables. The lead sheath is designed to prevent moisture from penetrating the cable and degrading the cables insulation. The HNP cables have an Okolene (black polyethylene) outer jacket. The lead sheath combined with the overall PE jacket has proven to be an effective barrier against moisture. HNP will clarify Table 3.6.2-1, Note 602 in an amendment to the LRA.

The mechanical components that support the oil-filled cable system are evaluated in Sections 2.3.3.81, and Table 3.3.2-69 (page 3.3-426) of the HNP LRA. Currently, the System Engineer performs visual inspections and walkdowns of the oil-filled cable system on a quarterly basis. For the PEO, external visual inspections of the cable systems oil filled tanks will become part of the External Surfaces Monitoring Program as shown in Table 3.3.2-69 (page 3.3-426) of the HNP LRA. The External Surfaces Monitoring Program is described in Section B.2.22 of the HNP LRA. The program includes visual inspections of the oil filled tanks, piping and piping components for items such as, leaking components, seepage, loose bolts or threaded fasteners, and damaged or missing parts.

To preserve the electrical continuity function of the oil-filled cable system during the PEO, a power factor (Doble) test will be performed on the oil-filled cables. This test will measure dielectric losses of the cables insulation to provide an indication of a breakdown of the cables insulation properties. The oil-filled cables are to be tested at least once every four years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. A four-year testing interval will provide multiple data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for License Renewal are to be completed prior to the PEO. The elements of this test program will be provided in an aging management program (AMP).

The insulating oil environment of the cable system is documented in Table 3.0-1 (page 3.0-7) of the HNP LRA. Periodic testing of the insulating oil in the cable system is not a vendor recommended activity. This is a closed system, with no moving parts, that should remain closed so as not to introduce contaminants. This activity would be performed as a corrective action based on the results of power factor testing. Corrective actions such as testing the insulating oil will be implemented through the HNP Corrective Action Program. The Corrective Action Program is implemented by the HNP QA Program in accordance with 10 CFR 50, Appendix B.

System leakage discovered during conduct of the External Surfaces Monitoring Program would warrant the need for corrective actions during the PEO. Corrective actions such as checking the torque of the pothead bolts and/or repairing leaking fittings will be implemented through the HNP Corrective Action Program. The cable system oil-filled tanks are equipped with high-low pressure switches that are calibrated every other outage by the Transmission Dept under Interface Agreement with the site. The pressure switches provide annunciation on a common alarm panel in the Relay Building located in the 230KV Switchyard and reflash in the Energy Control Center. The Energy Control Center calls the HNP control room directly should they receive an alarm. The HNP control room would, in turn, dispatch an outside Auxiliary Operator, and call the site Plant Transmission Activities Coordinator (PTAC) to investigate the nature of the alarm.

A License Renewal Application amendment is required.