Progress Energy

AUG 3 1 2007

SERIAL: HNP-07-119
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 2: CHANGES RESULTING FROM RESPONSES TO SITE AUDIT QUESTIONS REGARDING TIME-LIMITED AGING ANALYSES

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

Ladies and Gentlemen:

On November 14, 2006, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, requested the 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.

The Nuclear Regulatory Commission (NRC) staff conducted audits of HNP License Renewal activities related to time-limited aging analyses (TLAAs) during the periods from May 21 to 25, from June 25 to 29, and from August 13 to 15, 2007. In the course of these audits, questions were identified by the auditors. Responses to these TLAA-related questions are enclosed.

Also, enclosed is the list of regulatory commitments supporting License Renewal modified to reflect the information provided in the responses to TLAA-related audit questions. 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, 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

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P.O. Box 165
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changes to the LRA and the source of those changes. Enclosure 3 is a report of TLAA-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 3 1 2007).

Sincerely,

Thomas J. Natale
Manager - Support Services
Harris Nuclear Plant

TJN/mhf

Enclosures:

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

cc:

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 2				
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, 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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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 (not to exceed once every 18 months) of cumulative usage, (4) describe the acceptance criteria for maintaining fatigue usage below the design limit, and (5) address corrective actions, to be implemented through the Corrective Action Program, for components that have exceeded alarm limits, with options to include a revised fatigue analysis or repair or replacement of the component.	A.1.1.38	Prior to the period of extended operation	Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program LRA Section B.3.1, Response to Audit Questions B.3.1-RH-01 and B.3.1-RH-05
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

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35	When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.	A.1.2.2.11	As stated in the commitment	TLAA - Leak-Before-Break evaluation for Alloy 82/182 Welds LRA Section 4.3.4, Response to Audit Question LRA 4.3.4-1.

Amendment 2 Changes to the License Renewal Application

<table>
<tr><th>Source of Change</th><th>License Renewal Application Amendment 2 Changes</th></tr>
<tr><td>Audit Questions B.3.1-RH-01 and B.3.1-RH-05</td><td>Revise enhancements (3) and (5) of LRA Subsection A.1.1.38 to read:

(3) include a provision to utilize online fatigue analysis software for the periodic updating (not to exceed once every 18 months) of cumulative usage,
(5) address corrective actions, to be implemented through the Corrective Action Program, for components that have exceeded alarm limits, with options to include a revised fatigue analysis or repair or replacement of the component.

Revise LRA Subsection B.3.1 to address alarm limits, periodic updates, and corrective actions by revising the following Enhancements in LRA Subsection B.3.1:
<u>Program Elements Affected</u>
• Detection of Aging Effects
Enhance the Program to utilize online fatigue analysis software for the periodic updating (at least once every 18 months) of cumulative fatigue usage calculations for high fatigue usage RCPB (including auxiliary system) components.
• Corrective Actions
Enhance the Program to address corrective actions if an analyzed component is determined to have exceeded the alarm limit, with options to revise the fatigue analysis, repair, or replace the component. Corrective actions if required will be implemented through the HNP Corrective Action Program.

These changed enhancements impact License Renewal Commitment #32. In addition, delete the final sentence from the Operating Experience discussion in LRA Subsection B.3.1.</td></tr>
<tr><td>Audit Question LRA 4.3.3-1</td><td>Replace LRA Table 4.3-3 with the new Table 4.3-3 and accompanying notes shown in revised LRA Section 4.3, Metal Fatigue, in Enclosure 2, Attachment 1. Enclosure 2, Attachment 1 is a new LRA Section 4.3 that replaces Section 4.3 in the original LRA. Owing to the scope of changes necessitated by the TLAA audit questions, the entire section was revised. Also, LRA Appendix A, Subsection A.1.2.2, Metal Fatigue, which is based on Section 4.3, was rewritten and is provided in Enclosure 2, Attachment 2.

In addition, the revisions to the fatigue analyses have resulted in changes to the methods used to comply with 10 CFR 54.21(c)(1) as shown on LRA Table 4.1-1. Therefore, the following changes are required to that table:
<table>
<tr><td>Explicit Fatigue Analysis (NSSS Components)</td><td>-</td><td>4.3.1</td></tr>
<tr><td>Reactor Vessel</td><td>54.21(c)(1)(ii)</td><td>4.3.1.1</td></tr>
<tr><td>Reactor Vessel Internals</td><td>54.21(c)(1)(ii)</td><td>4.3.1.2</td></tr>
<tr><td>Control Rod Drive Mechanism</td><td>54.21(c)(1)(iii)</td><td>4.3.1.3</td></tr>
<tr><td>Reactor Coolant Pumps</td><td>54.21(c)(1)(i)</td><td>4.3.1.4</td></tr>
<tr><td>Steam Generators</td><td>54.21(c)(1)(iii)</td><td>4.3.1.5</td></tr>
<tr><td>Pressurizer</td><td>54.21(c)(1)(iii)</td><td>4.3.1.6</td></tr>
<tr><td>Reactor Coolant Pressure Boundary Piping (ASME Class 1)</td><td>54.21(c)(1)(iii)</td><td>4.3.1.7</td></tr>
<tr><td>Implicit Fatigue Analysis (ASME Class 2, Class 3, and ANSI B31.1 Piping)</td><td>-</td><td>4.3.2</td></tr>
<tr><td>ASME Class 2 and 3 Piping</td><td>54.21(c)(1)(ii)</td><td>4.3.2.1</td></tr>
<tr><td>ANSI B31.1 Piping</td><td>54.21(c)(1)(ii)</td><td>4.3.2.2</td></tr>
<tr><td>Environmental Fatigue Analysis</td><td>-</td><td>4.3.3</td></tr>
<tr><td>RCS Loop Piping Leak-Before-Break Analysis</td><td>54.21(c)(1)(ii)</td><td>4.3.4</td></tr>
<tr><td>Cyclic Loads That Do Not Relate to RCS Transients</td><td>-</td><td>4.3.5</td></tr>
<tr><td>Primary Sample Lines</td><td>54.21(c)(1)(i)</td><td>4.3.5.1</td></tr>
<tr><td>Steam Generator Blowdown Lines</td><td>54.21(c)(1)(i)</td><td>4.3.5.2</td></tr>
</table></td></tr>
</table>

Source of Change	License Renewal Application Amendment 2 Changes
Audit Question LRA 4.3.4-1	LRA Subsection 4.3.4 and in Subsection A.1.2.2.11 have been revised to delete Numbered List Item No. 6 and add the following words to the last paragraph. These revisions are shown in Attachments 1 and 2 to this Enclosure. When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach. This change is the genesis of License Renewal Commitment #35.
Audit Question LRA 4.3.5-1	In the **Analysis** discussion of Penetration M-78A - RCS Hot Legs, of LRA Subsection 4.3.5.1, replace the last sentence in the second paragraph with the following: For the purposes of this evaluation, a penetration isolation lasting for more than 5 minutes while the RCS hot Leg temperature exceeds 500°F is conservatively considered one thermal cycle as the temperature descrease was estimated to be less than 200°F. Consequently, revise the number of thermal cycles due to penetration isolation from 22 to 30 in the second and fourth sentences of paragraph four. Also, revise the number of thermal cycles resulting from penetration isolations from 198 to 270 for the piping components associated with containment penetration M-78A. This changes the total number of cycles associated with M-78A from 1,279 to 1,351. These changes are shown in Subsection 4.3.5.1 of Enclosure 2, Attachment 1, and the changes in cycles are shown in Subsection A.1.2.2.12 of Enclosure 2, Attachment 2.
Audit Question LRA 4.3-2	The change associated with this Audit Question would have corrected the 60-year projection value for LRA Table 4.3-1, Normal Condition 10, for feedwater cycling at hot standby, from 1500 cycles to 333 cycles. However, based on other required changes to LRA Section 4.3, the 60-year projected values have been deleted from the table. Therefore, this change has been superseded.
Audit Question LRA 4.3-3	Revise the LRA description for Reactor Vessel Internals fatigue to explain that the cumulative usage factor for 40 years of operation determined during the HNP steam generator/power uprating analysis does not exceed the design limit of 1.0 when projected through the period of extended operation. Refer to the discussion in Subsection 4.3.1.2 of Enclosure 2, Attachment 1 and in Subsection A.1.2.2.2 of Enclosure 2, Attachment 2.
Audit Question LRA 4.3-4	This Audit Question asked for a justification for the extrapolation method for cycle projections described in LRA Subsection 4.3.1. However, based on other required changes to LRA Section 4.3, the cycle projection methodology description has been removed. Therefore, the justification requested is no longer required. Refer to the revised Subsection 4.3.1 provided in Enclosure 2, Attachment 1.
Audit Question LRA 4.3-5	This Audit Question asked for an explanation of how cycle projections described in LRA Subsection 4.3.1 were utilized in fatigue analyses to determine cumulative usage factors. However, based on this and other related Audit Questions, the cycle projection methodology description has been removed from LRA Section 4.3. Therefore, the explanation requested by this Audit Question is no longer required. Refer to the revised Subsection 4.3.1 provided in Enclosure 2, Attachment 1.
Audit Question LRA 4.3-9	Change Note 1 to Table 4.3-2, Design Fatigue Usage Factors, to read: Due to the original design usage factors exceeding 1.0, these bolts were originally to be replaced based on a replacement schedule. However, these fatigue usage values have been superseded by the results of the license renewal fatigue evaluation described in Subsection 4.3.1.5, and a replacement schedule is no longer required. Refer to revised Table 4.3-2 in Enclosure 2, Attachment 1.

Source of Change	License Renewal Application Amendment 2 Changes
Audit Question LRA 4.3-12	The fatigue analysis discussion in LRA Subsections 4.3.2.1 and 4.3.2.2, relative to ASME Class 2 and 3 and ANSI B.31.1 piping systems, has been revised from that in the original LRA. Replace the corresponding text in the original LRA with that in new Subsections 4.3.2.1 and 4.3.2.2 in Enclosure 2, Attachment 1, and new Subsections A.1.2.2.8 and A.1.2.2.9 in Enclosure 2, Attachment 2.
Audit Question LRA 4.3-13	Revise the **Analysis** discussion in LRA Subsection 4.3.5.2 to delete "and 100 years" from the first paragraph, and revise the second and third sentences in the third paragraph to read: Based on a review of plant data over a period of approximately 5.5 years, the number of Blowdown flow interruptions that would result in a thermal cycle were estimated to be 37 cycles. Applying a ratio based on 60 years results in 404 cycles. Refer to the revised discussion in Subsection 4.3.5.2 of Enclosure 2, Attachment 1.
Audit Question LRA 4.3-16	Revise LRA Table 4.3-1 by: 1) Change the Cycles to Date for Normal Transients 8, 9, and 15 to "NC" (Not Counted), 2) Change the Cycles to Date for Normal Transient 16 to 1, 3) Change the Cycles to Date for Test Transient 1 to 7, and 4) Provide Notes 1 through 6 to explain the above and other changes. These changes are shown on revised Table 4.3-1 in Enclosure 2, Attachment 1.
Audit Question LRA 4.3-18	Revisee LRA Table 4.3-2 by:changing the Fatigue Usage for List Number 31, Main Feedwater Nozzle, from 0.93 to 0.98. This change is shown on revised Table 4.3-2 in Enclosure 2, Attachment 1.

Enclosure 2, Attachment 1

This new Section 4.3 replaces the Section 4.3, Metal Fatigue, in the original LRA.

4.3 METAL FATIGUE

Several thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses (TLAAs) for HNP. These are discussed in the following Subsections.

Subsection	TLAA
4.3.1	Explicit Fatigue Analyses (NSSS Components)
4.3.1.1	Reactor Vessel
4.3.1.2	Reactor Vessel Internals
4.3.1.3	Control Rod Drive Mechanism
4.3.1.4	Reactor Coolant Pumps
4.3.1.5	Steam Generators
4.3.1.6	Pressurizer
4.3.1.7	Reactor Coolant Pressure Boundary Piping (ASME Class 1)
4.3.2	Implicit Fatigue Analysis (ASME Class 2, Class 3, and ANSI B31.1 Piping)
4.3.2.1	ASME Class 2 and Class 3 Piping
4.3.2.2	ANSI B31.1 Piping
4.3.3	Environmentally-Assisted Fatigue Analysis
4.3.4	RCS Loop Piping Leak-Before-Break Analysis
4.3.5	Cyclic Loads That Do Not Relate to RCS Transients
4.3.5.1	Primary Sample Lines
4.3.5.2	Steam Generator Blowdown Lines

4.3.1 EXPLICIT FATIGUE ANALYSES (NSSS COMPONENTS)

The HNP approach is to identify the latest design fatigue analyses associated with each NSSS component within the Reactor Coolant Pressure Boundary (RCPB) in order to demonstrate that the design analyses will remain bounding through the period of extended operation. Components within the scope of this review include non-pressure boundary reactor internals components.

Original fatigue design calculations assumed a large number of design transients corresponding to relatively severe system dynamics over the original 40-year design life. In general, actual plant operations have resulted in only a fraction of the originally expected fatigue duty.

A review was performed to establish the current design basis for the major NSSS components. This review showed that the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections 4.3.1.6 and 4.3.1.7). Therefore the governing transients "NSSS Design Transients" are those identified in the HNP steam generator replacement/uprating analysis. The list of design transients used in these fatigue analyses is shown in Table 4.3-1. Forty-year design Cumulative Usage Factor (CUF) values were compiled from design documents including the recent HNP steam generator replacement/uprating analysis. These design fatigue usage factors are presented in Table 4.3-2.

The next evaluation was to factor the effects of the reactor water environment on fatigue. The environmental fatigue evaluation is addressed below.

The HNP evaluation of NSSS components was used to demonstrate compliance with 10 CFR 54.21(c)(1) by using a combination of methods (i), (ii), and (iii).

The following sections provide a summary of the evaluation results for each of the major NSSS components evaluated.

4.3.1.1 Reactor Vessel

Summary Description

TLAAs have been identified for several sub-components of the Reactor Vessel. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections 4.3.1.6 and 4.3.1.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/ uprating analysis. Refer to Table 4.3-1. Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The reactor vessel fatigue analysis demonstrated that if the reactor vessel components were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0.

Analysis

For the components that are part of the Reactor Vessel, the highest 40-year design fatigue usage value is 0.37 for the closure studs. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.56. This value does not exceed the design limit of 1.0 and is therefore acceptable. This 60-year fatigue usage bounds the maximum environmentally adjusted usage factor of 0.1740 for the Reactor Vessel Outlet Nozzles shown in Table 4.3-3.

Therefore, the analysis has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

Disposition: **10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.**

4.3.1.2 Reactor Vessel Internals

Summary Description

A TLAA has been identified for the Reactor Vessel Internals. The NSSS Design Transients are identified in the HNP steam generator replacement/uprating analysis (see Table 4.3-1). Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The Reactor Vessel Internals fatigue analysis demonstrated that if the Reactor Vessel Internals were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0.

Analysis

For the Reactor Vessel Internals, the 40-year design fatigue usage value is 0.52 for the core internals. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.78. This value does not exceed the design limit of 1.0.

Therefore, the analysis has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

Disposition: **10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.**

4.3.1.3 Control Rod Drive Mechanism

Summary Description

TLAAs have been identified for the Control Rod Drive Mechanism (CRDM). The NSSS Design Transients are identified in the HNP steam generator replacement/uprating analysis (see Table 4.3-1). Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The CRDM fatigue analysis demonstrated that if the CRDM were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0.

Analysis

For the CRDM, the highest 40-year design fatigue usage value is 0.99 for the "Lower Joint Canopy Area" (see Table 4.3-2). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.49. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP Reactor Coolant Pressure Boundary Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate re-evaluation or corrective action is taken. Therefore, the effects of fatigue on the CRDM will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.4 Reactor Coolant Pumps

Summary Description

The Reactor Coolant Pumps (RCPs) have been designed and analyzed to meet the ASME Code of record. The original design fatigue analysis was performed using fatigue waiver requirements. The pumps were therefore identified as having a TLAA. The RCP fatigue analysis demonstrated that if the RCPs were exposed to a bounding set of postulated transient cycles, the fatigue waiver would remain valid.

Analysis

The HNP RCP evaluation showed that the fatigue analysis was performed using the Code [NB-3222.4(d)] waiver of fatigue requirements. The original fatigue waiver evaluation was found to be bounding for the SGR/Uprating evaluation. Therefore, it was unnecessary to determine a 40-year or 60-year fatigue usage factor for the RCPs.

Disposition: **10 CFR 54.21(c)(1)(i) – The analysis remains valid for the period of extended operation.**

4.3.1.5 Steam Generators

Summary Description

TLAAs have been identified for several sub-components of the Steam Generators. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections 4.3.1.6 and 4.3.1.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis (see Table 4.3-1). Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The Steam Generator fatigue analysis demonstrated that if the Steam Generator subcomponents were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0 with the exception of the Secondary Manway Bolts and the 4 in. Inspection Port Bolts. These components are discussed in more detail below.

Analysis

Other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts, the highest 40-year design fatigue usage value is 0.98 for minor shell taps. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required.

The HNP Reactor Coolant Pressure Boundary Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate re-evaluation or corrective action is taken. Therefore, the effects of fatigue on the steam generator (other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts) will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

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

The usage factor for the Secondary Manway Bolts and the 4 in. Inspection Port Bolts was re-analyzed using 40-year design cycles for all transients except for the Unit Loading and Unit Unloading transients. These transients were limited to 2,000 cycles each compared to the 18,300 cycles for Normal Condition transients 3 and 4. The calculated usage for the bolts based on this transient set is as follows:

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

Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields:

- Secondary Manway Cover Bolts: Fatigue Usage = 1.245
- 4 in. Inspection Port Bolts: Fatigue Usage = 1.215

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

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.6 Pressurizer

Summary Description

TLAAs have been identified for several sub-components of the Pressurizer. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see below and Subsection 4.3.1.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis (see Table 4.3-1). Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The Pressurizer fatigue analysis demonstrated that if the Pressurizer subcomponents were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0 for all components. However, certain locations of the Pressurizer lower head are not bounded by the original design fatigue analysis, because the

original fatigue analysis did not consider insurge/outsurge transients that were identified subsequent to the original fatigue analysis.

Analysis

For the Pressurizer (other than the lower head and surge line nozzle), the highest 40-year design fatigue usage value is 1.00 for the "Trunnion Bolt Hole" (see Table 4.3-2). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.50.

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

Plant data from hot functional testing to January 20, 1994, was used to establish pre-Modified Operating Procedures transients that represent past plant heat-up and cooldown operations. For post-Modified Operating Procedures operations, the plant data from July 19, 1999, to October 18, 2004, was collected and was processed. These 5.26 years of data histories, in conjunction with the pre-Modified Operating Procedures transients, were projected and used to predict 60 year fatigue usage based on maintaining current operating practices.

Fatigue evaluations of the pressurizer lower head and surge line nozzle were performed using the online monitoring and Westinghouse proprietary design analysis features of the WESTEMS™ Integrated Diagnostics and Monitoring System. The fatigue evaluations follow the procedures of ASME Code, Section III, NB-3200. The stress ranges, cycle pairing and fatigue usage factors were calculated using WESTEMS™, consistent with the ASME Code and WOG recommendations.

The fatigue evaluations were performed at critical locations of the pressurizer lower head (including the pressurizer surge line nozzle) and of the surge line RCS hot leg nozzle. The evaluations were based upon pre-Modified Operating Procedures transients in conjunction with the post-Modified Operating Procedures transients that include the effects of insurge/outsurge and surge line stratification. These transients were developed based upon plant-specific data and the information and guidelines provided by the WOG. The predicted fatigue usage was determined assuming that future operations would follow current operating procedures (i.e., the 5.26 year sample period is representative of past and future post-Modified Operating Procedures operations).

For the 40 years of plant life, the pressurizer lower head has the highest fatigue usage of 0.36 at the inside surface of the lower head at the heater penetration region. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a fatigue usage of 0.54. This location was also evaluated to account for the effects of reactor water environment on fatigue. The 60-year fatigue usage for this location is 1.35 as shown in Table 4.3-3.

For the Pressurizer, the maximum fatigue usage for 60 years of operation is 1.35. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP Reactor Coolant Pressure Boundary Fatigue Monitoring Program will ensure that the design limit fatigue usage is not exceeded or that appropriate re-evaluation or corrective action is taken. Therefore, the effects of fatigue on the pressurizer will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.7 Reactor Coolant Pressure Boundary Piping (ASME Class 1)

Summary Description

TLAAs have been identified for components of the Reactor Coolant Pressure Boundary (RCPB) Piping. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see below and Subsection 4.3.1.6). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis (see Table 4.3-1). Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis (see Table 4.3-2). The RCPB Piping fatigue analysis demonstrated that, if the RCPB piping components were exposed to a bounding set of postulated transient cycles, the CUF values for the components do not exceed 1.0. However, the Pressurizer Surge Line is not bounded by the original design fatigue analysis.

In response to NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, HNP evaluated the pressurizer surge line stratification transients separately for 40 years of operations.

Analysis

For the components that are part of the RCPB Piping, the highest 40-year design fatigue usage value is 0.98 for the Pressurizer Spray Piping (see Table 4.3-2) before the evaluation of the effects of reactor water environment on fatigue (See Subsection 4.3.3). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47.

Accounting for the effects of reactor water environment on fatigue, the highest 60-year fatigue usage is 2.120 for the pressurizer surge line as shown in Table 4.3-3.

Since these values exceed the design limit of 1.0, an aging management program is required. The HNP Reactor Coolant Pressure Boundary Fatigue Monitoring Program will ensure that the design limit fatigue usage is not exceeded or that appropriate re-evaluation or corrective action is taken. Therefore, the effects of fatigue on the pressurizer will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.2 IMPLICIT FATIGUE ANALYSIS (ASME CLASS 2, CLASS 3, AND ANSI B31.1 PIPING)

The subject ASME Class 2, Class 3 and ANSI B31.1 piping within the scope of this review is identified in FSAR Table 3.2.1-1. This section discusses piping components that did not require an explicit design fatigue analysis.

4.3.2.1 ASME Class 2 and 3 Piping

Summary Description

HNP auxiliary piping that was designed to ASME Section III, Code Class 2 and 3 requirements did not require an explicit fatigue evaluation. Instead, for Class 2 and 3 piping, the Code includes implicit treatment of fatigue using a stress range reduction factor, f, which is a function of the total number of thermal expansion stress range cycles. The factor is equal to 1.0 for up to 7,000 cycles. For greater number of cycles, f may be further reduced, thereby reducing the thermal expansion range stress allowable.

Analysis

The affected Class 2 and 3 piping is effectively an extension of the adjacent Class 1 piping. Therefore, the cycle count depends closely on reactor operating cycles and can be estimated by a review of the limiting reactor coolant system design transients in FSAR Table 3.9.1-1. Of those Normal Conditions listed that are likely to produce full-range thermal cycles in a 40-year plant lifetime are the 200 Heatup and Cooldown cycles. Assuming that all Upset

Conditions lead to full-range thermal cycles adds an additional 980 cycles for a total of 1180 occurrences. The 980 cycles are equal to the summation of Upset Condition transients 1 through 12 plus 5 OBEs at 10 cycles each. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 1,770. This is only a fraction of the 7,000 full-range thermal cycles associated with a stress range reduction factor of 1.0. Therefore, the analysis for Class 2 and 3 piping has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

Disposition: **10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.**

4.3.2.2 ANSI B31.1 Piping

Summary Description

In addition to ASME Class 2 and Class 3 piping, the scope of License Renewal at HNP includes non-safety related piping designed to ANSI B31.1. HNP auxiliary piping that was designed to ANSI B31.1 requirements did not require an explicit fatigue evaluation. Instead, for ANSI B31.1 piping, the "Power Piping" Code includes implicit treatment of fatigue using a stress allowable reduction factor, f, which is a function of the total number of thermal expansion stress range cycles. The factor is equal to 1.0 for up to 7,000 cycles. For greater number of cycles, f may be further reduced, thereby reducing the thermal expansion range stress allowable.

Analysis

For the Main Feedwater System, and associated systems such as the condensate system, and Main Steam System, and associated systems such as the Steam Generator System, the thermal cycles anticipated correspond to Heatup and Cooldown cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 300. Therefore, main feedwater and main steam components will not experience 7,000 cycles during the period of extended operation.

The Auxiliary Feedwater System supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. The system provides an alternate to the Feedwater System during startup, hot standby and cooldown and also functions as an Engineered Safeguards System. The Auxiliary Feedwater System is directly relied upon to prevent core damage during plant transients resulting from a loss of normal feedwater flow, steam line rupture, main feedwater line rupture, loss of coolant accident (LOCA) and/or loss of off-site power by providing feedwater to the unaffected steam generators to maintain their inherent heat sink capability. The total number of cycles expected in 40 years of operation are as follows: 200 Heatup and Cooldown cycles, 2,000 cycles of feedwater cycling at hot standby, 980 cycles associated with all Upset Conditions, 240 cycles of quarterly AFW pump tests in accordance with ASME Section XI, and 40 cycles of tests per the plant Technical Specifications. This yields a total of 3,460 cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 5,190. Therefore, auxiliary feedwater components will not experience 7,000 cycles during the period of extended operation.

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

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

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

Therefore, the analysis for B31.1 piping has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

Disposition: **10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.**

4.3.3 ENVIRONMENTALLY-ASSISTED FATIGUE ANALYSIS

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

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

In addition to the above representative NUREG/CR-6260 locations, locations in the pressurizer lower head that are potentially subject to insurge/outsurge (I/O) transients were also evaluated considering reactor water environmental effects.

The methods used to evaluate environmental effects on fatigue were based on NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and NUREG/CR-6717, "Environmental Effects of Fatigue Crack Initiation in Piping and Pressure Vessel Steels." Environmental fatigue life correction factors (F_{en}) were used to obtain adjusted cumulative fatigue usage (U_{en}) which includes the effects of reactor water environments.

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

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

Where:

N	=	the number of equivalent full temperature cycles,
N_E	=	number of cycles at full temperature change for which expansion stress has been calculated
$N_1, N_2 \ldots N_n$	=	number of cycles at lesser temperature changes,
$r_1, r_2 \ldots r_n$	=	ratio of lesser temperature cycles to the cycle for which the expansion stress has been calculated.

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

4.3.4 RCS LOOP PIPING LEAK-BEFORE-BREAK ANALYSIS

Summary Description

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

Analysis

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," is a new LBB calculation applicable to HNP large bore RCS piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. This calculation concluded:

1. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation. Currently, an EPRI Material Reliability Program is underway to address the Alloy 82/182 Primary Water Stress Corrosion Cracking issue for the industry due to the V. C. Summer cracking incident. However, calculations for Alloy 82/182 locations were performed, and this material is not bounding.

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

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

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

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

The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. Therefore, the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation. When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.

Disposition: **10 CFR 54.21(c)(1)(ii) – The RCS loop LBB analysis has been projected to the end of the period of extended operation.**

4.3.5 CYCLIC LOADS THAT DO NOT RELATE TO RCS TRANSIENTS

This section contains components that are listed as involving thermal fatigue TLAAs where the number of thermal cycles that may not correspond to Class 1 components transient cycles. These components were originally designed in accordance with ASME Section III, Class 2 or Class 3 or ANSI B31.1, Power Piping Code, which requires implicit fatigue analyses using stress range reduction factors instead of explicit cumulative usage factor (CUF) values. These design codes account for cyclic loading by reducing the allowable stress for the component if the number of anticipated cycles exceeds certain limits. It requires the designer to determine the overall number of anticipated thermal cycles for the component and apply stress range reduction factors if this number exceeds 7,000. This implicit fatigue analysis method effectively reduces the allowable stress for the component, which keeps the applied loads below the endurance limit for the material.

The basic strategy in the following subsections considers the number of transient cycles postulated for 40 years. The License Renewal evaluation determines if the number of cycles for 60 years would require a reduction in stress beyond that originally applied during the original design process. These assessments can be made by comparing the design cycles projected to occur in 60 years against the 7,000 cycle criterion for a stress range reduction factor. If the total number of cycles projected for 60 years does not exceed 7,000, then the original design considerations remain valid.

4.3.5.1 Primary Sample Lines

Summary Description

In accordance with HNP FSAR Table 3.2.2-1, the portion of the Primary Sampling System equipment that are in the scope of this TLAA are the "System Piping and Valves" that are "a) Part of the RCPB" and "b) Normally or automatically isolated from the RCPB." The scope of piping for part a) is the portion of piping that is upstream of the piping anchor for the outboard isolation valves for penetrations M-78A, B, and C. These portions are essentially the safety related piping component in the system and a small portion of the non-safety related tubing up to the first anchor. The portion of piping under part b) that is downstream of the anchor on the non-safety related tubing is not considered relevant to the licensee in making safety determinations.

There are three sample line penetrations involved - RCS Hot Legs (M-78A), Pressurizer Liquid Space (M-78B), and Pressurizer Steam Space (M-78C). The following analyses determined the number of cycles to which the relevant equipment would be subjected and compared them to the implicit fatigue analysis acceptance criterion of 7,000 cycles. The applied cycles are based on the manner in which the equipment is used.

Analyses

<u>Penetration M-78A - RCS Hot Legs</u>

The piping downstream of M-78A include three parallel branch lines that supply the Post Accident Sample Panel in the Post Accident Sampling System (PASS), the Primary Sample Panel in the Reactor Coolant Sample System and the Gross Failed Fuel Detector in the Gross Failed Fuel Detection System. The Gross Failed Fuel Detector is in continuous operation during reactor startup, operation and shutdown and the base load follows the reactor thermal cycles. However, as a result of this configuration, the safety related portion of the reactor coolant sample lines has the potential to experience additional thermal cycles whenever flow through the detector is interrupted. This would occur when the containment isolation valves are closed, when flow is swapped between RCS Hot Leg 2 and Hot Leg 3, and when flow to the letdown line and Volume Control Tank and Boron Thermal Regeneration System is isolated. The cyclic operation of the Primary Sample Panel has no effect on the thermal cycles experienced by the flow through penetration M-78A due to the continuous flow through the Gross Failed Fuel Detector. Interruption of flow through the detector from downstream equipment would require isolation of the letdown line and Volume Control Tank and the Boron Thermal Regeneration System. The possibility of this latter event happening is very rare, and this occurrence is considered a negligible contributor to the number of cycles.

Based on this discussion, the total number of cycles experienced by the RCS Hot Leg Sample lines can be estimated by adding to the number of RCS thermal cycles, the number of times the hot leg is swapped and the number of cycles caused by penetration M-78A isolations of sufficient duration to permit cool-down of the sample lines. For the purposes of this evaluation, a penetration isolation lasting for more than 5 minutes while the RCS Hot Leg temperature exceeds 500°F is conservatively considered one thermal cycle as the temperature decrease was estimated to be less than 200°F.

Currently RCS flow is swapped between Hot Legs 2 and 3 on an approximate monthly schedule. Even though this results in six cycles on each supply from the hot legs, this evaluation conservatively considers this to be twelve cycles each year and simplifies the evaluation. Over 60 years of operation and ignoring shutdown this results in 720 cycles. Based on uncertainty in the early operating practice of the plant, this number has been rounded up to 1,000 cycles.

The number of cycles due to reactor shutdowns and the number of penetration M-78A isolations that would result in a thermal cycle were estimated based on a review of plant data over a period of approximately 6.75 years. During this time period there were 9 cycles due to reactor shutdowns and 30 thermal cycles due to penetration isolation valve closure. Applying a ratio of 60 years over 6.75 years yields 8.88. This value was rounded up to 9 and used to multiply 9 shutdown cycles and 30 penetration isolation cycles. This process yielded the following 60-year projections:

81 reactor thermal cycles, and
270 thermal cycles due to penetration isolations.

Therefore, the total number of hot leg thermal cycles for penetration M-78A is 1,351 cycles, which is less than the requisite 7,000 cycles.

Penetration M-78B - Pressurizer Liquid Space

The Pressurizer liquid space is sampled weekly per HNP Environmental and Chemistry Sampling and Analysis Program. This penetration supplies the primary sample panel and is cycled every time it is used to sample the Pressurizer liquid. The number of cycles was estimated by multiplying 52 cycles per year times 60 years resulting in 3120 thermal cycles. Because this line is connected to the Pressurizer, the estimated number of reactor thermal cycles over 60 years (81 cycles - see penetration M-78A) was added to 3,120 cycles. This results in 3201 thermal cycles, which is less than the requisite 7,000 cycles.

Penetration M-78C - Pressurizer Steam Space

The sample line associated with the Pressurizer steam space penetration (M-78C) is normally used during degassing for reactor shutdown conditions and also for sampling following a postulated accident. A portion of the sample line is exposed to steam from the Pressurizer. Every time condensate is removed, the line has an opportunity to heat up again. It is assumed that the condensate will be drained at least once an RCS cycle during testing of the isolation valves. Consequently, the number of thermal cycles for the sample lines will be equal to the number of RCS (Class 1 piping) cycles. The number of cycles estimated to occur is approximately 81 cycles, which is less than the requisite 7,000 cycles.

Since the total number of thermal cycles for the sample lines is less than 7,000 cycles, no reanalysis of the piping design calculations is necessary. Therefore, an evaluation was performed as required by 10 CFR 54.21(c)(1) and was successful in demonstrating under 10 CFR 54.21(c)(1) (i) that the reactor coolant sample line design analyses of record remain valid for the period of extended operation (60 years).

Disposition: **10 CFR 54.21(c)(1)(i) – The analyses of qualification for Primary Sample Lines remain valid for the period of extended operation.**

4.3.5.2 Steam Generator Blowdown Lines

Summary Description

The portion of the Steam Generator Blowdown lines included in this TLAA are listed in FSAR Table 3.2.1-1 as the portion of the system designed to ASME Section III, Class 2 and ANSI B31.1 codes. This FSAR table identifies these components as "a) the system piping and valves from the steam generator to and including outboard containment isolation valves," and "b) from containment isolation valves to RAB Wall" and "Other." Components in the turbine building may also be designed to ANSI B31.1 as noted in the "Other" listing, but these components do not have any bearing on equipment within the scope of license renewal.

Blowdown flow is normally maintained during operation in order to maintain Steam Generator water chemistry. A thermal cycle in the Blowdown lines may result whenever Blowdown flow to the flash tank is interrupted. There are many potential reasons for interruption of Blowdown flow during periods of operation. For example, Blowdown flow would be interrupted due to an Auxiliary Feedwater Pump actuation signal, a safety injection signal, high condenser hotwell level signal, Steam Generator Flash Tank Hi-Hi level, containment isolation or other testing purposes. These interruptions have the potential to result in thermal cycles over and above the heat-up and cooldown cycles of the reactor.

Analyses

The method of estimating the number of cycles is to review data over a recent time period and count the number of cycles that the Blowdown flow at HNP was interrupted. The resulting number of cycles will be multiplied by a ratio based on years to estimate the total number of cycles that could be expected over 60 years of operation. The potential to undercount comes from the assumption that the number of cycles over the period counted is representative of the past operation and will be representative of future operation. Additionally, no partial cooldown cycles are counted. To offset the potential to undercount, a conservative counting criterion is adopted and the total number of cycles will be extrapolated to 60 years.

The conservative method chosen adopted the criterion that one cycle will be counted when Blowdown flow is interrupted for more than 30 minutes. For the purposes of thermal fatigue, a complete thermal cycle is defined as a heat up from ambient to operating temperature followed by a cooled down to ambient temperature. The criterion adopted for counting thermal cycles is conservative because it includes interruptions of Blowdown flow in which a significant decrease in temperature is <u>not</u> expected. This is based on the operating practice for reestablishing Blowdown flow following a Blowdown isolation valve closure. This operating practice states that if the isolation valves are closed for more than 30 minutes the downstream piping must be warmed-up prior to opening the isolation valves. It follows then that if an isolation valve is closed for less than 30 minutes this does not constitute a significant cooldown period.

The number of cycles due to reactor shutdowns is included in the Blowdown cycles counted. Based on a review of plant data over a period of approximately 5.5 years, the number of Blowdown flow interruptions that would result in a thermal cycle were estimated to be 37 cycles. Applying a ratio based on 60 years results in 404 cycles. Since the total number of thermal cycles for the Steam Generator Blowdown lines is less than 7,000 cycles, no reanalysis of the piping design calculations is necessary. Therefore, an evaluation was performed as required by 10 CFR 54.21(c)(1) and was successful in demonstrating under 10 CFR 54.21(c)(1) (i) that the Steam Generator Blowdown Line design analyses of record remain valid for the period of extended operation (60 years).

Disposition: **10 CFR 54.21(c)(1)(i) – The analyses of qualification for the Steam Generator Blowdown Lines in the scope of License Renewal remain valid for the period of extended operation.**

TABLE 4.3-1 NSSS TRANSIENT CYCLE HISTORY

Transient Classification	No.	Transient Name	Design Cycles	Cycles To-Date (18 Years)
Normal Condition				
	1	Plant Heatup	200	40
	2	Plant Cooldown	200	40
	3	Unit Loading at 5%/Min from 15% to 100% Power	18,300	535
	4	Unit Unloading at 5%/Min from 100% to 15% Power	18,300	444
	5	Step Load Increase of 10% of Full Power	2,000	Note 1
	6	Step Load Decrease of 10% of Full Power	2,000	13
	7	Large Step Load Decrease	200	6
	8	Steady State Fluctuations (Initial)	1.50E+05	Not Counted Note 2
	9	Steady State Fluctuations (Random)	3.00E+06	Not Counted Note 2
	10	Feedwater Cycling at Hot Standby	2,000	100
	11	Unit Loading from 0% to 15% Power	500	Note 1
	12	Unit Loading from 15% to 0% Power	500	Note 1
	13	Loop Out of Service Shutdown	80	Not Applicable Note 5
	14	Loop Out of Service Startup	70	Not Applicable Note 5
	15	Boron Concentration Equalization	26,400	Not Counted Note 2
	16	Turbine Roll Test	80	1 Note 3
	17	Refueling	80	24
Upset Condition				
	1	Loss of Load	200	0 Note 4
	2	Loss of Power	40	1
	3	Partial Loss of Flow	80	2
	4	Reactor Trip Case A No Cooldown	230	44
	5	Reactor Trip Case B Cooldown, No SI	160	7
	6	Reactor Trip Case C Cooldown with SI	10	2

TABLE 4.3-1 NSSS TRANSIENT CYCLE HISTORY

Transient Classification	No.	Transient Name	Design Cycles	Cycles To-Date (18 Years)
	7	Inadvertent RCS Depressurization	20	1
	8	Inadvertent Startup of an Inactive Loop	10	Not Applicable Note 5
	9	Control Rod Drop	80	0
	10	Inadvertent Safety Injection	60	2
	11	Excess Feedwater Flow	30	1
	12	RCS Cold Over-pressurization	10	5
	13	Operating Basis Earthquake (OBE)	5	0
Emergency Condition				
	1	Small LOCA	5	0
	2	Small Steam Line Break	5	0
	3	Complete Loss of Flow	5	0
Faulted Condition				
	1	Large LOCA	1	0
	2	Large Steam Line Break	1	0
	3	Feedwater Line Break	1	0
	4	Reactor Coolant Pump Locked Rotor	1	0
	5	Control Rod Ejection	1	0
	6	Steam Generator Tube Rupture	6	0
	7	Safe Shutdown Earthquake (SSE)	1	0
Test Condition				
	1	Primary Side Hydrostatic Test	10	7 Note 6
	2	Secondary Side Hydrostatic Test	10	3
	3	Primary Side Leak Test	200	1
	4	Secondary Side Leak Test	80	0
	5	Steam Generator Tube Leak Test	800	0

Notes:

1. Normal Transients 5, 11, and 12, "Step Load Increase of 10% of Full Power," "Unit Loading from 0% to 15% Power," and "Unit Unloading from 15% to 0% Power." Cycles to-date are zero, based on the analysis of a 5.26-year sample of Plant Instrumentation data. Normal Transients 11 and 12 exist to address low power feedwater operations. Feedwater operations during the 0% to 15% power range are assumed to transition from auxiliary feedwater to main feedwater. Additionally, main feedwater is assumed to start at 32°F during the power increase and cool to 32°F during the power decrease. At HNP, feedwater operations at low power conditions that could potentially affect the primary systems and equipment are being tracked through HNP's cycle counting program. This tracking provides a much better understanding of the transients than simply monitoring the numbers of occurrences of the 0% to 15% transients. The HNP cycle counting program is used to track the following related events:

- Feedwater Cycling at Hot Standby, 2000 cycles with an alarm limit of 1500.
- Main Feedwater Nozzle Temperature - Plant Loading Between 0 & 15% Power With Feedwater < 100°F, 60 cycles with an alarm limit of 42.
- Main Feedwater Nozzle Temperature - Plant Loading Between 0 & 15%, 180 cycles with an alarm limit of 126.
- Auxiliary Feedwater Nozzle Temperature and Flow Cycle, 2000 cycles with an alarm limit of 1500.

Progress Energy operates HNP as a base-load generator (i.e., HNP is not a "load following" plant). Normal Transient 5, while not precluded, the accumulation rate is considered very low in comparison to the allowed design cycles.

2. Normal Transients 8, 9, and 15, "Steady State Fluctuations (Initial)," "Steady State Fluctuations (Random)," and "Boron Concentration Equalization," are not counted. Prior to plant operation and the establishment of plant cycle counting procedure, it was concluded that the design limits would never be reached, based on the expected number of cycles. The temperature changes associated with Steady State Fluctuations (Initial)" and "Steady State Fluctuations (Random)" are small (≤ 3°F). Therefore, these transients are not contributors to CUF. "Boron Concentration Equalization" transients are associated with the operation of a "load following" plant. Progress Energy operates HNP as a baseload generator (i.e., HNP is not a "load following" plant). Normal Transient 15, while not precluded, the accumulation rate is considered very low in comparison to the allowed design cycles. The "Cycles To-Date" for these entries should be changed to "Not Counted."
3. One Turbine Roll Test in 1986 during initial construction, and none since that time.
4. Upset Transient 1, "Loss of Load" has been counted since the start of plant operations in accordance with the plant cycle counting procedure. The number of occurrences has been zero.
5. Normal Transients 13, 14, and Upset Transient 8 were included in the qualifications performed by WCAP-14778, Revision 1, "Carolina Power and Light Harris Nuclear Plant Steam Generator Replacement/ Uprating Analysis and Licensing Project NSSS Engineering Report," September 2000. "Normal Condition" transients 13 and 14 ("Loop Out of Service") are not applicable to the current HNP license. HNP is not currently licensed to operate with N-1 loops. The "Loop Out of Service" transients were included in the Westinghouse System Standard Design Criteria (SSDC 1.3, Rev. 2) so that the components are designed in case the plant is licensed to operate with N-1 loops. It was recommended by Westinghouse. that the "Loop Out of Service" transients continue to be considered for the SGR/Uprating Project. This also applies to "Upset" Transient 8 ("Inadvertent Startup of an Inactive Loop").
6. FSAR 3.9.1.1.5 states that primary side hydrostatic tests include both shop and field hydrostatic tests. These tests can occur as a result of component or system testing. FSAR 3.9.1.1.5 also states that four additional hydrostatic tests, in accordance with ASME Section XI inservice inspection (ISI) requirements, are expected over the lifetime of the plant. Since four additional tests in accordance with ASME Section XI ISI requirements were expected to be performed, and since one of these tests was performed and recorded, then no more than six shop tests were possible. Therefore, the total number of primary side hydrostatic can be no more than seven.

TABLE 4.3-2 DESIGN FATIGUE USAGE FACTORS

List Number	Component	Location	Fatigue Usage
1	Reactor Vessel	Outlet Nozzles	0.0458
2		Inlet Nozzles	0.0061
3		Closure Head Flange	0.0037
4		Vessel Flange	0.0041
5		Closure Studs	0.3744
6		CRDM Housings	0.0021
7		Vent Pipe	0.0062
8		Bottom Head Juncture	0.0129
9		Bottom Head Instrumentation Tubes	0.0062
10		Main Vessel Shell	0.0052
11		Core Support Pads	0.0221
12		Head Adapter Plug	0.0042
13	Reactor Vessel Internals	Core Internals	0.52
14	CRDMs	Generic Report, Lower Joint Canopy Area	0.4392
15		Lower Joint Canopy Area, ASN 23 Outside Surface	0.9862
16		Lower Joint Head Adapter, ASN 8 Inside Surface	0.3115
17		Lower Joint Latch Housing, ASN 14 Inside	0.1009
18		Middle Joint Latch Housing, ASN 43 Outside Surface	0.0584
19		Middle Joint Rod Travel Housing, ASN 53 Inside Surface	0.0007
20		Middle Joint Canopy Area, ASN 69 Inside Surface	0.1508
21		Upper Joint Rod Travel Housing, ASN 95 Inside Surface	0.0696
22		Upper Joint CRDM Cap, ASN 131 Outside Surface	0.0053
23		Upper Joint Vent Plug, ASN 142 Inside Surface	0.001
24		Upper Joint Canopy Area, ASN 118 Inside Surface	0.0001
25		CLH Latch Housing, ASN 47 Outside Surface	0.1184
26		CLH Cap, ASN 152 Inside Surface	0.7241
27		CLH Vent Plug, ASN 142 Inside Surface	0.0001
28		CLH Canopy Area, ASN 70 Inside Surface	0.1294
29	RCP	RCP	Fatigue waiver
30	Steam Generator	Primary Side Components	0.62
31		Main Feedwater Nozzle	0.98
32		Auxiliary Feedwater Nozzle	0.16
33		Secondary Manway Opening	0.59

TABLE 4.3-2 DESIGN FATIGUE USAGE FACTORS

List Number	Component	Location	Fatigue Usage
34		Secondary Manway Bolts	1.462 Note 1
35		Hand-Hole	0.525
36		Hand-Hole Bolts	0.804
37		4" Inspection Port	0.728
38		4" Inspection Port Bolts	1.233 Note 1
39		2" Un-Reinforced Inspection Port Opening	0.495
40		2" Un-Reinforced Inspection Port Opening Bolts	0.747
41		Minor Shell Taps	0.9844
42		Tubesheet/Stub Barrel Junction	0.34
43		Upper Shell/Transition Cone/Lower Shell	0.576
44		Spray Nozzles	0.95
45	Pressurizer	Surge Nozzle	0.672
46		Spray Nozzle Safe End	0.42
47		Spray Nozzle Body/Shell intersection	0.85
48		Safety and Relief Nozzle	0.064
49		Lower Head	0.909
50		Heater Well	0.198
51		Upper Shell	0.992
52		Support Skirt/Flange	0.393
53		Support Skirt/Lower Shell	0.143
54		Seismic Lug at lug/shell junction	0.37
55		Manway Cover Bolts	0.875
56		Trunnion Bolt Hole	0.995
57		Instrument Nozzle	0.338
58		Immersion Heater/Sheath	0
59		Valve Support Bracket	0.24
60	RCL Piping	Hot Leg	0.9
61		Crossover Leg	0.2
62		Cold Leg	0.4
63	RHR Loop 1 Piping	12" x 8" Trunnion	0.921
64		RCL nozzle	0.1
65	RHR Loop 3 Piping	12" x 1" Branch	0.173
66		RCL nozzle	0.1
67	Pressurizer Surge Line	RCL nozzle	0.85
68	Accumulator Loop 1 Piping	12" Long radius elbow	0.09
69		RCL nozzle	0.45

TABLE 4.3-2 DESIGN FATIGUE USAGE FACTORS

List Number	Component	Location	Fatigue Usage
70	Accumulator Loop 2 Piping	12" Long radius elbow	0.09
71		RCL nozzle	0.45
72	Accumulator Loop 3 Piping	12" Long radius elbow	0.09
73		RCL nozzle	0.45
74	Pressurizer Spray Piping	Lp. 1,2 Cold leg 4" nozzle	0.4
75		Section 5 4" butt weld	0.983
76	Normal Charging Line Piping	Lp. 2 Cold leg 3" nozzle	0.79
77		Section 2 branch	0.92
78	Alternate Charging Line Piping	Lp. 1 Cold leg 3" nozzle	0.85
79		Sec 2 branch	0.92
80	Safety Injection - Cold Leg Lines Loops 1,2,3	Cold leg 6" nozzle	0.7
81		Section 3 2" Socket weld	0.9
82	Safety Injection - Hot Leg Lines Loops 1,2,3	Hot leg 6" nozzle	0.7
83	Safety Injection - Hot Leg Lines Loops 1,2,3	Section 2 6"x2" branch	0.61
84	Normal Letdown Line	Loop 1 Crossover leg 3" nozzle	0.2
85		Section 1 3"x2" reducer	0.53
86	Excess Letdown Line	Loop 3 Crossover leg 3" nozzle	0.4
87		Section 4 2" elbow	0.05
88	Loop Drain Line	Loop 2 Crossover leg 2" nozzle	0.21
89		Section 3 Socket weld to valve	0.82
90	Pressurizer Safety Line	Section 2 6"x3/4" Branch	0.9
91	Pressurizer Relief Line	Section 2 3" Transition weld to valve	0.9

Note 1. Due to the original design usage factors exceeding 1.0, these bolts were originally to be replaced based on a replacement schedule. However, these fatigue usage values have been superseded by the results of the license renewal fatigue evaluation described in Subsection 4.3.1.5 and a replacement schedule is no longer required.

TABLE 4.3-3 60-YEAR ENVIRONMENTALLY-ADJUSTED CUF VALUES

Component	60-year Environmentally Adjusted CUF (U_{en})	Fen	A	B	C
Bottom Head Juncture	0.0491	2.532			
Reactor Vessel Inlet Nozzles	0.0231	2.532			
Reactor Vessel Outlet Nozzles	0.1740	2.532			
Surge Line	2.120 (Note 3)	8.27 (Note 1)	X	X	X
Charging Nozzle	0.89	7.2 (Note 1)	X		X
Safety Injection Nozzle	0.93	5.3 (Note 1)	X		
RHR Class 1 Piping	0.465	2.55 (Note 2)			
Pressurizer (Lower Head at Heater Penetration)	1.35 (Note 4)	2.532	X	X	X

A. Reduced cycles used in the evaluation
B. Refined calculations performed
C. Redefined transients used in the evaluation

Note 1. The "overall effective Fen" based on evaluation of the fatigue transient pairings. Each transient pair has its own unique Fen.

Note 2. The transients used for the RHR line qualification include only one significant transient defined for "RHR section 2" (RHR section 2 is the section of piping downstream of the isolation valve that is normally at ambient conditions), for RHR initiation when this part of the line goes from ambient conditions to the 350°F RHR letdown temperature. The return phase of the transient is a gradual cooldown with which no appreciable stress is associated. Since the temperature shock for the RHR initiation transient is positive, the stresses on the inside surface of the piping components are compressive. Since the strain rate is compressive, Fen = 1.0 for this controlling condition would be appropriate, based upon the methodology of NUREG/CR-5704. However, a maximum value of 2.55 has been used for the Fen for this evaluation.

Note 3. 40-year design transients were used in the evaluation except for heatups and cooldowns. The number of heatup and cooldown transients used in the analysis is 133 versus 200 original design transients. The fatigue usage for this location based on this transient set is 0.94. Multiplying by 200/133 to account for the full set of design heatups and cooldowns yielded a 40-year fatigue usage of 1.414. Multiplying by 1.5 (that is 60 years/40 years) yields a 60-year fatigue usage of 2.120.

Note 4. 40-year design transients were used in the evaluation except for heatups and cooldowns. The number of heatup and cooldown transients used in the analysis is 133 versus 200 original design transients. The fatigue usage for this location based on this transient set is 0.5984. Multiplying by 200/133 to account for the full set of design heatups and cooldowns yielded a 40-year fatigue usage of 0.9. Multiplying by 1.5 (that is 60 years/40 years) yields a 60-year fatigue usage of 1.35.

Enclosure 2, Attachment 2

This new Subsection A.1.2.2 replaces the Subsection A.1.2.2, Metal Fatigue, in the original LRA.

A.1.2.2 Metal Fatigue

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

The first step in the evaluation was to establish the current design basis for the major NSSS components. This was done by reviewing the current design transients and past operational transients. This review showed that the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections A.1.2.2.6 and A.1.2.2.7). Therefore the governing transients "NSSS Design Transients" are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design Cumulative Usage Factor (CUF) values were compiled from design documents including the steam generator replacement/uprating analysis.

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

A.1.2.2.1 Reactor Vessel Fatigue Analyses

TLAAs have been identified for several sub-components of the Reactor Vessel. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see Subsections A.1.2.2.6 and A.1.2.2.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the HNP steam generator replacement/uprating analysis. The reactor vessel fatigue analysis demonstrated that if the reactor vessel components were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0. For the components that comprise the Reactor Vessel, the highest 40-year design fatigue usage value is 0.37 for the closure studs. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.56. This value does not exceed the design limit of 1.0 and is therefore acceptable. This 60-year fatigue usage bounds the maximum environmentally adjusted usage factor of 0.1740 for the Reactor Vessel Outlet Nozzles. Therefore, the Reactor Vessel Fatigue Analyses have been projected to the end of the period of extended operation.

A.1.2.2.2 Reactor Vessel Internals Fatigue Analyses

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

A.1.2.2.3 Control Rod Drive Mechanism Fatigue Analysis

TLAAs have been identified for several sub-components of the Control Rod Drive Mechanism (CRDM). NSSS Design Transients were identified and forty-year design CUF values determined as part of the HNP steam generator replacement/uprating analysis. For the CRDM, the highest 40-year design fatigue usage value is 0.99 for the "Lower Joint Canopy Area." Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF

of 1.49. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the CRDM will be managed for the period of extended operation.

A.1.2.2.4 Reactor Coolant Pumps Fatigue Analysis

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

A.1.2.2.5 Steam Generators Fatigue Analysis

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

Other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts, the highest 40-year design fatigue usage value is 0.98 for minor shell taps. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47. This value exceeds the design limit of 1.0; and, therefore, an aging management program is required. The HNP RCPB Fatigue Monitoring Program will ensure that the fatigue usage design limit is not exceeded or that appropriate reevaluation or corrective action is taken. Therefore, the effects of fatigue on the steam generator (other than the Secondary Manway Bolts and the 4 in. Inspection Port Bolts) will be managed for the period of extended operation. The Steam Generator Secondary Manway Bolts and 4 in. Inspection Port Bolts were identified as having 40-year design fatigue usage factors over 1.0. These components were characterized as "to be replaced based on a replacement schedule." However, HNP reanalyzed the Steam Generator Secondary Manway Cover Bolts and 4 in. Inspection Port Bolts to remove unnecessary conservatisms. The updated evaluation changed only the number of Unit Loading and Unit Unloading transient cycles relative to the previous design analysis. Each transient was considered to occur 2000 times over the life of the plant, a number which is still greater than the best estimate number provided in the previous design analysis. Fatigue usage values for the bolts based on the reduced Unit Loading and Unit Unloading cycles are as follows:

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

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

A.1.2.2.6 Pressurizer Fatigue Analysis

TLAAs have been identified for several sub-components of the Pressurizer. As stated above, the use of transients provided by the HNP steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exception of the pressurizer surge line and portions of the pressurizer lower head which were analyzed separately (see below and Subsection A.1.2.2.7). Therefore the NSSS Design Transients are those identified in the HNP steam generator replacement/uprating analysis. Forty-year design CUF values were also determined as part of the steam generator replacement/uprating analysis. The Pressurizer fatigue analysis demonstrated that if the Pressurizer subcomponents were exposed to a bounding set of postulated transient cycles, the CUF values for the components would not exceed 1.0 for all components. However, certain locations of the Pressurizer lower head are not bounded by the original design fatigue analysis, because the original fatigue analysis did not consider insurge/outsurge transients that were identified subsequent to the original fatigue analysis.

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

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

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

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

A.1.2.2.7 Reactor Coolant Pressure Boundary Piping (ASME Class 1) Fatigue Analysis

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

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

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

A.1.2.2.8 ASME Class 2 and 3 Piping Fatigue Analysis

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

A.1.2.2.9 ANSI B31.1 Piping Fatigue Analysis

Summary Description

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

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

The diesel generators in the Emergency Diesel Generator System undergo monthly surveillance tests in accordance with plant Technical Specifications. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 720. Therefore, the emergency diesel

generator diesel exhaust piping will experience significantly fewer than 7,000 equivalent full temperature cycles during the period of extended operation.

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

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

Therefore, the analysis for B31.1 piping has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

A.1.2.2.10 Environmentally-Assisted Fatigue Analysis

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

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

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

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

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

Where:

N	=	the number of equivalent full temperature cycles,
N_E	=	number of cycles at full temperature change for which expansion stress has been calculated
$N_1, N_2 \ldots N_n$	=	number of cycles at lesser temperature changes,
$r_1, r_2 \ldots r_n$	=	ratio of lesser temperature cycles to the cycle for which the expansion stress has been calculated.

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

A.1.2.2.11 RCS Loop Piping Leak-Before-Break Analysis

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

WCAP-14549-P, Addendum 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Harris Nuclear Plant for the License Renewal Program," is a new LBB calculation applicable to HNP large bore RCS piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. This calculation concluded:

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

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

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

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

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

The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. Therefore, the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation. When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.

A.1.2.2.12 Primary Sample Lines Fatigue Analysis

Portions of the Primary Sampling System may be subjected to thermal transients that are unrelated to those experienced by Class 1 components. There are three sample line penetrations involved - RCS Hot Leg (M-78A),

Pressurizer Liquid Space (M-78B), and Pressurizer Steam Space (M-78C). The following analyses determined the number of cycles to which the relevant equipment would be subjected and compared them to the implicit fatigue analysis acceptance criterion of 7,000 cycles. The applied cycles are based on the manner in which they are used.

Penetration M-78A - RCS Hot Legs

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

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

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

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

Penetration M-78B - Pressurizer Liquid Space

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

Penetration M-78C - Pressurizer Steam Space

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

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

A.1.2.2.13 Steam Generator Blowdown Lines Fatigue Analysis

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

Blowdown interruptions were determined using actual plant data and projecting the number of interruptions through the period of extended operation. A conservative method was chosen such that one cycle was counted when blowdown flow was interrupted for more than 30 minutes. For the purposes of thermal fatigue, a complete thermal cycle is defined as a heat up from ambient to operating temperature followed by a cooldown to ambient temperature. The criterion adopted for counting thermal cycles is conservative because it includes interruptions of blowdown

flow in which a significant decrease in temperature is not expected. This is based on the operating practice for reestablishing blowdown flow following a isolation which requires the downstream piping to be warmed-up prior to opening the isolation valves if the isolation valves were closed for more than 30 minutes. It follows then that if an isolation valve is closed for less than 30 minutes this does not constitute a significant cooldown period.

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

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: B.3.1-RH-01

NRC Request:
Please provide responses to the following questions related to the Fatigue Monitoring Program described in Section B.3.1 of the LRA:

- What components are included in the stress-based fatigue monitoring portion of the HNP program?
- Please explain the adequacy of the alarm limits for those components in the stress-based fatigue monitoring portion of the HNP program.
- As stated in the LRA, the program uses online fatigue monitoring software for the periodic updating of the CUF calculation for the high CUF RCPB (including auxiliary system) components. Please provide information on the timing of these periodic updates and whether or not the CUF update is for all components under the FMP or only those components with stress-based fatigue monitoring.
- Please define the alarm limit for those components under the cycle-counting portion of the FMP.

HNP Response:
(First Bullet) What components are included in the stress-based fatigue monitoring portion of the HNP program?

The HNP Fatigue Evaluation for License Renewal (WCAP-16353-P) resulted in the following locations recommended for inclusion into the program.

- Pressurizer Lower Head
- Pressurizer Surge Line
- CVCS Piping and Heat Exchanger

Based on the Westinghouse recommendations, the HNP fatigue monitoring program will be enhanced to include the above components by monitoring fatigue usage for these locations using online fatigue monitoring software.

Harris Nuclear Plant License Renewal Audit Question and Response Database

(Second Bullet) Please explain the adequacy of the alarm limits for those components in the stress-based fatigue monitoring portion of the HNP program.

The alarm limit for the stress-based fatigue monitoring portion of the HNP program will be set at 0.9. The table below lists the locations to be monitored:

Item No.	Description	Comp ID	ASN ID	Node#	WESTEMS Calculated Baseline fatigue usage (Note 1)	Fen (Note 2)	WESTEMS Calculated Baseline fatigue usage with environmental effects (Note 3)
1	Lower Head to Shell Weld	151201	1	1	0.0189	2.532	0.0479
2	Lower Head to Shell Weld	151201	1	7	0.0006	1.000	0.0006
3	SS Lower Head at Heater Penetration	151202	1	1	0.0211	15.348	0.3238
4	Lower Head at Heater Penetration	151202	2	1	0.1701	2.532	0.4307
5	Lower Head at Heater Penetration	151202	2	7	0.0521	1.000	0.0521
6	Lower Head to Nozzle Weld	151203	3	1	0.0208	2.532	0.0527
7	Lower Head to Nozzle Weld	151203	3	7	0.0059	1.000	0.0059
8	Surge Nozzle Knuckle	151501	4	1	0.0313	2.532	0.0793

Harris Nuclear Plant License Renewal Audit Question and Response Database

Item No.	Description	Comp ID	ASN ID	Node#	WESTEMS Calculated Baseline fatigue usage (Note 1)	Fen (Note 2)	WESTEMS Calculated Baseline fatigue usage with environmental effects (Note 3)
9	Surge Nozzle Knuckle	151501	4	7	0.0465	1.000	0.0465
10	Surge Nozzle to Safe End Weld	151502	5	1	0.0097	2.532	0.0246
11	Surge Nozzle to Safe End Weld	151502	5	11	0.0015	1.000	0.0015
12	Surge Nozzle Safe End to Pipe Weld	151503	6	1	0.0285	15.348	0.4377
13	Surge Nozzle Safe End to Pipe Weld	151503	6	11	0.0232	1.000	0.0232
14	PSL RCS surge nozzle Loc 19	152619	0	0	0.0174	8.270	0.1439
15	PSL RCS surge nozzle Loc 22	152622	0	0	0.0345	8.270	0.2853
16	PSL RCS surge nozzle Loc 25	152625	0	0	0.1087	8.270	0.8989

Notes

1 The WESTEMS calculated baseline date is 10/18/2004.
2 For locations not exposed to primary system coolant, the Fen factor is 1.0.
3 WESTEMS Calculated Baseline with environmental effects = Fen * WESTEMS Calculated Baseline.
4 Years of operation at WESTEMS calculated baseline = 10/18/2004 - 10/24/1986 = 18 years.

The current fatigue usage for 18 years of operation is (as of 10/18/2004) ≤ 0.4377 for all locations except Item No. 16. The maximum fatigue usage accumulation rate for these locations is 0.0243/year (0.4377/18 years). The margin between

the alarm limit of 0.9 and the design limit of 1.0 is 0.1. Dividing the margin by the maximum fatigue usage accumulation rate (0.1/0.0243) yields a margin of more than 4 years between the time a component reaches a fatigue usage of 0.9 to reach the design limit of 1.0.

For the PSL RCS surge nozzle Loc 25 (pressurizer surge line reactor coolant system surge nozzle, location 25 - Nozzle safe-end to pipe weld, inside surface), the usage to date including environmental effects is 0.8989. However, the bulk of this fatigue usage (0.887 out of 0.8989) is related to plant operation prior to the use of Modified Operating Procedures as recommended by the Westinghouse Owner's Group (the start of MOP operations began after 12/20/1993). The fatigue usage at this location is driven primarily by plant heatups and cooldowns. The incremental fatigue usage including environmental effects at this location when operating under the Modified Operating Procedures is 0.0023/(Heatup/Cooldown). Dividing the margin by the incremental fatigue usage accumulation rate (0.1/0.0023) yields a margin of more than 40 Heatup/Cooldown cycles between the time this component reaches a fatigue usage of 0.9 to reach the design limit of 1.0.

(Third Bullet) As stated in the LRA, the program uses online fatigue monitoring software for the periodic updating of the CUF calculation for the high CUF RCPB (including auxiliary system) components. Please provide information on the timing of these periodic updates and whether or not the CUF update is for all components under the FMP or only those components with stress-based fatigue monitoring.

The current HNP Fatigue Monitoring Program requires a monthly evaluation of cyclic and transient data. When enhanced, the program will require a periodic update (at least once every 18 months) and review of monitored usage values in addition to the monthly cyclic and transient data monitored. This update will include an update to the accumulated fatigue usage for the 16 components listed in the response to Bullet 2.

(Fourth Bullet) Please define the alarm limit for those components under the cycle-counting portion of the FMP.

The cycles and transients currently tracked and associated alarm limits are shown in the following table:

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CYCLE OR TRANSIENT	ALARM LIMIT	CYCLIC OR TRANSIENT LIMIT
RCS HEATUP CYCLE (TOTAL = COMPLETE + PARTIAL)	150	200
RCS COOLDOWN CYCLE (TOTAL = COMPLETE + PARTIAL)	150	200
PRESSURIZER COOLDOWN CYCLE	150	200
LOSS OF LOAD CYCLE	150	200
LOSS OF OFF-SITE POWER	30	40
LOSS OF FLOW IN ONE REACTOR COOLANT LOOP	60	80
REACTOR TRIP	300	400
REACTOR TRIP/NO COOLDOWN (Temp > 536.5°F)	172	230
REACTOR TRIP/COOLDOWN NO SI	120	160
REACTOR TRIP/COOLDOWN WITH SI	7	10
RCS DEPRESSURIZATION TRANSIENT	15	20
AUXILIARY SPRAY TRANSIENT	7	10
FEEDWATER CYCLING AT HOT STANDBY	1500	2000
MAIN FEEDWATER NOZZLE TEMPERATURE - PLANT LOADING BETWEEN 0 & 15% POWER WITH FEEDWATER < 100°F	42	60
MAIN FEEDWATER NOZZLE TEMPERATURE - PLANT LOADING BETWEEN 0 & 15%	126	180
AUXILIARY FEEDWATER NOZZLE TEMPERATURE AND FLOW CYCLE	1500	2000
RCS LEAK TEST	150	200
RCS HYDROSTATIC PRESSURE TEST	7	10
STEAM LINE BREAK	N/A	1
STEAM GENERATOR TUBE RUPTURE	4	6
SECONDARY SIDE HYDROSTATIC PRESSURE TEST	7	10
PRESSURIZER THERMAL TRANSIENT (HEATUP)	12	18
PRESSURIZER THERMAL TRANSIENT (COOLDOWN)	12	18
OPERATING BASIS EARTHQUAKE (5 EARTHQUAKES, 10 CYCLES EACH)	30	50
EXCESSIVE FEEDWATER FLOW TRANSIENT	27	30
CONTROL ROD DROP	60	80

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CYCLE OR TRANSIENT	ALARM LIMIT	CYCLIC OR TRANSIENT LIMIT
RCS COLD OVERPRESSURIZATION	7	10
SMALL LOSS-OF-COOLANT	3	5
SMALL STEAM LINE BREAK	3	5
COMPLETE LOSS OF FLOW	3	5
SECONDARY SIDE LEAKAGE TEST	60	80
TUBE LEAKAGE TEST	See Definition Below	
REFUELING	60	80

HNP re-analyzed the Steam Generator Secondary Manway Cover Bolts and 4 in. Inspection Port Bolts to remove unnecessary conservatisms. The updated evaluation changed only the number of Unit Loading and Unit Unloading transient cycles relative to the previous design analysis. Each transient was considered to occur 2000 times over the life of the plant. Therefore, the HNP Fatigue Management Program will be enhanced to track these cycles with a limit of 2000 cycles and an alarm limit of 1500 cycles.

The cycles and transients are defined as follows:

RCS Heatup/Cooldown Cycle - The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F/hr. (These operations can take place at lower rates approaching the minimum of 0°F/hr. The expected normal rates are 50°F/hr.).

Pressurizer Cooldown Cycle - Pressurizer cooldown from 650°F or greater to less than or equal to 200°F at a rate of less than or equal to 200°F in any one hour period.

Loss of Load Cycle - Decrease in load from greater than or equal to 15% rated thermal power to 0% rated thermal power, due to a step load decrease in turbine power, without immediate turbine or reactor trip.

Loss of Off-Site Power - Loss of off-site AC electrical power to Emergency Buses 1A-SA and 1B-SB.

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Loss of Flow in One Reactor Coolant Loop - Loss of only one reactor coolant pump with reactor at greater than 15% power.

Reactor Trip - Any time the reactor trip breakers open causing one or more full length control rods to be inserted from greater than 15% power to 0% power.

RCS Depressurization Transient - A range of upset conditions such as failing open of a pressurizer safety valve, pressurizer spray valve or aux spray valve resulting in a rapid depressurization of the primary system. Also considered in this transient is the impact of the pressurizer spray nozzle from the failing open of the aux spray valve and the potential for thermal shock to the spray nozzle.

Auxiliary Spray Transient - Pressurizer spray water temperature differential greater than 320°F, but less than 625°F.

Feedwater Cycling at Hot Standby - These transients can occur when the plant is at "no load conditions, during which intermittent (slug) feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases at a rate of approximately 2°F/min for a total decrease of 24°F. The temperature immediately begins to return to normal no-load temperature at a heatup rate of approximately 0.7°F/min. One cycle will be counted for each time auxiliary feedwater or main feedwater is initiated to the steam generator with RCS temperature above 250°F when there has been an interruption of flow for greater than 1 minute.

Main Feedwater Nozzle Temperature - Plant loading between 0 and 15% power with feedwater temperature at 40°F - Any time during shutdown that the water in feedwater piping reaches a temperature less than 100°F. This is expected to occur rarely in the event of an extended outage where cold ambient temperature results in feedwater temperatures of less than 100°F in portions of the feedwater system piping. This transient may subject the feedwater nozzles/feedring to increased stresses during the initial plant heatup, startup, or normal condition transient, unit loading between 0 and 15% power.

Auxiliary Feedwater Nozzle Temperature and Flow Cycle - AFW flow or feedwater flow that is less than 250°F when the average steam generator temperature is greater than 250°F.

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RCS Leak Test - RCS pressurized to greater than or equal to 2485 psig.

RCS Hydrostatic Pressure Test - RCS pressurized to greater than or equal to 3107 psig.

Steam Line Break - Break in greater than 6-inch steam line.

Steam Generator Tube Rupture

Secondary Side Hydrostatic Pressure Test - Secondary side of steam generator pressurized to greater than or equal to 1481 psig.

Pressurizer Thermal Transient - A pressurizer heatup or cooldown in excess of Tech Spec Limits. The eighteen cooldown and eighteen heatup transients are specified as follows:

- Ten cooldown or heatup transients involving RCS/pressurizer fluid in/out surges with the difference in temperature between RCS and pressurizer greater than 130°F
- Eight cooldown or heatup transients involving RCS/pressurizer fluid in/out surges with the difference in temperature between RCS and pressurizer less than 130°F.

Operating Basis Earthquake - Any earthquake in excess of the Operating Basis Earthquake. The allowed number of earthquakes analyzed is 5. Each earthquake is assumed to contain 10 cycles for a total of 50 cycles. The mechanical stresses resulting from the operating basis earthquake are considered on a component basis. Fatigue analyses, where required by the codes, are performed by the supplier as part of the stress report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis and so are tracked by the program.

Excessive Feedwater Flow Transient - Inadvertent opening of a feedwater control valve while the plant in hot standby or at no load condition. The feedwater control valves close on a Lo Tavg signal, and a safety injection occurs.

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Control Rod Drop - This transient occurs if a bank of control rods (worth 1% reactivity) drops into the fully inserted position due to a single component failure. The reactor is tripped on either low pressurizer pressure or negative flux rate, depending on time in core life and magnitude of the reactivity insertion.

RCS Cold Overpressurization - RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without the existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. This cycle is received any time that LTOPs actuates or should have actuated. The event is inadvertent, and can potentially occur by any one of a variety of malfunctions or operator errors. All events which have occurred to date may be categorized as belonging to either events resulting in the addition of mass (mass input transient) or events resulting in the addition of heat (heat input transients). All of these possible transients are represented by composite "umbrella" design transients, referred to here as RCS cold overpressurization.

Small Loss-Of-Coolant - For design transient purposes the small loss-of-coolant accident is defined as a break equivalent to the severance of a 1-inch ID branch connection. (Breaks smaller than 0.375-inch ID can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much larger than 1-inch will cause accumulator injection soon after the accident and are regarded as Large Loss-of-Coolants. It should be assumed that the Safety Injection System is actuated immediately after the break occurs and delivers water at a minimum temperature of 32°F to the RCS.

Small Steam Line Break - For design transient purposes, a small steam break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. The following conservative assumptions are made:

a. *The reactor is initially in a hot, zero-power condition.*

b. *The small steam break results in immediate reactor trip and S.I. actuation.*

c. *A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.*

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d. *The safety injection system operates at design capacity and repressurizes the reactor coolant system within a relatively short time.*

Complete Loss of Flow - This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on under-voltage followed by automatic opening of the steam dump system.

Secondary Side Leakage Test - The secondary side is pressurized to just below its design pressure to prevent safety valves from lifting with the secondary side temperature between 120°F and 250°F. In addition the primary side is also pressurized to prevent exceeding 670 psid.

Tube Leakage Test - The secondary side of the steam generator is pressurized with water and the under side of the tube sheet is inspected for leaks. The number of cycles is dependent on the test pressure as follows:

Steam Generator Shell Pressure (PSIG)	Alarm	Cycles
200	300	400
400	150	200
600	90	120
840	60	80

Refueling - A the beginning of the refueling operation, the RCS is assumed to have been cooled down to 140°F. The vessel head is removed, and the refueling canal is filled. This is done by pumping water from the refueling water storage tank (RWST), which is outdoors and conservatively assumed to be at 32°F, into the loops by means of the residual heat removal pumps. It should be conservatively assumed that the cold water flows directly into the reactor vessel and that all the fluid in the RCS is replaced with the colder water within 10 minutes. One cycle will be counted each time the refueling canal is flooded from the RWST. For example, if during a refueling outage, the cavity is filled, drained and subsequently refilled, then two cycles have occurred.

A License Renewal Application amendment is required.

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Question No: B.3.1-RH-02

NRC Request:
In the scope element of the Fatigue Monitoring Program, it states that the CVCS piping and Heat Exchangers are monitored.

- What is the purpose of monitoring these locations?
- How is this information applied to the FMP?

HNP Response:
(First Bullet) What is the purpose of monitoring these locations?

As stated in WCAP-16353-P, the primary reasons for including the CVCS Piping and Heat Exchangers are:

- It is difficult for operations to readily identify when a fatigue significant transient is happening.
- Fatigue significant transients may occur due to the automatic response of the pressurizer level control system.
- Multiple transients may occur during each plant heatup and cooldown operation.

(Second Bullet) How is this information applied to the FMP?

The current (unenhanced) HNP Fatigue Monitoring Program relies upon counting plant cycles and transients. This methodology may not identify all transients which could affect the CVCS Piping and Heat Exchangers for the reasons cited above. By monitoring actual fatigue usage of controlling locations, the CVCS Piping and Heat Exchangers can be monitored without relying only on counting plant cycles and transients. The charging nozzle is recommended as a controlling location for the CVCS Piping and Heat Exchangers. WCAP-16353-P recommends inclusion of the "CVCS piping and heat exchanger" into the HNP Fatigue Monitoring Program. At HNP, the CVCS System heat exchangers (including the Regenerative Heat Exchanger) are not classified as Class 1 and will therefore not be included in the program. The charging nozzle was identified in NUREG/CR-6260 as one of the components evaluated for environmental effects. The components identified in NUREG/CR-6260 were chosen to give a representative overview of components having higher CUFs and/or are important from a risk perspective. Industry wide, it is recognized that the nozzle is

generally limiting for most locations and therefore is a good representative component for the CVCS system and heat exchangers. HNP currently plans to include the charging nozzle to the components monitored using online fatigue monitoring software as a representative location to track fatigue usage for Class 1 CVCS piping. Monitored usage will consider both alternate and normal charging nozzles.

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Question No: B.3.1-RH-03

NRC Request:
Please explain how the periodic updates are documented (signature and records management). Also, do these updates undergo peer (either internal or external) review?

HNP Response:
The current (prior to enhancement) HNP Fatigue Monitoring Program requires signatures by the Shift Technical Advisor and the Superintendent, Shift Operations. The Shift Technical Advisor is responsible for the monthly evaluation of cyclic and transient data and the Superintendent, Shift Operations reviews the evaluation and cycle logs and forwards to document services to be stored as permanent records. Operations personnel provide an internal peer review to verify the monthly evaluations have been correctly completed.

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Question No: B.3.1-RH-04

NRC Request:
Does HNP have documented past records for cycle-counting prior to the Westems software installation?
If so, how is the documentation referenced?.

HNP Response:
HNP does not currently use the Westems thermal event monitoring module. Administrative cycle and transient counting is performed using HNP procedure: Cycle and Transient Monitoring Program. Documentation of each periodic update is maintained as a permanent plant record. The updated documentation provides a cumulative total number of design cycles and transients. Therefore, for the HNP fatigue evaluation for license renewal, only the latest cycle-counting records used and referenced in the basis documents.

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Question No: B.3.1-RH-05

NRC Request:
For the Fatigue Monitoring Program, please provide the following information:

- What are the established alarm limits?
- Do they vary from component to component?
- When an alarm limit is approached, how is the program owner informed?
- How is the above procedurally controlled?

HNP Response:
(Bullet 1) What are the established alarm limits?

See response to Question B.3.1-RH-01.

(Bullet 2) Do they vary from component to component?

- Cycle Counting Alarm Limits

  Yes, only the Steam Generator Secondary Manway Cover Bolts and 4 in. Inspection Port Bolts are managed differently as described in the response to B.3.1-RH-01.

- Stress-based Fatigue Monitoring

  No.

(Bullet 3) When an alarm limit is approached, how is the program owner informed?

As identified in LRA B.3.1 ("Corrective Actions"), the HNP Fatigue Monitoring Program will be enhanced to address corrective actions if an analyzed component is determined to be approaching the design limit, with options to revise the

fatigue analysis, repair, or replace the component. When program enhancements are implemented, the program will have established alarm limits for plant cycle and transient counts and alarm limits for monitored component usage values. When alarm limits are reached, corrective actions will be taken.

The corrective actions will include the following options:

Reanalysis

- The analysis methods for determination of stresses and fatigue usage will be in accordance with an NRC endorsed Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Power Plant Components Division 1 Subsection NB, Class 1 Components, Subarticles NB-3200 or NB-3600 as applicable to the specific component.
- HNP will utilize design transients from HNP Design Specifications as well as design transient information from typical PWR references to bound all operational transients. The numbers of cycles used for evaluation will be based on the design number of cycles and actual HNP cycle counts projected out to the end of the license renewal period (60 years).
- Environmental effects on fatigue usage will be assessed using methodology consistent with the Generic Aging Lessons Learned Report, NUREG-1801, Rev. 1, (GALL) that states; "The sample of critical components can be evaluated by applying environmental life correction factors (Fen Methodology) to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels."

Repair/Replacement

- Repair or replacement of the affected component(s) will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by HNP's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

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(Bullet 4) How is the above procedurally controlled?

When alarm limits are reached, corrective actions will be taken as described above. Corrective actions are procedurally controlled by the corporate Corrective Action Program (CAP) procedure. Corrective Actions are implemented in accordance with the requirements of Appendix B to 10 CFR 50.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 4.1-1

NRC Request:
Section 4.1- Identification of Time Limited Aging Analyses

Question 4.1-1 (SG Flow-Induced Vibration and Wear)

The HNP Model delta 75 RSGs were analyzed at uprated power conditions for flow-induced vibration and wear. Table 4.1-2 of the LRA does not identify this as a TLAA. Explain why flow-induced vibration (FVI) was not identified as a potential TLAA. Also, provide the basis the power uprate FVI and wear analyses remain valid for the period of extended operation.

HNP Response:
The time-dependent assumptions used in the flow-induced vibration calculation are based on a 40-year design life for the steam generators. The replacement steam generators were installed in 2001 and their design life, which extends to 2041, surpasses the current operating period. Therefore, criterion (3) of 10 CFR 54.3 (Involve time-limited assumptions defined by the current operating term, for example, 40 years) is not met. Loss of material for the steam generator tubes is managed by a combination of the Steam Generator Tube Integrity and the Water Chemistry Programs (See Table 3.1.1, Item Number 3.1.1-72, Page 3.1-34 of the LRA).

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Question No: LRA 4.1-2

NRC Request:
Section 4.1- Identification of Time Limited Aging Analyses

Question 4.1-2 (RCP Flywheel Evaluation)

NUREG-1800, Table 4.1-3 identifies the fatigue analysis of the reactor coolant pump flywheel as a potential TLAA. Table 4.1-2 of the LRA indicates that the fatigue analysis of the reactor coolant pump flywheel did not meet the TLAA criteria. Provide the basis for this statement.

HNP Response:
The evaluation prepared to support the inspection interval for inservice inspections of the reactor coolant pump flywheels is based on a plant life to be 60 years. Therefore, criterion (3) of 10 CFR 54.3 (Involve time-limited assumptions defined by the current operating term, for example, 40 years) is not met.

During discussions of the initial response to this question, the NRC staff requested information regarding the current inspection regimen for the RCP flywheels. This information is provided below.

Plant Technical Specification Amendment No. 119, Section 4.4.10, states:

> Each Reactor Coolant Pump Motor Flywheel be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August, 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

These inspection requirements are captured in the current Inservice Inspection Program Plan.

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Question No: LRA 4.3-1

NRC Request:
Section 4.3 Metal Fatigue

Question 4.3-1 (NSSS Transient Cycle Projections for 60-year Operation)

Explain the following differences between Table 4.3-1 of LRA (NSSS Transient Cycle Projections for 60-year Operation) and Table 3.9-1 of UFSAR (Summary of Limiting Reactor Coolant System Design Transients):

- LRA Normal Transients No.13 and 14 are not in UFSAR Table 3.9-1
- UFSAR Transient No. 5b (Inadvertent Auxiliary Spray Cooling) is not in LRA Table 4.3-1
- LRA Upset Transient 8 (Inadvertent Startup of an Inactive Loop) is not in the UFSAR Table 3.9-1
- LRA Test Transients are not the same as UFSAR Table 3.9-1

HNP Response:

1. LRA Transients No. 13 and 14 were introduced by WCAP-14778 (see Item Number 2 of LRA 4.3-16)
2. The inadvertent auxiliary spray transient is a sub category of the umbrella transient Inadvertent RCS Depressurization. The Inadvertent RCS Depressurization has 20 cycles with 10 of those cycles being the postulated as inadvertent auxiliary spray events. The inadvertent auxiliary spray events were not specifically listed, since the inadvertent auxiliary spray events were already included in the Inadvertent RCS Depressurization transients.
3. The inadvertent startup of an inactive loop was introduced in WCAP-14778 (see Item Number 2 of LRA 4.3-16)
4. The test transients in Table 4.3-1 of the LRA and FSAR Table 3.9-1 are the same but are listed in different locations. Please use the table below for reference:

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Transient	LRA Table 4.3-1 (page 4.3-22)	FSAR Table 3.9.1-1
Primary Side Hydrostatic Test	Test Transient 1	Test Condition 1
Secondary Side Hydrostatic Test	Test Transient 2	Test Condition 2
Primary Side Leak Test	Test Transient 3	Normal Condition 11
Secondary Side Leak Test	Test Transient 4	Normal Condition 12
Steam Generator Tube Leak Test	Test Transient 5	Test Condition 3

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Question No: LRA 4.3-2

NRC Request:
Section 4.3 Metal Fatigue

Question 4.3-2 (60 Year Projected Cycles)

Describe the method used to estimate the number of cycles for 60 years operation for the following transients listed in LRA Table 4.3-1:

- Normal Transient No. 10 (feedwater cycling)
- Upset Transient No.12 transient (RCS cold over-pressurization)

HNP Response:
The cycle projections will be removed from the License Renewal Application. Cycle projections will not be used to justify acceptability of fatigue-related TLAAs by 10 CFR 54.21(c)(1)(i) - the analyses remain valid for the period of extended operation.

The general deletion of cycle projections from the License Renewal Application affects the response to the following questions:

- LRA 4.3-3
- LRA 4.3-4
- LRA 4.3-5
- LRA 4.3-10
- LRA 4.3-16

The response to those questions will reference the response to this question concerning this issue.

A License Renewal Application amendment is required (to revise Table 4.3-1 and various subsections of Section 4.3 that used the cycle projection discussion on page 4.3-2 of the LRA as a qualification basis).

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Question No: LRA 4.3-3

NRC Request:
Table 4.3-2 (Design Fatigue Usage Factors) of the LRA indicates that the design fatigue usage factor for core internals is 0.52. Section 4.3.1.2 also states that, "Forty-year design CUF values were also determined as part of the HNP SG replacement/uprating analysis."

- Clarify whether the current licensing basis (CLB) for the 60-years fatigue evaluation of the reactor internals is based on the original design analysis or the SG replacement/uprating analysis.

HNP Response:
The 60-year fatigue evaluation for license renewal was done by WCAP-16353-P, Harris Nuclear Plant Fatigue Evaluation for License Renewal. This evaluation relies in part on previous evaluations including the HNP steam generator replacement/uprating analysis.

The CLB for 40-years of operation is based on a design usage factor of 0.52 for the reactor core internals. This is based on the HNP steam generator replacement/uprating report which identifies the design usage factor of 0.52 for the reactor internals (for 40 years of operations).

Multiplying this 40-year fatigue usage of 0.52 by 1.5 to account for 60 years of operation yields a CUF of 0.78 which does not exceed the design limit of 1.0. Therefore, the analysis has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii).

Since this evaluation no longer relies on cycle projections, the LRA requires revision.

The general deletion of cycle projections as a qualification basis for fatigue-related TLAAs is addressed in the response to LRA 4.3-2.

A License Renewal Application amendment is required.

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Question No: LRA 4.3.1.6-1

NRC Request:
Section 4.3 Metal Fatigue

Question 4.3.1.6-1 (Pressurizer Fatigue Analysis)

LRA section 4.3.1.6 states, "Certain locations of the pressurizer lower head are not bounded by the original design fatigue analysis" and "Recommendations of the Westinghouse Owners Group (WOG) were used to address operational pressurizer insurge/outsurge effects."

- Discuss the modified operating procedure used to mitigate the pressurizer insurge/outsurge transients.
- Describe how the fatigue usage prior to the use of modified operating procedures was captured in the fatigue evaluation.
- Describe how the estimated 60 year fatigue usage factors for the pressurizer lower head locations and the surge line RCS hot leg location shown in Table 4.3-2 were calculated.

HNP Response:

- Discuss the modified operating procedure used to mitigate the pressurizer insurge/outsurge transients.

The HNP operating procedures for plant startup, normal plant heatup, power operation, normal cooldown and shutdown operations have been modified to include measures to manage insurge/outsurge transients. Westinghouse Owner's Group recommendations provided in WCAP-14950, *Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients*, were included in the modified operating procedures for these plant operating modes. These recommendations were proceduralized on 01/20/94.

Generic operational strategies to mitigate pressurizer insurge/outsurge transients and their effects fall into in two main categories, overall and local. The overall strategies are aimed at establishing conditions during the overall heatup and cooldown procedures that would generally help to prevent or reduce the severity of insurge/outsurge transients. The local strategies focused on particular operations, their effects on transients, and counteracting these effects.

Overall strategies

1. Maintain continuous pressurizer outsurge flow during heatup and cooldown operations
2. Minimize the system ΔT

The first overall strategy helps to decrease the occurrences of insurge transients, keeping flow in the opposite direction and maintaining the lower head at essentially uniform pressurizer saturation conditions. The second reduces the severity of insurge and subsequent outsurge transients if they happen to occur.

Local Strategies

Local strategies deal with controlling RCS coolant mass inventory and temperature. Major areas included control of charging and letdown operations and heat inputs. The major goal in avoiding pressurizer transients during heatup and cooldown is to anticipate particular operations that may cause insurges, and counteract their effects.

For other operations that could cause insurges the strategy is to identify those that could cause insurges, and first attempt to revise the procedure so as to avoid the insurge. If the insurge cannot be avoided, then the operation should be performed when the system ΔT is low enough to make the transient insignificant.

Details of these and other specific measures taken to mitigate the effects of insurge/outsurge transients are included in the plant operating procedures which are available for review.

- Describe how the fatigue usage prior to modification of the operating procedure was captured in the fatigue evaluation.

WCAP-16376-P, *Evaluation of Pressurizer Insurge/Outsurge Transients for Harris Nuclear Plant*, and WCAP-16353-P, *Harris Nuclear Plant Fatigue Evaluation for License Renewal* contains the evaluation of the components affected by insurge/outsurge transients. Details concerning treatment of pre-Modified Operating Procedure transients and the fatigue evaluations for license renewal are contained therein are available for review.

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- Describe how the estimated 60 year fatigue usage factors for the pressurizer lower head locations and the surge line RCS hot leg location shown in Table 4.3-2 were calculated.

The detailed evaluation of the pressurizer lower head and surge line locations is contained in WCAP-16376-P, *Evaluation of Pressurizer Insurge/Outsurge Transients for Harris Nuclear Plant*, and WCAP-16353-P, *Harris Nuclear Plant Fatigue Evaluation for License Renewal* and are available for review.

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Question No: LRA 4.3.1-1

NRC Request:
The basis document (CN-PAFM-04-136 Rev. 0) indicated that cycles up-to-date were evaluated by scaling cycles using the plant data from 7/19/1999 to 10/18/2004 to determine the cycles for early operation years. Please discuss the validity for this backward projection since operating experience has demonstrated that more transient cycles were experienced in early operation of nuclear plants.

HNP Response:
Two locations used sample data for the purposes of estimating past and future transient cycles. These were the pressurizer, which included the surge line and pressurizer lower head region, and the charging nozzles. Each was treated in a slightly different manner, which will be discussed below. However, both systems are subject to local thermal transient conditions that are the result of the normal control systems' actions that occur during normal plant operations, such as plant heatup and cooldown operations. These local thermal events, such as the thermal stratification of the pressurizer surge line or the de-stratification of the pressurizer surge line, are too subtle to be tracked in the normal manner that utilizes macroscopic changes in the plant status. These events are generally the result of the normal automatic response of the plant control systems.

The original plant computer systems were never designed to retain all of the information in the resolution necessary for accurate reconstruction of local thermal transients acting on the charging line nozzles and pressurizer lower head and surge line. Therefore, a method of sampling combined with operations interview was employed to estimate the effects of past operations. In some cases, additional information was obtained through the use of direct measurement with temporary sensors attached to the outside of piping system as was the case with the HNP pressurizer surge line. For each of the systems under consideration, the sample observations were correlated with known plant events, which are tracked like plant heatup and cooldown operations.

Pressurizer Surge Line

In the case of the pressurizer surge line, the sample data set was further categorized by what is termed pre-Modified Operating Procedures and Modified Operating Procedures (MOPs), which are employed to reduce the frequency and

severity the thermal stratification events in the pressurizer lower head. Data collected from operations prior to implementation of the MOPs was treated separately from the data collected after the implementation of MOPs. In this manner the favorable reduced frequencies associated with the current operating practices are not introduced into the estimates used for past operations, where the MOPs were not employed. This resulted in development of at least two distinct transient sets that were analyzed for the pressurizer lower head and surge line. The data collected was also used to create a conservative set of surge line only events for the period of operation that pre-dated the implementation of the Modified Operating Procedures. The frequency and severity of events observed in the sample sets were then used with the known events to establish the total transients experienced to date.

Charging Nozzles

The charging nozzles experience their worst transients when the plant is at normal operating conditions. Charging events that occur during plant heatups and cooldown experience lower ΔT events because of the lower temperatures of the RCS. Therefore, to address this and considering the impact of the equivalent cycles reduction based on observed ΔT, the total charging cycles were developed using the total number of years of operations times the total number of equivalent cycles observed in the sample data set. It should be noted that the Modified Operating Procedures do not affect the frequency of charging or letdown isolation events.

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Question No: LRA 4.3.1-3

NRC Request:
In the basis document (CN-PAFM-04-128, R0 (page 41)), the applicant applied cycle reduction methodology derived from B.31.1 code to reduce 36 cycles to 1 cycle for ASME CL 1 components. Please provide justification to address ASME Code compliance issue.

Note: B.31.1, ASME CL 2 & 3 thermal effects do not consider the temperature rate change as ASME CL 1 component class. ASME Code does not allow class 1 component uses this reduction methodology.

HNP Response:
This response is related to the response to LRA 4.3-7.

An independent ASME Section III, Division I, Subsection NB fatigue evaluation has been performed in order to establish a quantitative basis for the application of the ANSI B31.1 cycle reduction methodology to cycle counting of the HNP charging nozzle transients. The discussion below summarizes this effort.

A model representative of the HNP charging nozzle, created using the finite element program ANSYS, was evaluated for fatigue using the program WESTEMS™, developed by Westinghouse. The ANSYS finite element model was a 3 dimensional model consisting of a 100-inch section of the RCS Loop piping section including the entire charging nozzle and branch piping for a length of approximately 4 pipe diameters. The model was loaded with mechanical and thermal loadings in a manner consistent with the assumptions used in the original design basis fatigue evaluation.

Two sets of transients were created to make the comparison between the reduced equivalent full cycle transient set and the actual observed transient set. To make the comparison and to maintain conservatism, the original design transient temperature time histories and rates were employed with reduced ΔT's. This effectively removed the added effect of lower rates experienced by the actual transients (which would make them even less severe with respect to stress). The result is a conservative representation of the actual events. The tables below summarize the two sets used. Table 1 lists the original design transient cycles, the actual (reduced ΔT) counts, and equivalent full range counts.

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Table 1 Charging design transients used to make this comparison

Transient Name	Allowable (Design) Cycles	Enveloped Counts from actual 5.26 years	Equivalent Full Range Cycles
Charging and Letdown Flow Shutoff and Return to Service	36	36	1
Letdown Flow Shutoff with Prompt Return to Service	120	36	4
Letdown Flow Shutoff with Delayed Return to Service	12	9	0
Charging Flow Shutoff with Prompt Return to Service	12	0	0
Charging Flow Shutoff with Delayed Return to Service	12	0	0
Charging Flow Step Decrease and Return to Normal	14400	0	0
Charging Flow Step Increase and Return to Normal	14400	0	0
Letdown Flow Step Decrease and Return to Normal	1200	11	3
Letdown Flow Step Increase and Return to Normal	14400	364	11

Only the flow shutoff transients were considered in this investigation. This is valid due to the low number of observed events of the flow increase/decrease transients, and the fact that the charging and letdown flow step decrease and charging and letdown flow step increase transients do not significantly contribute to fatigue.

After excluding the flow increase/decrease transients, the 81 highest ΔT events observed were used. To reduce the number of transient curves that had to be created, the 81 actual ΔT events were conservatively grouped into the ΔT

transient categories shown in Table 2:

Table 2 Actual Charging Transients Analyzed

Transient	Cycles	Temperature ΔT
Charging and Letdown Flow Shutoff and Return to Service	1	433
Charging and Letdown Flow Shutoff and Return to Service	1	333
Charging and Letdown Flow Shutoff and Return to Service	2	247
Charging and Letdown Flow Shutoff and Return to Service	1	207
Charging and Letdown Flow Shutoff and Return to Service	20	120
Charging and Letdown Flow Shutoff and Return to Service	11	33
Letdown Flow Shutoff with Prompt Return to Service	4	360
Letdown Flow Shutoff with Prompt Return to Service	4	340
Letdown Flow Shutoff with Prompt Return to Service	3	320
Letdown Flow Shutoff with Prompt Return to Service	3	290
Letdown Flow Shutoff with Prompt Return to Service	4	250
Letdown Flow Shutoff with Prompt Return to Service	2	175
Letdown Flow Shutoff with Prompt Return to Service	12	115
Letdown Flow Shutoff with Prompt Return to Service	4	20
Letdown Flow Shutoff with Delayed Return to Service	9	112

Table 3 shows the calculated equivalent full design transients and cycles analyzed to make the comparison.

Table 3 Design Transients Analyzed

Transient	Cycles	Temperature ΔT
Charging and Letdown Flow Shutoff and Return to Service	1	460
Letdown Flow Shutoff with Prompt Return to Service	4	460

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The equivalent full design transient cycles were determined using the concept provided in Section 102.3.2 of the B31.1 Power Piping Code, which provides a rule for computation of equivalent full temperature cycles of expansion stress when the range of temperature varies. Per the B31.1 approach, equivalent full temperature cycles are computed as follows:

$$N = N_e + r_1^5 * N_1 + r_2^5 * N_2 + \ldots r_n^5 * N_n \qquad \text{(Eq. B-1)}$$

Where:

N_e = number of cycles at full temperature change, T_e, for which the thermal design stress was calculated
$N_1, N_2, \ldots N_n$ = number of cycles at lesser temperature changes $\Delta T1, \Delta T2, \ldots \Delta Tn$
$r_1, r_2, \ldots r_n$ = $\Delta T1/\Delta Te, \Delta T2/\Delta Te, \ldots \Delta Tn/\Delta Te$

The application of this method resulted in the equivalent full cycles listed in Table 1 above. To show the validity of the B31.1 Code application to this component, two detailed fatigue evaluations were performed using the actual cycles in Table 2 and using the 5 cycles of the design transients in Table 3. The expected result is that the fatigue usage factor from the equivalent full design transients will be the same or greater than the fatigue usage from the actual transients.

Several locations were investigated using this method. The table shows the results for the limiting locations in the branch piping section (safe end) and the reinforcement region (crotch) of the charging nozzle.

Table 4 Fatigue Usage Comparison

Component	Analysis Section Number	Node	Node Number	Actual Cycles Usage	Equivalent Design Cycles Usage	Equivalent Design - Actual (dU)
212100	1	Inside	1	0.001	0.002	0.00100
212100	1	Outside	11	2.97E-06	9.65E-05	0.00009
212100	2	Inside	1	0.001	0.002	0.00100

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Component	Analysis Section Number	Node	Node Number	Actual Cycles Usage	Equivalent Design Cycles Usage	Equivalent Design - Actual (dU)
212100	2	Outside	11	2.97E-06	9.65E-05	0.00009
212100	3	Inside	1	0.001	0.002	0.00100
212100	3	Outside	11	2.97E-06	9.64E-05	0.00009
212100	4	Inside	1	0.0037	0.0143	0.01060
212100	4	Outside	11	4.51E-06	3.00E-04	0.00030
212100	5	Inside	1	0.0037	0.0141	0.01040
212100	5	Outside	11	4.74E-06	3.00E-04	0.00030
212100	6	Inside	1	0.0036	0.0139	0.01030
212100	6	Outside	11	5.02E-06	3.00E-04	0.00029
212100	7	Inside	1	0.0026	0.011	0.00840
212100	7	Outside	11	0.0011	0.0015	0.00040
212100	8	Inside	1	0.0017	0.0101	0.00840
212100	8	Outside	11	0.0008	0.0011	0.00030
212100	9	Inside	1	0.0019	0.0093	0.00740
212100	9	Outside	11	0.0005	0.0007	0.00020
212100	10	Inside	1	0.0107	0.0276	0.01690
212100	10	Outside	11	6.59E-07	6.99E-07	0.00000
212100	11	Inside	1	0.0069	0.0224	0.01550
212100	11	Outside	11	7.11E-07	7.95E-07	0.00000

The limiting location is ASN 10, inside node, which is a cut located in the reinforcement region of the charging nozzle and RCS loop piping intersection.

The results shown in the table above demonstrate that the use of the equivalent full cycles is conservative, since the equivalent design cycles usage values are all greater than the values obtained using actual cycles.

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Question No: LRA 4.3.3-1

NRC Request:
Section 4.3 Metal Fatigue

Question 4.3.3-1 (Environmentally- Assisted Fatigue Analysis)

Table 4.3-3 provides the 60-year Environmentally-Adjusted CUFs.

- Provide Fen value for each component
- Describe how the 60-year environmentally adjusted CUF and 40-year design CUFs have the same CUF value of 0.173 for RHR Class 1 Piping.

HNP Response:
Provide Fen value for each component

Table 4.3-3 in the LRA will be revised as follows:

TABLE 4.3-3 60-YEAR ENVIRONMENTALLY-ADJUSTED CUF VALUES

Component	60-Year Environmentally Adjusted CUF (U_{en})	F_{en}	A	B	C
Bottom Head Juncture	0.0491	2.532			
Reactor Vessel Inlet Nozzles	0.0231	2.532			
Reactor Vessel Outlet Nozzles	0.1740	2.532			
Surge Line	2.120 (Note 3)	8.27 (Note 1)	X	X	X
Charging Nozzle	0.89	7.2	X		X

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Component	60-Year Environmentally Adjusted CUF (U_{en})	F_{en}	A	B	C
		(Note 1)			
Safety Injection Nozzle	0.93	5.3 (Note 1)	X		
RHR Class 1 Piping	0.465	2.55 (Note 2)			
Pressurizer (Lower Head at Heater Penetration)	1.35 (Note 4)	2.532	X	X	X

A. Reduced cycles used in the evaluation
B. Refined calculations performed
C. Redefined transients used in the evaluation

Note 1. The "overall effective Fen" based on evaluation of the fatigue transient pairings. Each transient pair has its own unique Fen.

Note 2. The transients used for the RHR line qualification include only one significant transient defined for "RHR section 2" (RHR section 2 is the section of piping downstream of the isolation valve that is normally at ambient conditions), for RHR initiation when this part of the line goes from ambient conditions to the 350°F RHR letdown temperature. The return phase of the transient is a gradual cooldown with which no appreciable stress is associated. Since the temperature shock for the RHR initiation transient is positive, the stresses on the inside surface of the piping components are compressive. Since the strain rate is compressive, Fen = 1.0 for this controlling condition would be appropriate, based upon the methodology of NUREG/CR-5704. However, a maximum value of 2.55 has been used for the Fen for this evaluation.

Note 3. 40-year design transients were used in the evaluation except for heatups and cooldowns. The number of heatup and cooldown transients used in the analysis is 133 versus 200 original design transients. The fatigue usage for this location based on this transient set is 0.94. Multiplying by 200/133 to account for the full set of

design heatups and cooldowns yielded a 40-year fatigue usage of 1.414. Multiplying by 1.5 (that is 60 years/40 years) yields a 60-year fatigue usage of 2.120.

Note 4. 40-year design transients were used in the evaluation except for heatups and cooldowns. The number of heatup and cooldown transients used in the analysis is 133 versus 200 original design transients. The fatigue usage for this location based on this transient set is 0.5984. Multiplying by 200/133 to account for the full set of design heatups and cooldowns yielded a 40-year fatigue usage of 0.9. Multiplying by 1.5 (that is 60 years/40 years) yields a 60-year fatigue usage of 1.35.

Describe how the 60-year environmentally adjusted CUF and 40-year design CUFs have the same CUF value of 0.173 for RHR Class 1 Piping.

To address the question regarding the effect of the pressure and moment stresses on the total stress and resulting Fen, a detailed time history of the stress during the full RHR transient cycle was developed for the 12" x 1" branch component. The stresses were calculated consistent with the HNP design basis inputs and NB-3600 evaluation methodology. Applicable thermal transient, moment, and pressure histories were applied representing the RHR system design transient in the section of the RHR line containing the branch connection. The transient temperature goes from ambient conditions, at 70°F, to 350°F simulating the initiation of RHR flow through the otherwise isolated section of piping. The design transient description also assumes after the transient that gradual cooling of the system occurs at 100°F/hr, which was considered to represent negligible stress in the design. The RHR flow is initiated by design at approximately 425 psig system pressure. Therefore, a pressure load of 450 psia was used in the calculation. The moment stress from the piping design analysis of the branch was realistically ramped over the transient as a function of the pipe temperature. Pressure and moment stresses were considered only as tensile. The thermal stresses were appropriately signed to represent the inside wall stress in response to the thermal transient. The stresses were adjusted with the branch stress indices consistent with the Code analysis. The resulting stress history is illustrated in Figure 1.

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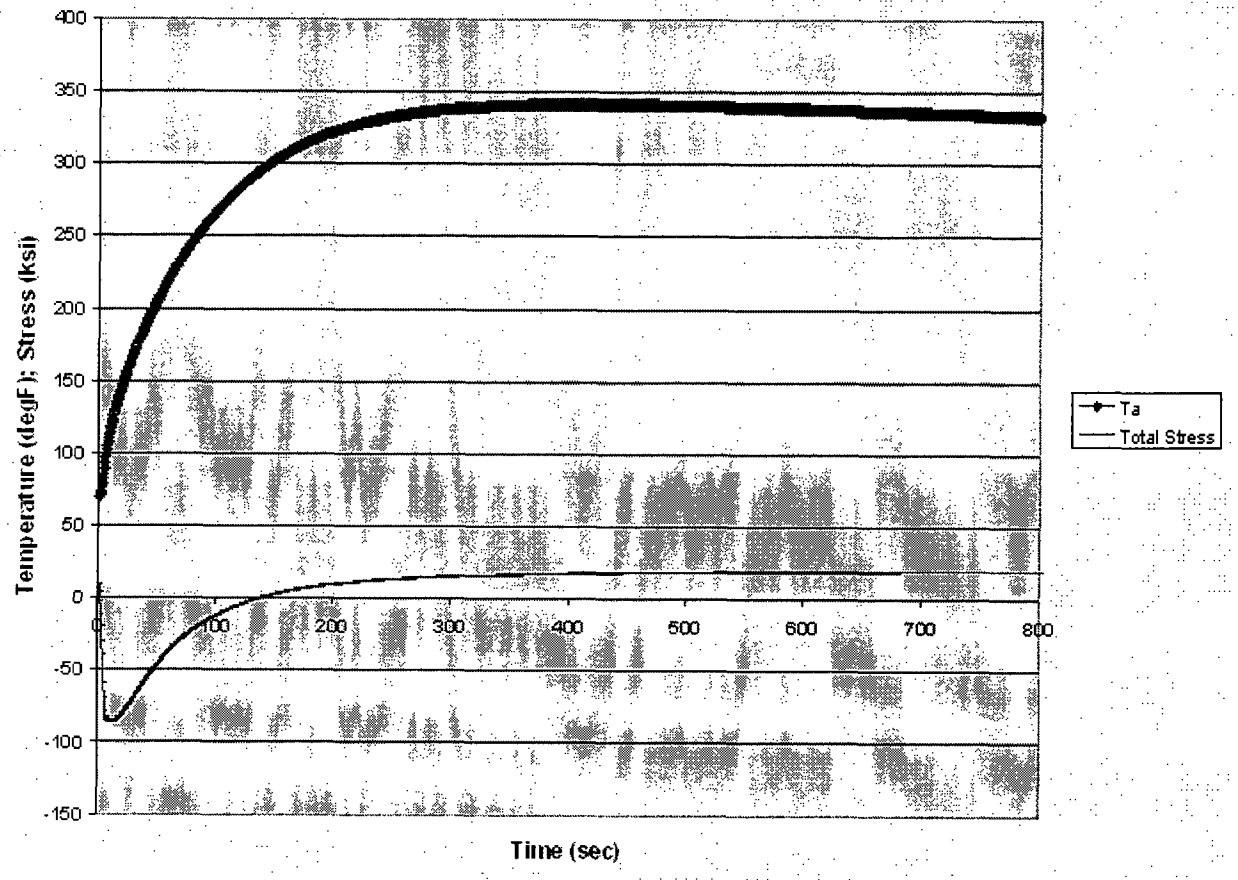

Figure 1 - RHR Transient Temperature and Stress Response for Branch Component

This demonstrates that for the significant part of the transient history, the component is in the compressive state on the inside surface where environmental effects may be applicable. For this reason, in the initial estimation of environmental effects on this component, it was judged that environmental effect would be small and Fen = 1.0 was applied. However,

the time history shown does include some periods where stress is in the tensile regime due to the pressure and moment loads. Therefore, an overall effective Fen was calculated for the entire transient cycle. Strain was determined as the stress divided by elastic modulus (E). The Fen value was calculated by integrating the Fen over the strain history using the modified rate approach (integrating with respect to strain range; methodology discussed in the literature, e.g. reference 1), represented simply by the following equation:

$Fen = \sum (Fen_i * \Delta\varepsilon_i) / \sum \Delta\varepsilon_i$

where:

Fen = exp (0.935 - T' ε' O') [see NUREG/CR-5704 for definition of terms], ε= strain

The integration was carried out to the essentially steady-state condition at the end of the transient (around 400 sec). For the beginning portion of the cycle, with a negative strain rate, there is no environmental effect based on NUREG/CR-5704, which states that the environmental effect is only effective in the tensile direction of the cycle. In the second portion of the cycle, with a positive strain rate, the applicable Fen is the threshold value for stainless steel, which is due to the temperature input to the stainless steel Fen equation. The resulting overall integrated value is:

Fen = 1.15

This should be applied to the usage calculated from the stress pair formed from the stress peaks represented by the minimum stress state (maximum compression) and the maximum stress state (maximum tension) of the transient. The total stress range in Figure 1 is approximately 102 ksi. Using the ASME design fatigue curve for stainless steels, this corresponds to approximately 1680 allowable cycles. For 200 design cycles of the RHR transient, the usage factor is 0.12. Applying Fen = 1.15, Uen = U * Fen = 0.13.

This results in a lower fatigue usage than was conservatively reported from the HNP design evaluation, which reported the 0.173 value, and supports the judgment that for this predominantly compressive cycle that application of Fen = 1.0 to the design usage is adequate.

A conservative alternative is to consider the maximum value of Fen obtained during the entire cycle, which was 2.55, and

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apply to the usage to obtain Uen = 0.31, which is still acceptable with respect to the ASME allowable of 1.0. Accounting for 60 years of operation by multiplying by 60/40 yields 0.465.

Reference 1: Sakaguchi, Katsumi, et. al., "Applicability of the Modified Rate Approach Method Under Various Conditions Simulating Actual Plant Conditions," PVP2006-ICPVT-11-93220, Proceedings of PVP2006-ICPVT-11, 2006 ASME Pressure Vessels and Piping Division Conference, July 23-27, 2006, Vancouver, BC, Canada.

A License Renewal Application amendment is required.

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Question No: LRA 4.3.3-2

NRC Request:
Applicant's basis document, "Harris License Renewal Piping Environmental Fatigue Evaluation", (CN-PAFM-04-136, Rev. 0, page 7), stated that Fen = 1.0 for the pair if any one of the following conditions satisfied:

T <= 180°C
ε' > 0.4% / sec
DO => 0.05 ppm
Strain amplitude, <= 0.10%

Please clarify the above statement with its basis since the minimum Fen value should be 2.54 as indicated below. If the statement is not valid, please clarify what impact to the entire calculation.

Note: Fen = exp (0.935 - T' ε' O')
Where T' = 0 (T < 180°C) or any condition would make the second term of the equation equal 0, then Fen = exp (0.935) = 2.54

HNP Response:
The wording cited from the text of the calculation is ambiguous and can be misleading about what was actually calculated with respect to the threshold values. The only application of Fen = 1.0 for a threshold limit used in the actual calculations was for a strain amplitude threshold of 0.10%. (For the threshold values of the other parameters, when strain amplitude is > 0.10%, the minimum value of Fen is 2.54.) This is evident upon review of the detailed Fen calculation tables provided in the document. The strain amplitude threshold was based on the NUREGs as discussed below.

The value of Fen = 1.0 is used in the calculations to represent the condition where the environmental effect is insignificant as stated in NUREG/CR-5704, and clarified in NUREG/CR-6717. In these cases, these documents state that no additional environmental penalty is required beyond the factors included in the ASME design fatigue curve. This was based on the statements in the NUREGs noted below:

- In Section 6 of NUREG/CR-5704, it states: "the existing fatigue data indicate a threshold strain range of ≈0.32%,

below which environmental effects on the fatigue life of austenitic SSs either do not occur or are insignificant. ... Thus a threshold strain amplitude of 0.097% (stress amplitude of 189 MPa) was selected, below which environmental effects on life are modest and are represented by the design curve for temperatures <200°C..."

- NUREG/CR-6717 provides further study of the ANL data and results presented in NUREG/CR-5704, and reiterates the strain amplitude threshold for stainless steels in section 5.3: "The strain threshold is represented by a ramp, i.e., a lower strain amplitude below which environmental effects are insignificant, a slightly higher strain amplitude above which environmental effects are significant, and a ramp between the two values... The two strain amplitudes are ... 0.10 and 0.11% for austenitic SSs..." The lower strain amplitude value of 0.10% is a consistent rounded value from the 0.097% value discussed in NUREG/CR-5704.

- NUREG/CR-6909 for new plant applications also continues to reiterate the strain amplitude threshold for stainless steels in Appendix A: "For wrought and cast austenitic stainless steels, a threshold value of 0.10% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. Thus, Fen,nom =1 ($\varepsilon a \leq 0.10\%$)."

Based on the statements in NUREG/CR-5704 and NUREG/CR-6717, as well as the confirmation of the application provided in NUREG/CR-6909, when the strain amplitude for a fatigue pair was below 0.10%, no additional Fen penalty was applied to the pair. This is effected numerically in the calculations by setting Fen = 1.0. In CN-PAFM-04-136, this is reflected in tables calculating Fen (e.g., Table 6-8) for fatigue pairs with alternating stress (Sa) less than 28.3 ksi, which corresponds to 0.10% strain amplitude for this case.

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Question No: LRA 4.3.4-1

NRC Request:
Section 4.3 Metal Fatigue

Question 4.3.4-1 (RCS Loop Piping LBB Analysis)

LRA Section 4.3.4 indicates that a new LBB calculation applicable to HNP large bore RCS piping and components was performed and documented in WCAP-14549-P, Addendum 1. Provide the following information regarding the LBB calculation:

- LRA Section 4.3.4 states, "However, the calculations for Alloy 82/182 locations were performed, and this material is not bounding." Explain the basis for this statement.
- LRA Section 4.3.4 states," Ample margin exists in stability using the 60 year, the end of life thermal aging material properties." Discuss the acceptance criteria used for the evaluation and quantify the margin.
- The LRA states that "SCC is precluded," please clarify.
- Please clarify why the A82/182 material is not bounding in the evaluation.

HNP Response:

- Plant-specific calculations were performed (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*) to account for PWSCC crack morphology in that a conservative factor between fatigue cracking and PWSCC was used. When the EPRI MRP methodology described in MRP-140, *Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds*, has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.

- Bullet 6 is not required and will be deleted.

- In the context of the discussion of Bullet 1 on page 4.3-15 of the LRA, SCC is precluded for the materials other than A82/182. PWSCC of the A82/182 material has been accounted for in the plant-specific calculation (WCAP-

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14549-P, Addendum 1) performed for License Renewal.

The A82/182 material is not bounding because the flaw sizes yielding a leak rate of 10 gpm were larger than those calculated at other locations that do not contain A82/182 material. As stated in the response to Bullet 1 above, HNP will review its plant-specific calculation for conformance with the endorsed approach.

A License Renewal Application amendment is required.

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Question No: LRA 4.3.3-3

NRC Request:
During review the basis documentation of fatigue evaluation, the original transients were used in the analysis and did not mention the fatigue cycling related to Bulletin 88-08. Please explain how Bulletin 88-08 was addressed in the metal fatigue.

HNP Response:
Fatigue cycling related to Bulletin 88-08 is managed by procedure titled, *SI Thermal Stratification Monitoring Program.*

This program was implemented in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," which requested that all LWR licensees review any unisolable piping connected to the RCS which may be subjected to unacceptable thermal stresses caused by thermal stratification. The Bulletin further called for licensees to "take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. Guidelines for evaluating the effectiveness of the Thermal Stratification Monitoring Program were established by the NRC in a follow-up letter dated September 16, 1991. The guidelines provide evaluation criteria for licensee responses to NRC Bulletin 88-08, Action 3, and Supplement 3. Four acceptable actions were presented by the NRC. Of these, HNP chose option 3 which was to install temperature instrumentation to monitor pipe temperature gradients due to containment isolation valve leakage. HNP has identified High Head Safety Injection Cold and Hot Legs, Normal Charging, and Alternate Charging lines as being susceptible to thermal stratification. Thermocouples were installed at the following check valve locations: 1SI-81, 1SI-82, 1SI-83, 1SI- 136, 1SI-137, 1SI-138, 1CS-486, and 1CS-500. The thermocouples measure the top and bottom pipe temperatures immediately downstream to each check valve.

The Reactor Coolant Pressure Boundary Fatigue Monitoring Program addresses the possibility of fatigue cracking due to thermal stresses caused by thermal stratification in the safety injection, normal charging, and alternate charging lines if certain thermal limits, identified by this procedure, are exceeded.

Per the current plant procedure, the temperature time-histories are evaluated by the Responsible Engineer for the following conditions:

- Average top-to-bottom temperature differences exceed 100°F.
- Top and bottom temperature time histories become significantly out of phase.
- Bottom temperature oscillations exceed 50°F peak to peak.
- External leakage is detected in closed isolation valves.

The screening criterion of a 100°F average top-to-bottom temperature difference was chosen because HNP performed a plant-specific analysis that demonstrated that the resulting loads and stresses will not threaten the ability of the piping systems to operate safely.

Since HNP monitors these lines to ensure that the piping will not be subjected to unacceptable thermal stresses, fatigue cycling related to Bulletin 88-08 did not need to be incorporated in the basis documents.

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Question No: LRA 4.3.5-1

NRC Request:
LRA 4.3.5-1 states "For the purposes of this evaluation, a penetration isolation lasting for more than 10 minutes while the RCS hot leg temperature exceeds 500F is conservatively considered one thermal cycle. Please provide the basis for why transient has to last more than 10 minutes to be considered one thermal cycle.

What is HNP's RCS sampling interval? Is it cycled every time sampling piping is used to sample the RCS liquid?

HNP Response:
The subject TLAA tubing lines are not cycled every time sampling piping is used to sample the RCS liquid. Therefore, HNP's RCS sampling interval has no influence on cycle counting.

This TLAA is associated with the safety related portion of the sample system, which extends from the RCS hot legs in the Reactor Containment Building to the normally open outboard isolation valve in the Reactor Auxiliary Building. License Renewal Scoping Drawing, Attachment 15 (5-G-0052-LR) of the basis document shows this configuration. There is continuous flow through penetration M78A during reactor startup, power operation and shutdown. The RCS coolant continuously flows to the Gross Failed Fuel Detector (GFFD). The GFFD is located downstream of the penetration in the non-safety related portion of the system and outside the boundary of the TLAA. The source of coolant for the Primary Sample Panel (shown on the scoping drawing), which samples RCS coolant, taps off the tubing line between the outboard isolation valve and the GFFD. RCS flow through the safety related portion of the sample system is not interrupted when the primary sample panel is used. Consequently, operation of the sample panel has no influence on thermal cycles.

The current methodology considered one cycle as an interruption of a continuous supply of hot RCS coolant through penetration M78A for more than 10 minutes. The HNP methodology has been changed to consider an interruption of primary coolant flow for more than 5 minutes as one cycle. In 5 minutes the temperature of the primary sample system tubing was conservatively estimated to decrease less than 200°F. Appendix H, Section 4.1.1 of the EPRI Report TR-1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 4, discusses the basis for the 270°F system screening criterion for fatigue in stainless steel. This temperature value was chosen as a temperature change in components of more than 200°F is not anticipated in these systems. Consequently, 200°F was chosen as a

thermal cycle in the primary sample system.

A review of the original data trends was performed using 5 minute closure times. This review resulted in counting an additional 8 cycles; changing 22 cycles in the original analysis to 30 cycles. The projected increase over 60 years is an additional 72 cycles. The total number of cycles for the primary sampling system is then increased from 1279 cycles to 1351 cycles. The overall conclusion remains the same as the number of thermal cycles remains far below the criterion where stress range reduction factors would have to be reduced below unity for fatigue i.e. 7,000 cycles.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-4

NRC Request:
In Section 4.3.1 of the LRA, there is a discussion of the extrapolation methodology for cycle projections. Please justify the validity of the method in projecting various transient cycles to the end of the extended period of operation.

HNP Response:
For our purposes we have chosen to use linear extrapolation of the average observed frequency to estimate the total cycles for 60 years of operations. This is deemed acceptable for four reasons:

1) The use of typical trend analysis tools like linear regression using least squares curve fitting would yield unrealistic results due to the reduced frequencies of unplanned events like reactor trips in the most recent data. These methods result in trends with negative slopes that fail to account for the normal planned operational cycles.
2) Use of the average of the observed events extrapolated out through the extended operational life yields non-zero numbers for future events and conservatively introduces higher rates than one would predict using traditional trend analysis methods.
3) The cycles predicted by linear extrapolation yield higher numbers of plant heatup and cooldown events than one would get if fuel cycle durations only were used.
4) A transient and fatigue cycle monitoring will be employed to insure that adequate margins exist between the actual cycles and the design cycles.

However, the discussion of cycle projections will be removed from the License Renewal Application and the individual evaluations in Section 4.3 will be updated accordingly. The column labeled "60 Year Projected Cycles" will be removed from Table 4.3-1.

The general deletion of cycle projections as a qualification basis for fatigue-related TLAAs is addressed in the response to LRA 4.3-2.

A License Renewal Application amendment is required.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 4.3-5

NRC Request:
In Section 4.3.1 of the LRA (page 4.3-2, third paragraph), there is a discussion on the comparison of the 60-year cycle projections and its comparison to 40-year design cycles. Please clarify why it is appropriate to use 60-year projection to determine the adequacy of the current fatigue analyses based on 40 years.

Please clarify whether (or how) the projected cycles are used in developing Tables 4.3-2 and 4.3-3.

HNP Response:

1. The use of the 60 years cycles projections shows that the existing design frequencies are conservative and therefore still valid even after 60 years of operations. This statement remains true as long as the 60 years projected cycles are less than or equal to the current design frequencies.
2. In general, design numbers of cycles were used for fatigue calculations whose results are presented in Table 4.3-2. The exceptions to this are provided in Notes 1 and 2 to Table 4.3-2; specific Steam Generator sub-components, the Pressurizer lower head, and pressurizer surge line were re-qualified based on the projected cycles. In Table 4.3-3, the results for the surge line, charging nozzle, safety injection nozzle and pressurizer lower head used projected cycles. More information on these locations is provided in the response to question LRA 4.3.3-1. An online stress and fatigue cycle monitoring system has been installed to maintain continuous surveillance of the operational fatigue damage accumulation rate for the pressurizer lower head and surge line.

However, the discussion of cycle projections will be removed from the License Renewal Application and the individual evaluations in Section 4.3 will be updated accordingly. The column labeled "60 Year Projected Cycles" will be removed from Table 4.3-1.

The general deletion of cycle projections as a qualification basis for fatigue-related TLAAs is addressed in the response to LRA 4.3-2.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-6

NRC Request:
In Section 4.3.1 of the LRA (page 4.3-2, first paragraph), there is a discussion of pressurizer surge line analyses. Please provide the chronology for the qualification of the surge line beginning with IE Bulletin 88-11 through License Renewal.

HNP Response:
In September 1991, WCAP-12962, *Structural Evaluation of the H. B. Robinson Unit 2 and Shearon Harris Pressurizer Surge Lines, Considering the Effects of Thermal Stratification* was issued. This report was prepared to demonstrate compliance with the requirements of NRC Bulletin 88-11 for H. B. Robinson Unit 2 and Shearon Harris. Prior to the issuance of the bulletin, the Westinghouse Owners Group had a program in place to investigate the issue and to recommend actions by member utilities. That program provided the technical basis for the analysis in WCAP-12962 for H. B. Robinson Unit 2 and Shearon Harris.

As part of the Steam Generator Replacement/Power Uprate (SGR/PU) project, a reanalysis was performed by Westinghouse (CN-SMT-99-66) that evaluated the effects of steam generator replacement, the revised center of gravity of the steam generator, and the power uprate. This calculation was incorporated as an addendum to HNP calculation 3043W. The results of this analysis were captured in WCAP-15398, *Carolina Power and Light Harris Nuclear Plant Steam Generator Replacement/Uprating Analysis and Licensing Project NSSS Licensing Report* in Section 5.5.3, Class 1 Auxiliary Lines. WCAP-15398 was included as Enclosure 8 to HNP letter Serial: HNP-00-142, *Shearon Harris Nuclear Power Plant, Steam Generator Replacement, Technical Specification Amendment Application, Docket NO. 50-400, License No. NPF-63*, October 4, 2000.

As part of the license renewal effort, WCAP-16376-P, *Evaluation of Pressurizer Insurge/Outsurge Transients for Harris Nuclear Plant* was prepared. This report describes the development of plant-specific pressurizer insurge/outsurge and surge line stratification transients, and the evaluation of fatigue usage for the pressurizer lower head and surge line RCS hot leg nozzle for 60 years of predicted plant operation. Finally, the 60-year surge line environmental fatigue usage was determined in WCAP-16353-P, *Harris Nuclear Plant Fatigue Evaluation for License Renewal.*

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Question No: LRA 4.3-7

NRC Request:
In Section 4.3.1 of the LRA (page 4.3-2), there is a discussion of partial cycle counting using the methodology described in ANSI B31.1 Power Piping Code, 1967 Edition, Section 102.3.2.

- Is this methodology applicable to Class 1 piping?
- Is this methodology applicable to Class 1 components other than piping?
- Where was this methodology used?

HNP Response:
1. The methodology is not applicable to ASME Section III Class 1 piping. In the response to LRA 4.3.1-3, HNP provided a technical evaluation comparing the results from an independent ASME Section III, Division I, Subsection NB fatigue evaluation against the application of the ANSI B31.1 cycle reduction methodology to show that it was reasonable to use in the specific case of the charging nozzles.
2. The applicability of this methodology to other than piping components was not investigated, as it was not used for components other than piping.
3. This methodology was used to determine equivalent numbers of full design transient cycles on the charging nozzles.

See response to LRA 4.3.1-3 for supporting information as to the appropriateness of using this methodology for ASME Section III Class 1 piping.

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Question No: LRA 4.3-8

NRC Request:
For the evaluations performed for Class 1 components in Section 4.3 of the LRA, please provide the following information:

What is the original Code of Record? If so, what is the latest edition?

HNP Response:
Original Codes of Record are provided in the HNP FSAR Table 5.2.1-1 (below).

The latest edition that was used for Class 1 equipment analysis is taken from WCAP-15399, "Carolina Power and Light Harris Nuclear Plant Steam Generator Replacement/Uprating Analysis and Licensing Project, NSSS Licensing Report," September 2000 (Westinghouse Non-Proprietary Class 3)

SHNPP FSAR, TABLE 5.2.1-1 APPLICABLE CODE ADDENDA FOR RCPB COMPONENTS			WCAP-15399
Component	Required by 10CFR 50.55a	Designed and Fabricated	
Reactor vessel	Summer 1972	Winter 1971	1971 Edition with Addenda through Winter 1971
Full Length CRDM	Summer 1972	Summer 1974	1974 Edition with Addenda through Summer 1974
CRDM Head Adapter plugs	Summer 1972	Winter 1976	1974 Edition with Addenda through Winter 1976
Reactor Coolant Pump	Winter 1972	Summer 1972	1971 Edition with Addenda through Summer 1972

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SHNPP FSAR, TABLE 5.2.1-1 APPLICABLE CODE ADDENDA FOR RCPB COMPONENTS			WCAP-15399
Component	Required by 10CFR 50.55a	Designed and Fabricated	
Replacement Steam Generator	Summer 1972	Summer 1972 (Design)* 1986 Edition (Fabrication)	1971 Edition, Summer 1972 Addendum, Material strength from Summer 1972 Addendum to 1971 Edition. Where not available, material strength from 1986 Edition
Pressurizer	Summer 1972	Summer 1972	1971 Edition through Summer 1972 Addenda
Reactor coolant loop pipe	Winter 1972	Summer 1973	Section III (NB) through the Winter 1979 Addenda
Connecting Systems piping	Winter 1972	Summer 1973	(Class 1 auxiliary Lines) Subsection NB and NC 1977 Edition and addenda through Summer 1979 **

* 1986 Code Edition is applicable for materials not available in the Summer 1972 Code.

** Note: S-N curves from the 1986 Code Edition were used in the surge line stratification analysis.

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Question No: LRA 4.3-9

NRC Request:
In Section 4.3.1.5 (page 4.3-7), there is a discussion of "replacement schedule" for certain secondary side steam generator bolting. Has this bolting ever been replaced? Note 1 of Table 4.3-2 also needs to be clarified.

HNP Response:

- Has the bolting ever been replaced?

  Bolting for the steam generators has been replaced but not in accordance with any replacement schedule that would be associated with fatigue. Stud replacements were performed in 2001, 2003, and 2004 on secondary manways.

- Note 1 of Table 4.3-2 needs to be clarified.

  Note 1 of Table 4.3-2 will be modified as follows (additional text is shown in italics):

  Due to the original design usage factors exceeding 1.0, these bolts were originally *to be* replaced based on a replacement schedule, however, these fatigue usage values have been superseded by the results of the license renewal fatigue evaluation described in Subsection 4.3.1.5 *and a replacement schedule is no longer required.*

As stated in the response to B.3.1-RH-01, the updated evaluation changed only the number of Unit Loading and Unit Unloading transient cycles relative to the previous design analysis. The HNP Fatigue Management Program will be enhanced to track these cycles with a limit of 2000 cycles and an alarm limit of 1500 cycles.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-10

NRC Request:
In Section 4.3.1.6 (page 4.3-9) of the LRA, please explain whether design cycles or projected cycles were used in the 60-year fatigue evaluation.

HNP Response:
(This response is the same as Item Number 2 in LRA 4.3-5) Section 4.3.1.6 of the HNP LRA provides the evaluation of fatigue-related pressurizer TLAAs. In general, the projected cycles were not used for design fatigue calculations. There are two exceptions; however, the Pressurizer lower head and pressurizer surge line were re-qualified based on the projected cycles. An online stress and fatigue cycle monitoring system has been installed to maintain continuous surveillance of the operational fatigue damage accumulation rate.

The general deletion of cycle projections as a qualification basis for fatigue-related TLAAs is addressed in the response to LRA 4.3-2. Also, refer to the response to LRA 4.3.3-1.

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Question No: LRA 4.3-11

NRC Request:
In Section 4.3.1.6 (page 4.3-9), there is a discussion of the 60-year fatigue evaluation for the pressurizer components including the surge line.
• Has the RCS piping specification ever been updated and re-certified?

HNP Response:
The D-Spec, *Piping Design Specification, ANS Safety Class 1 - RCS, SIS, RHRS, CVCS, RVHVS, RVLIS*, was updated to Revision 3 in 1986 and re-certified at that time. Revision 3 eliminated AIB (arbitrary intermediate breaks) and added LBB (leak-before-break) criteria, allowed use of PVRC (Pressure Vessel Research Committee) damping, allowed use of Westinghouse SSDC 1.3X thermal transients, and updated allowable acceleration for valves.

Per WCAP-15398, *Carolina Power and Light, Harris Nuclear Plant, Steam Generator Replacement/Uprating Analysis and Licensing Project NSSS Licensing Report* (Submitted under HNP Letter Serial HNP-00-142, October 4, 2000, Accession Number ML003758761), Westinghouse drew from previous experience performing the V.C. Summer Steam Generator Replacement program, in which Westinghouse Model Δ75 replacement steam generators were installed, and the J. M. Farley Uprate program. The engineering and licensing reports produced for the Farley Uprate program were used as guides for preparing the HNP engineering and licensing documentation. In Section 3.1.1 of this report it states that:

> As part of the original design and analyses of the NSSS components for the HNP, NSSS design transients (i.e., temperature and pressure transients) were specified (Reference 1) for use in the analyses of the cyclic behavior of the NSSS components.

References

1. Systems Standard Design Criteria (SSDC) 1.3, Revision 2, April 15, 1974.

In the specific discussion of the pressurizer components and surge line, a chronology of the qualifications performed is included in the response to question LRA 4.3-6.

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Question No: LRA 4.3-12

NRC Request:
In Sections 4.3.2.1 and 4.3.2.2 of the LRA, please clarify why the Class 1 piping thermal transients are applicable to Class 2 and 3 and B31.1 piping. Please also justify the text in the analysis sections of page 4.3-11, 4.3-12, and 4.3-13.

HNP Response:
Background

The piping specification associated with Class 2 and 3 and B31.1 piping states:

> Stress analysis of piping shall be performed in accordance with ASME Section III using a stress range reduction factor (f) equal to 1.0.

The basis for the use of a stress range reduction factor (f) equal to 1.0 was provided to the NRC in the response to NRC Question 210.67 (Draft SER Open Item No. 354) in HNP Letter Serial LAP-83-429, dated September 19, 1983. The response states:

> The use of f = 1.0 is justified by the fact that the total number of full temperature cycles over 40 years during which the various system are expected to be in service is less than 7,000 cycles. This applies to *any* system on Shearon Harris Project.

A copy of the piping specification was provided for NRC review in response to NRC Question 210.80 in HNP Letter Serial NLS-85-338, dated September 26, 1985.

Supplement 3 to NUREG-1038, Safety Evaluation Report related to the Operation of Shearon Harris Nuclear Plant, Unit 1, dated May, 1986 states:

> ...the staff has concluded that the applicants' design specifications, design reports, and calculation files comply with ASME Code requirements and that adequate traceability exists in these documents relative to design commitments in

the Final Safety Analysis Report (FSAR). Therefore, the staff considers Confirmatory Issue 4 resolved.

- Class 2 and 3 piping

  The affected Class 2 and 3 piping is effectively an extension of the adjacent Class 1 piping. Therefore, the cycle count depends closely on reactor operating cycles and can be estimated by a review of the limiting reactor coolant system design transients in FSAR Table 3.9.1-1. Of those Normal Conditions listed that are likely to produce full-range thermal cycles in a 40-year plant lifetime are the 200 Heatup and Cooldown cycles. Assuming that all Upset Conditions lead to full-range thermal cycles adds an additional 980 cycles for a total of 1180 occurrences. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 1,770. This is only a fraction of the 7,000 full-range thermal cycles associated with a stress range reduction factor of 1.0. Therefore, the analysis for Class 2 and 3 piping has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii). A License Renewal Application amendment is required to revise the Section 4.3.2.1 of the LRA.

- B31.1 piping

  Main Feedwater System (and associated systems such as the condensate system) and Main Steam System (and associated systems such as the Steam Generator System) thermal cycles anticipated correspond to Heatup and Cooldown cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 300. Therefore, main feedwater and main steam components will not experience 7,000 cycles during the period of extended operation.

  The Auxiliary Feedwater System supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. The system provides an alternate to the Feedwater System during startup, hot standby and cooldown and also functions as an Engineered Safeguards System. The Auxiliary Feedwater System is directly relied upon to prevent core damage during plant transients resulting from a loss of normal feedwater flow, steam line rupture, main feedwater line rupture, loss of coolant accident (LOCA) and/or loss of off-site power by providing feedwater to the unaffected steam generators to maintain their inherent heat sink capability. The total number of cycles expected in 40 years of operation are as follows: 200 Heatup and Cooldown cycles, 2,000

cycles of feedwater cycling at hot standby, 980 cycles associated with all Upset Conditions, 240 cycles of quarterly AFW pump tests in accordance with ASME Section XI and 40 cycles of tests per the plant Technical Specifications. This yields a total of 3,460 cycles. For the 60-year extended operating period, the number of full-range thermal cycles for these piping analyses would be proportionally increased to 5,190. Therefore, auxiliary feedwater components will not experience 7,000 cycles during the period of extended operation.

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

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

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

Therefore, the analysis for B31.1 piping has been projected to the period of extended operation using 10 CFR 54.21(c)(1) method (ii). An amendment is required to revise the Section 4.3.2.2 of the LRA.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-13

NRC Request:
In Section 4.3.5.2 (page 4.3-19, Analyses) of the LRA, it states that the total number of cycles will be extrapolated to 60 years and 100 years.

Why were the cycles extrapolated to 100 years?

HNP Response:
The discussion of cycle extrapolation was added to show the conservative nature of the evaluation. The extraneous text related to 100 years will be removed.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-14

NRC Request:
In Table 4.3-2 (page 4.3-25) of the LRA, why do List Numbers 80 and 82 (Cold Leg Safety Injection and Hot Leg Safety Injection for recirculation only) have exactly the same fatigue usage factor?.

HNP Response:
The design fatigue evaluations of the Hot Leg safety injection nozzles were performed conservatively with an enveloping evaluation using the Cold Leg safety injection nozzles analysis. Therefore, the reported design usage values are the same.

Safety injection occurs in three types of nozzles for HNP: Safety Injection Cold Leg (SI CL), Safety Injection Hot Leg (SI HL), and Accumulator Injection to Cold Leg. There is one of each type on each of three loops. The summary of design fatigue usage indicates that the SI CL and SI HL nozzles all report design usage of 0.7, while the Accumulator injection nozzles report 0.45. Review of other documentation reveals that the SI HL nozzles were qualified by using the SI CL nozzle transients and generic basis, since the injection transients conservatively enveloped the additional RHR return transient specified for the HNP SI HL piping. Based on these observations, the controlling SI nozzle location for evaluation of environmental effects is actually the SI CL nozzle. Elements of the SI HL nozzle thermal effects are also taken into consideration during the Fen evaluation, to assure that any potential differences of the nozzle on the hot leg (e.g., higher loop temperature and loop transient differences) are covered in the evaluation of the SI CL nozzle.

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Question No: LRA 4.3-15

NRC Request:
In Table 4.3-3 (page 4.3-26) of the LRA, please provide the following clarifying information:

- The charging nozzle CUF is 0.89. Which nozzle is this when compared to the information provided in Table 4.3-2?
- The safety injection nozzle CUF is 0.93. Which nozzle is this when compared to the information provided in Table 4.3-2?

HNP Response:
1. The charging nozzle CUF of 0.89 is the result of an enveloping evaluation of both the normal and alternate charging nozzles.
2. The limiting safety injection nozzle evaluated was the 6 inch Cold Leg safety injection nozzle.

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Question No: LRA 4.3-16

NRC Request:
Table 4.3-1 (pages 4.3-21 and 4.3-22) provides 60-year NSSS Transient Cycle Projections. Please provide the following information:

- Explain why the cycles to-date and the 60-year projected cycles can be zero.
- It appears that some transients may not be specifically applicable to Shearon Harris. Please explain why these cycles have been added to this table (Normal Transients 13 and 14 and Upset Transient 8)
- How many cycles are associated with the 5 OBEs (Upset Transient 13)?
- For the Test Condition, please explain where pre-operational testing cycles were considered.

HNP Response:
The general deletion of cycle projections as a qualification basis for fatigue-related TLAAs is addressed in the response to LRA 4.3-2.

1. The cycle projections will be removed from the License Renewal Application. Cycle projections will not be used to justify acceptability of fatigue-related TLAAs by 10 CFR 54.21(c)(1)(i) - the analyses remain valid for the period of extended operation.

   Normal Transients 5, 11, and 12, "Step Load Increase of 10% of Full Power", "Unit Loading from 0% to 15% Power", and "Unit Unloading from 15% to 0% Power": Cycles to-date are zero, based on the analysis of a 5.26-year sample of Plant Instrumentation (PI) data. Normal Transients 11 and 12 exist to address low power feedwater operations. Feedwater operations during the 0% to 15% power range are assumed to transition from auxiliary feedwater to main feedwater. Additionally, main feedwater is assumed to start at 32°F during the power increase and cool to 32°F during the power decrease. At HNP, all feedwater operations at low power conditions that could potentially affect the primary systems and equipment are being tracked through HNP's cycle counting program. This tracking provides a much better understanding of the transients than simply monitoring the numbers of occurrences of the 0% to 15% transients. The HNP cycle counting program is used to track the following related events:

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Feedwater Cycling At Hot Standby, 2000 cycles with an alarm limit of 1500.

Main Feedwater Nozzle Temperature - Plant Loading Between 0 & 15% Power With Feedwater < 100°F, 60 cycles with an alarm limit of 42.

Main Feedwater Nozzle Temperature - Plant Loading Between 0 & 15%, 180 cycles with an alarm limit of 126.

Auxiliary Feedwater Nozzle Temperature And Flow Cycle, 2000 cycles with an alarm limit of 1500.

Progress Energy operates HNP as a base-load generator (i.e., HNP is not a "load following" plant). Normal Transient 5, while not precluded, the accumulation rate is considered very low in comparison to the allowed design cycles.

Normal Transients 8, 9, and 15, "Steady State Fluctuations (Initial)", "Steady State Fluctuations (Random)", and "Boron Concentration Equalization", are not counted. Prior to plant operation and the establishment of plant cycle counting procedure, it was concluded that the design limits would never be reached, based on the expected number of cycles. The temperature changes associated with Steady State Fluctuations (Initial)" and "Steady State Fluctuations (Random)" are small (≤ 3°F). Therefore, these transients are not contributors to CUF. "Boron Concentration Equalization" transients are associated with the operation of a "load following" plant. Progress Energy operates HNP as a base-load generator (i.e., HNP is not a "load following" plant). Normal Transient 15, while not precluded, the accumulation rate is considered very low in comparison to the allowed design cycles. The "Cycles To-Date" for these entries should be changed to "Not Counted".

Normal Transient 16, "Turbine Roll Test": During the development of one of the basis documents, Operations personnel were queried specifically as to how many Turbine Roll Tests had been performed. As part of the inquiry, the NSSS vendor provided details concerning the transient as follows:

The transient is assumed for turbine cycle checkout. The assumption is that RCP power is used to heat the primary system to normal operating pressure and temperature (no load conditions). The steam generated is

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used to perform a turbine roll test. The NSSS is assumed to cool down with a rate greater than 100°F /hr during the test. The total cooldown is approximately 110°F down from the no load temperature. The test is assumed to occur 20 times over the plant life.

Operations stated that HNP performed one Turbine Roll Test in 1986 during initial construction, and none since that time.

Upset Transient 1, "Loss of Load" has been is counted since the start of plant operations in accordance with the plant cycle counting procedure. The number of occurrences has been zero.

2. Normal Transients 13, 14, and Upset Transient 8 were included in the qualifications performed by WCAP-14778, Revision 1, "Carolina Power and Light Harris Nuclear Plant Steam Generator Replacement/Uprating Analysis and Licensing Project NSSS Engineering Report", September 2000. As noted in the license renewal basis document, "Normal Condition" transients 13 and 14 ("Loop Out of Service") are not applicable to the current HNP license. HNP is not currently licensed to operate with N-1 loops. The "Loop Out of Service" transients were included in the Westinghouse System Standard Design Criteria (SSDC 1.3, Rev. 2) so that the components are designed in case the plant is licensed to operate with N-1 loops. It was recommended by Westinghouse that the "Loop Out of Service" transients continue to be considered for the SGR/Uprating Project. Therefore, the transients were carried forward to the License Renewal fatigue evaluation. This also applies to "Upset" Transient 8 ("Inadvertent Startup of an Inactive Loop").

3. 10 cycles for each global OBE cycle as specified in the FSAR.

4. As defined in the HNP FSAR: The "Test Condition" transients include "Leak (Leakage) Tests" and "Hydrostatic Tests". The leakage tests are applicable during life of plant. Secondary side hydrostatic tests were to have been performed prior to plant startup or subsequently following shutdown for major repairs, or both. "Test Condition" transients 1, 2 and 5 are defined in FSAR 3.9.1.1.5 and "Test Condition" transients 3 and 4 are defined in FSAR 3.9.1.1.1.

   FSAR 3.9.1.1.5 states that primary side hydrostatic tests include both shop and field hydrostatic tests. These tests can occur as a result of component or system testing. FSAR 3.9.1.1.5 also states that four additional hydrostatic

tests, in accordance with ASME Section XI inservice inspection (ISI) requirements, are expected over the lifetime of the plant. Since four additional tests in accordance with ASME Section XI ISI requirements were expected to be performed, and since one of these tests was performed and recorded, then no more than six shop tests were possible. Therefore, the total number of primary side hydrostatic can conservatively be no more than seven.

A License Renewal Application amendment is required.

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Question No: LRA 4.3-17

NRC Request:
Please explain how stresses are input to apply the stress transfer function of WESTEMS™, from the 6 stress components or the stress intensity. Please provide input and results of any benchmarking problems for pressure, temperature, or moment loadings.

HNP Response:
WESTEMS™ uses the transfer function method (TFM) [1] to calculate six (6) components of stresses due to time varying mechanical and thermal loads. Time varying component stresses are calculated through wall as a function of the time varying mechanical and thermal boundary conditions. The resulting through wall stress components are processed and categorized according ASME Section III, Division 1, Subsection NB criteria. The processing first involves the calculation and categorization of the membrane and bending and peak components mechanical and thermal stresses. These calculations are performed at the component stress levels, for each time step and for each applied loading type. The resulting stresses are then added to form the total stress and primary plus secondary stress according to ASME rules. Stress peak selection for fatigue evaluation purposes is based on analysis of the total stress time history and of the primary plus secondary stress time history. Both the total stress and primary plus secondary stress are retained for future consideration in online fatigue evaluations. The discussion below will help to clarify the transfer function methodology, the transfer function database role, and provide an example of the current benchmarking process.

The transfer function method is a mathematical device that is capable of quantifying the effects experienced by a system due to an external disturbance, or excitation, with the aid of a characteristics function known as transfer function. In essence, the transfer function method is a means that correlates time-dependent behavior, in terms of input and output, of a system as seen in the thermal and dynamical problems. Examples of "disturbance" are mechanical forces, and thermal transients, etc. Examples of "effects" include stresses, strains, displacements, and temperature, etc. For typical structural applications, the "disturbance" can be surface temperature changes T(t), pressure P variation, forces (Fx, Fy, Fz), and moments (Mx, My, Mz) in a structural body (in vector notations: $\vec{F}$, and $\vec{M}$), whereas the typical "effects" mostly refers to the stresses, displacements and metal interior temperatures.

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In WESTEMS™, the transfer function methodology uses 2 or more unit load databases that have 4 or 6 components of stress depending on the nature of the original finite element model method that was used. If a two dimensional finite element model was used to create the transfer function database then 4 components of stress are capable (Sxx, Syy, Szz, Sxy). If a three dimensional model was used then there are 6 components of stress in the transfer function databases (Sxx, Syy, Szz, Sxy, Syz, and Szx). The total number of stress states in the transfer function databases is dependent on the complexity of the thermal and mechanical boundary conditions being simulated.

For thermal applications, the transfer function is a characteristics function of a thermal-mechanical system. The characteristics include geometry, boundary conditions, insulation conditions, material properties, and thermal zones. These characteristics are all built into the transfer function for a predefined thermal-mechanical system. Therefore, a transfer function database is fixed for a particular type of thermal-mechanical problem. However, a single set of transfer function database can be used to evaluate the system responses caused by any kind of transients. This means that transfer function database is created only once but can be used to obtain solutions for unlimited numbers of transient cases.

It is important to realize that thermal stresses in materials or any structural systems arisen from temperature transients are evolving because heat transfer is an energy transport process that will continue until thermal equilibrium is established. This means that it requires appreciable amount of time for a thermally disturbed material or structural system to come to a steady state even if the disturbance is as brief as an impulse. In short, thermal transient is a time-dependent problem. On the contrary, all mechanical loads, pressure, direct forces, and moments, encountered in the general structural applications are treated as static problems unless the loading rates are so high that the dynamic effects can not be ignored. To appropriately reflect to the types of loads being dealt with, the databases are split into two types:

- Thermal transfer Function DataBase (TFDB)
- MEchanical transfer function DataBase (MEDB).

Westinghouse has validated the thermal stress capability of the WESTEMS™ transfer function method by performing identical analyses using the Westinghouse transfer function method and an independent finite element program like ANSYS or WECAN. Examples of the predicted stress components results for benchmarking the transfer function models are shown below. The benchmarking process is generally performed for every transfer function database created. The following example was taken directly from the appendix of a recent Westinghouse Transfer function database calculation

note. The verification of WESTEMS™ thermal and mechanical stress calculations have been performed in the programs verification and validation documentation. However, each application verification of the finite element models and of the final thermal transfer function databases should be performed in order to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results Westinghouse performs thermal stress analyses using both the finite element program and WESTEMS™. The example below shows these comparisons and results. Certain information has been removed and text has been modified in order to clarify the example.

Verification of the Surge Nozzle With Reducer TFDB and MEDB

Verification of the databases being used for the WESTEMS™ analyses is a required step to ensure good analysis results. All databases are herein examined through suitable benchmarking problems.

The database files, TFDB and MEDB, generated in the unit load finite element analyses represent the thermal and mechanical characteristics of the structural component considered. By using these databases, the stresses at the specified analysis sections (ASN or cut) can be evaluated for any combination of load conditions. To correctly produce the results, each load type requires an appropriate scaling factor which is being developed in the following subsections. The scaling factor provides a means to correct the effects arisen from differences on the stress units used in ANSYS and in WESTEMS™. It also is a means which permits non-standard unit loads to be used to generate the database.

Verifying the bending moment database - M_z

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_z represents bending about the global z-axis. According to Reference 1, the z-axis is perpendicular to the x and y axes. The analysis for this bending case was performed and documented in Reference 1. The applied moment is 1000 in-kips.

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Consider the well known bending stress equation

$\sigma = Mr/I$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-1, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

R_o = 5.250 in.

R_i = 6.375 in.

$I = \pi (R_o^4 - R_i^4) / 4 = 700.55 \text{ in}^4$

Comparison of the ANSYS FE and analytical results are shown in Table B-1. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two and the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $f_b = f_1 * f_2 = 10^{-6}$.

Table B-1: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	7494.11	7533.20	-0.52
Outside node	187	9099.99	9067.20	0.36

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Verifying the torsion database - M_y

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the torsion moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The moment M_y represents the moment about the global y-axis. The y-axis is in coincidence with the centerline of the nozzle. Moment M_y therefore represents twist of the nozzle. The analysis for this torsion case was performed and documented. The applied moment is 1000 in-kips.

Consider the well known torsion shearing stress equation

$$\tau = Mr/J$$

where M is the applied torque, r is distance from the neutral axis, and J is the polar moment of inertia in torsion of the cross sectional area.

Two nodes, as listed in Table B-2, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above torsion equation can be applied.

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

$J = \pi (R_o^4 - R_i^4) / 2 = 1401.1 \text{ in}^4$

Comparison of the ANSYS FE and analytical results are shown in Table B-2. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied

torque is in-kips whereas 1000 in-kips of torque was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two and the scaling factor for the torsion load to be used for WESTEMS™ analyses is found to be $f_t = f_1 * f_2 = 10^{-6}$.

Table B-2: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	3747.05	3747.10	0.00
Outside node	187	4550.00	4550.00	0.00

Verifying the pressure database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the pressure portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the pressure loading case was performed and documented. The applied pressure is 1000 psi.

Consider the well known hoop stress equation for a pressurized pipe

$$\sigma_\theta = \frac{p R_i^2}{R_o^2 - R_i^2}\left(1 + \frac{R_o^2}{r^2}\right)$$

where p is the internal pressure, R_o is the outside radius, R_i is the inside radius, and r is the radius at any point.

Two nodes, as listed in Table B-3, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above hoop stress equation can be applied.

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

Comparison of the ANSYS FE and analytical results are shown in Table B-3. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied pressure is psi whereas 1000 psi of pressure was used in the database creation, a second factor, f_2=0.001, is required. Combining the two factors and the scaling factor for the pressure load to be used for WESTEMS™ analyses is found to be $f_p = f_1 * f_2 = 10^{-6}$.

Table B-3: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	275	5215.05	5230.30	-0.29
Outside node	187	4215.05	4208.60	0.15

Verifying the axial database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the axial force portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the axial force loading case was performed and documented. The applied axial force is 1000 lb.

Consider the well known axial stress equation for a pipe

$$\sigma = \frac{F}{A}$$

where F is the applied force, and A is the cross sectional area.

Two nodes, as listed in Table B-4, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above axial stress equation can be applied.

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

$A = \pi (R_o^2 - R_i^2) = 41.09 \text{ in}^2$

Comparison of the ANSYS FE and analytical results are shown in Table B-4. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied force is 1 kip whereas 1 kip of force was used in the database creation, a second factor, f_2=1.0, is required. Combining the two factors and the scaling factor for the axial load to be used for WESTEMS™ analyses is found to be $f_a = f_1 * f_2 = 10^{-3}$.

Table B-4: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	24.34	24.38	0.18
Outside node	187	24.34	24.30	0.16

Verifying the thermal stress database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the transfer function thermal stress database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

To benchmark and verify this portion of the database and determine the appropriate scaling factor for the thermal loads, an arbitrary transient was used. The transient used for this benchmarking problem is defined in the data shown in Table B-6 and Figure B-1.

Table B-6: Temperature and Film Coefficient used for the Benchmarking.

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
0.001	550	550	550	0.007716	0.007716	0.007716
10	550	550	550	0.007716	0.007716	0.007716
12	250	250	250	0.007716	0.007716	0.007716
13	250	250	250	0.007716	0.007716	0.007716
14	250	250	250	0.007716	0.007716	0.007716
15	250	250	250	0.007716	0.007716	0.007716
16	250	250	250	0.007716	0.007716	0.007716
18	250	250	250	0.007716	0.007716	0.007716
20	250	250	250	0.007716	0.007716	0.007716
30	250	250	250	0.007716	0.007716	0.007716
40	250	250	250	0.007716	0.007716	0.007716
55	250	250	250	0.007716	0.007716	0.007716
70	250	250	250	0.007716	0.007716	0.007716
90	250	250	250	0.007716	0.007716	0.007716
110	250	250	250	0.007716	0.007716	0.007716
135	250	250	250	0.007716	0.007716	0.007716
160	250	250	250	0.007716	0.007716	0.007716
185	250	250	250	0.007716	0.007716	0.007716
210	250	250	250	0.007716	0.007716	0.007716

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Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
212	550	550	550	0.007716	0.007716	0.007716
213	550	550	550	0.007716	0.007716	0.007716
214	550	550	550	0.007716	0.007716	0.007716
215	550	550	550	0.007716	0.007716	0.007716
216	550	550	550	0.007716	0.007716	0.007716
217	550	550	550	0.007716	0.007716	0.007716
219	550	550	550	0.007716	0.007716	0.007716
221	550	550	550	0.007716	0.007716	0.007716
225	550	550	550	0.007716	0.007716	0.007716
230	550	550	550	0.007716	0.007716	0.007716
235	550	550	550	0.007716	0.007716	0.007716
250	550	550	550	0.007716	0.007716	0.007716
265	550	550	550	0.007716	0.007716	0.007716
280	550	550	550	0.007716	0.007716	0.007716
300	550	550	550	0.007716	0.007716	0.007716
320	550	550	550	0.007716	0.007716	0.007716
345	550	550	550	0.007716	0.007716	0.007716
370	550	550	550	0.007716	0.007716	0.007716
395	550	550	550	0.007716	0.007716	0.007716
410	550	550	550	0.007716	0.007716	0.007716
470	548	548	548	0.007716	0.007716	0.007716
530	546	546	546	0.007716	0.007716	0.007716
590	544	544	544	0.007716	0.007716	0.007716
650	542	542	542	0.007716	0.007716	0.007716
710	540	540	540	0.007716	0.007716	0.007716
770	538	538	538	0.007716	0.007716	0.007716
830	536	536	536	0.007716	0.007716	0.007716
890	534	534	534	0.007716	0.007716	0.007716
950	532	532	532	0.007716	0.007716	0.007716
1010	530	530	530	0.007716	0.007716	0.007716
1070	528	528	528	0.007716	0.007716	0.007716

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Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
1130	526	526	526	0.007716	0.007716	0.007716
1190	524	524	524	0.007716	0.007716	0.007716
1250	522	522	522	0.007716	0.007716	0.007716
1310	520	520	520	0.007716	0.007716	0.007716
1370	518	518	518	0.007716	0.007716	0.007716
1430	516	516	516	0.007716	0.007716	0.007716
1490	514	514	514	0.007716	0.007716	0.007716
1550	512	512	512	0.007716	0.007716	0.007716
1610	510	510	510	0.007716	0.007716	0.007716
1670	508	508	508	0.007716	0.007716	0.007716
1730	506	506	506	0.007716	0.007716	0.007716
1790	504	504	504	0.007716	0.007716	0.007716
1850	502	502	502	0.007716	0.007716	0.007716
1910	500	500	500	0.007716	0.007716	0.007716
1970	498	498	498	0.007716	0.007716	0.007716
2030	496	496	496	0.007716	0.007716	0.007716
2090	494	494	494	0.007716	0.007716	0.007716
2150	492	492	492	0.007716	0.007716	0.007716
2210	490	490	490	0.007716	0.007716	0.007716
2212	490	490	490	0.007716	0.007716	0.007716
2213	490	490	490	0.007716	0.007716	0.007716
2214	490	490	490	0.007716	0.007716	0.007716
2215	490	490	490	0.007716	0.007716	0.007716
2216	490	490	490	0.007716	0.007716	0.007716
2217	490	490	490	0.007716	0.007716	0.007716
2219	490	490	490	0.007716	0.007716	0.007716
2221	490	490	490	0.007716	0.007716	0.007716
2225	490	490	490	0.007716	0.007716	0.007716
2230	490	490	490	0.007716	0.007716	0.007716
2235	490	490	490	0.007716	0.007716	0.007716
2250	490	490	490	0.007716	0.007716	0.007716

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
2265	490	490	490	0.007716	0.007716	0.007716
2280	490	490	490	0.007716	0.007716	0.007716
2300	490	490	490	0.007716	0.007716	0.007716
2320	490	490	490	0.007716	0.007716	0.007716
2345	490	490	490	0.007716	0.007716	0.007716
2370	490	490	490	0.007716	0.007716	0.007716
2395	490	490	490	0.007716	0.007716	0.007716
2410	490	490	490	0.007716	0.007716	0.007716

In this benchmarking transient, Zone 1, Zone 2, and Zone 3 undergo a severe thermal shock. The transient considered is hypothetical but is intentionally made severe on the temperature rate so as to allow a vigorous examination of the integrity of the transfer function database. The transient is shown in the Figure B-1. This transient was analyzed by both WESTEMS™ and ANSYS. Note that the ANSYS results represent full finite element analyses whereas the WESTEMS™ results are produced by the transfer function method, which utilizes the transfer function databases produced by ANSYS.

The metal surface temperatures for all three zones were calculated using the 1D Simplified Stress Model (SSM) in WESTEMS. The input data for this part of the calculations are shown in Table B-7. The metal surface temperature solutions from the SSM evaluations are represented by tagnames tzone1m, tzone2m, and tzone3m, which are saved in the WESTEMS benchmark history file.

Table B-7: Simplified Stress Models for Metal Surface Temperature Calculations.

Component ID	Name	OutPut Tag	T ambi	Material_A	Temp Tag_A	HFilm Tag_A	Wall Thick_A	Inside Diameter_A	Num Nodes_A
200	PZR Surge Nozzle with Reducer Zone 1	tzone1m	70	3	tzone1	hzone1	1.1	11.3	0
201	PZR Surge	tzone2m	70	60	tzone2	hzone2	1.56	11.88	0

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Component ID	Name	OutPut Tag	T ambi	Material_A	Temp Tag_A	HFilm Tag_A	Wall Thick_A	Inside Diameter_A	Num Nodes_A
	Nozzle with Reducer Zone 2								
202	PZR Surge Nozzle with Reducer Zone 3	tzone3m	70	2	tzone3	hzone3	3.0	12.0	0

Note:

1) Material 3 is SA 182 F316 SS 1989

2) Material 60 is SB166 Alloy 600 (Rod) 1998

3) Material 2 is SA 216 Gr WCC 1989

The results, as shown in Figures B-2 through B-11 (units: stress ksi, time seconds), are then graphically compared on both the shapes and the magnitudes. It can be seen from these figures that the WESTEMS™ results compare very well with those calculated by ANSYS, both in magnitudes and curve shapes.

The shapes of the curves of the stresses from the WESTEMS™ analysis are visually compared with those from the ANSYS full finite element analysis. In general, good comparisons are observed for all cases.

Overall, very good benchmarking results have been achieved which assures good results can be produced through the TFDB created in Reference 1. In order to maintain a conservative answer a correction factor of 4% is applied, that is, f_1=1.04. Since the stress unit in the ANSYS FEA results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a factor, f_2=0.001, is required. Combining the two factors for the thermal load to be used for WESTEMS™ analyses is $f_T = f_1 * f_2$ = 0.00104, which is to be registered to the "TFDB_Factor" box in the WESTEMS™ ASN Analysis Models.

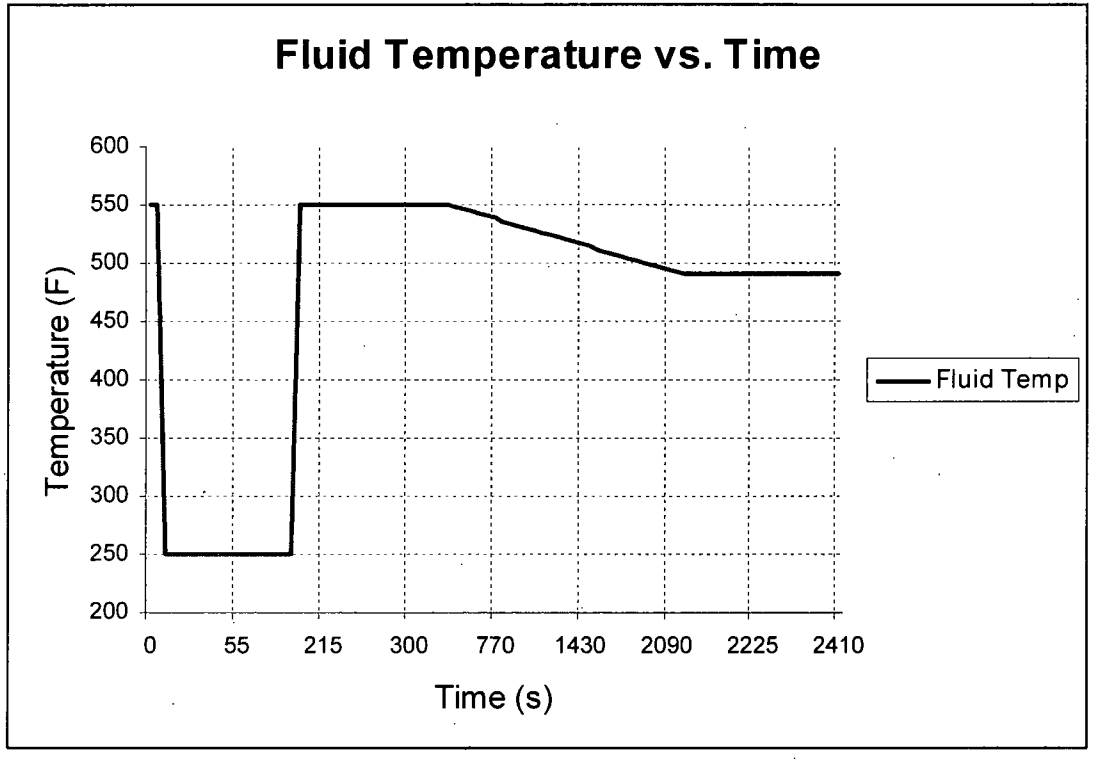

Figure B-1: Benchmark Transient.

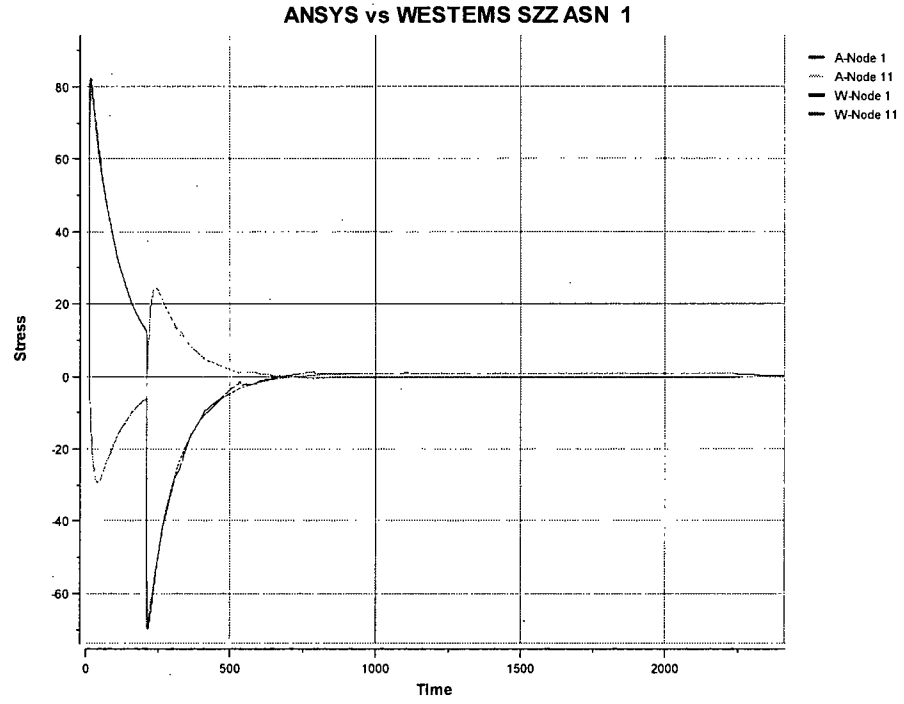

Figure B-2: ASN 1 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

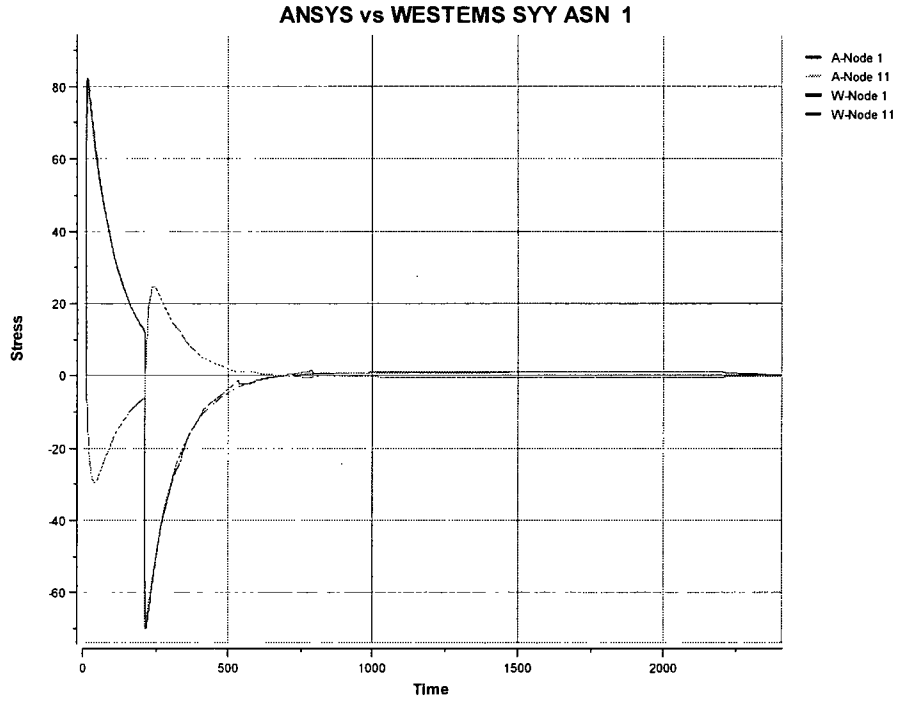

Figure B-3: ASN 1 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

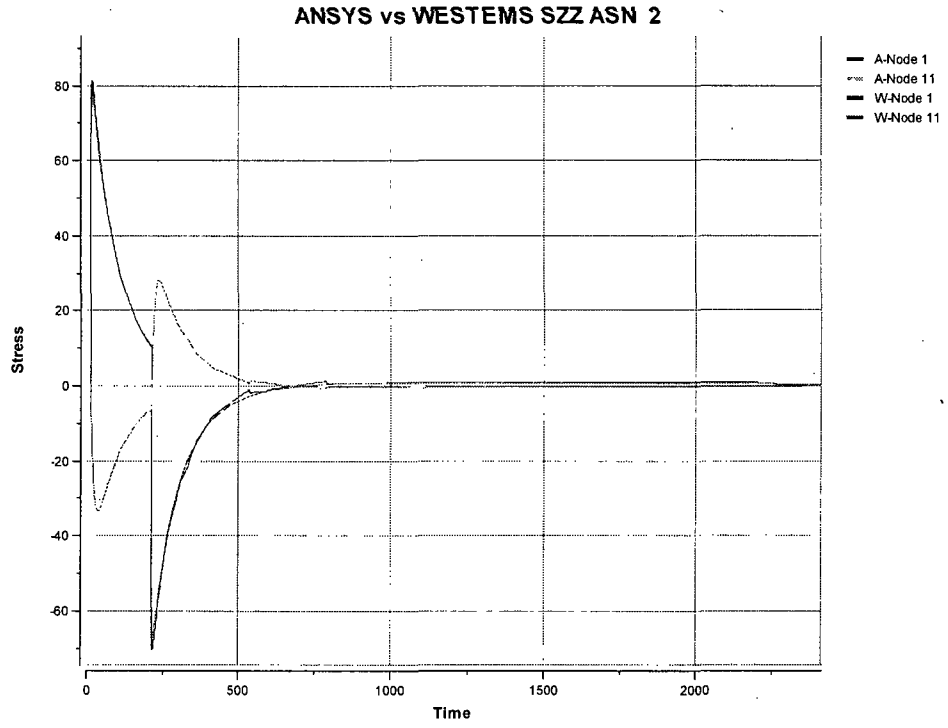

Figure B-4: ASN 2 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

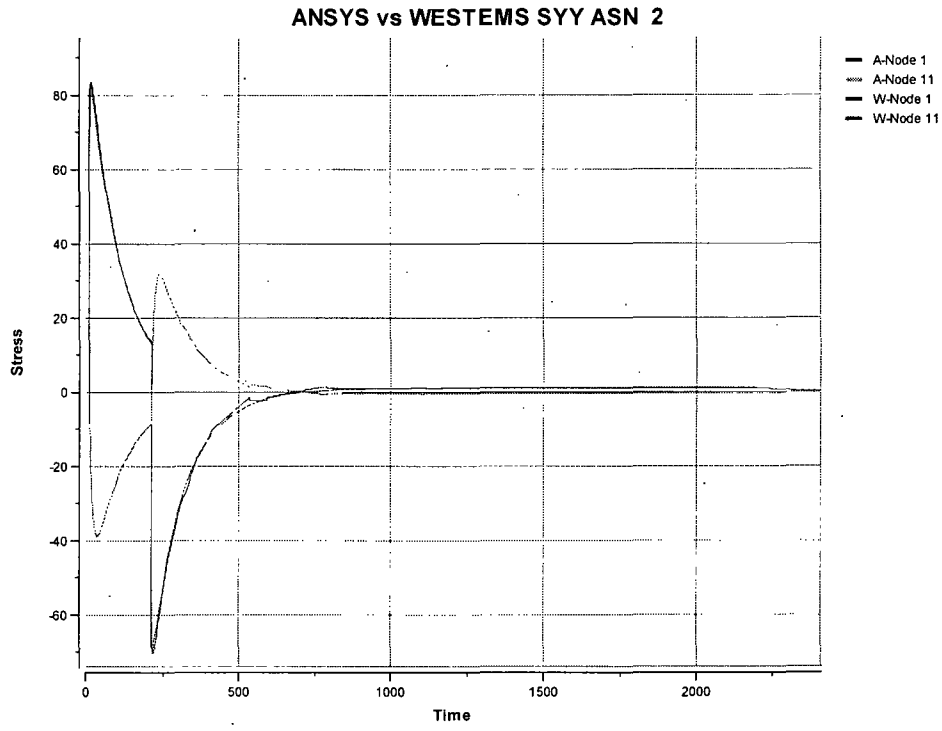

Figure B-5: ASN 2 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

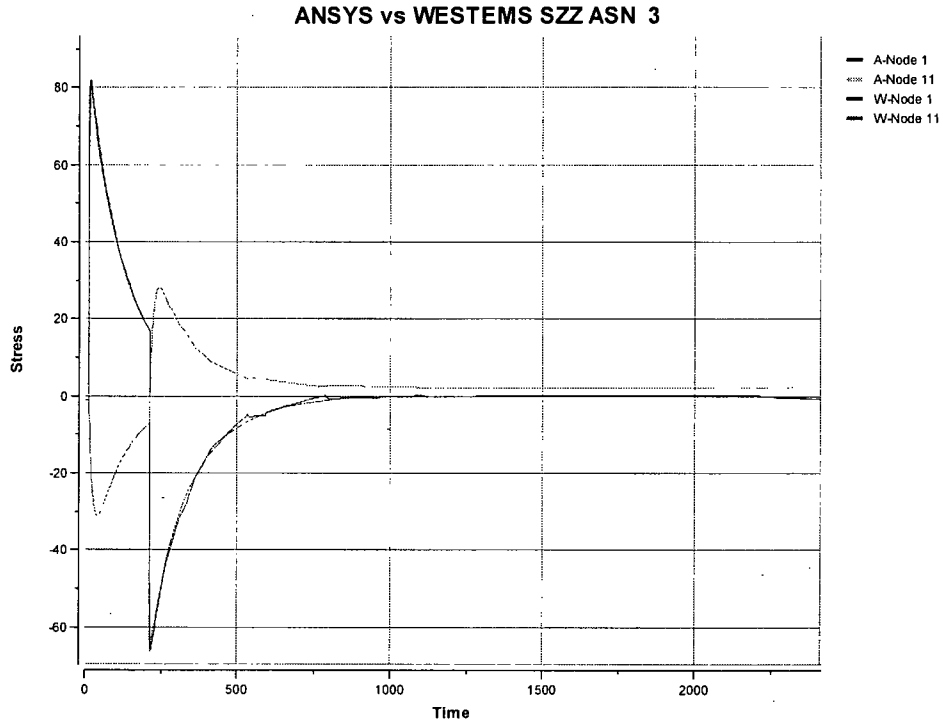

Figure B-6: ASN 3 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

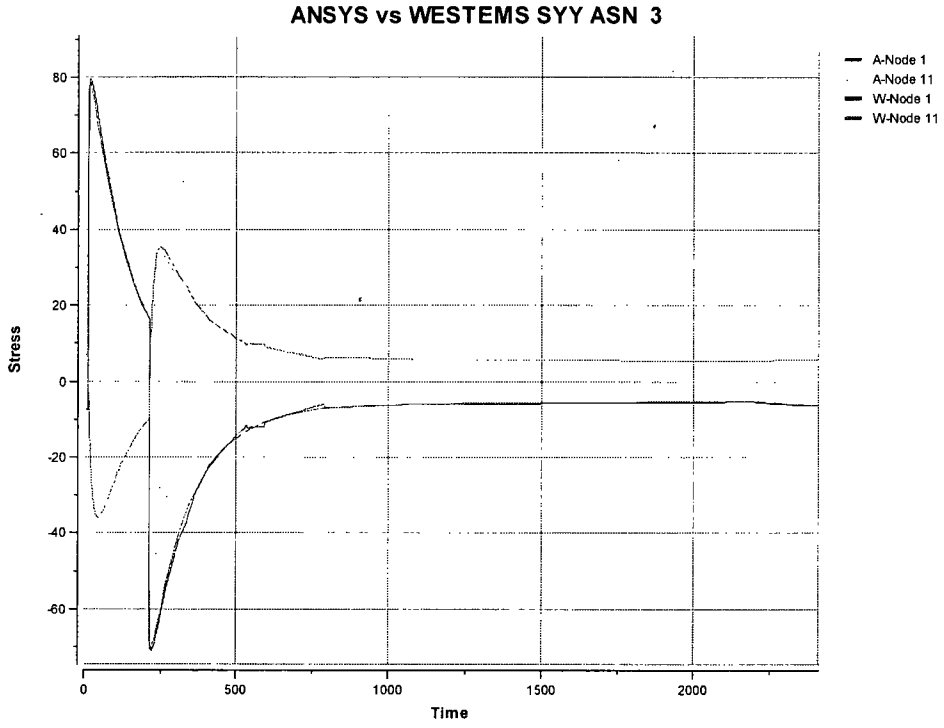

Figure B-7: ASN 3 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

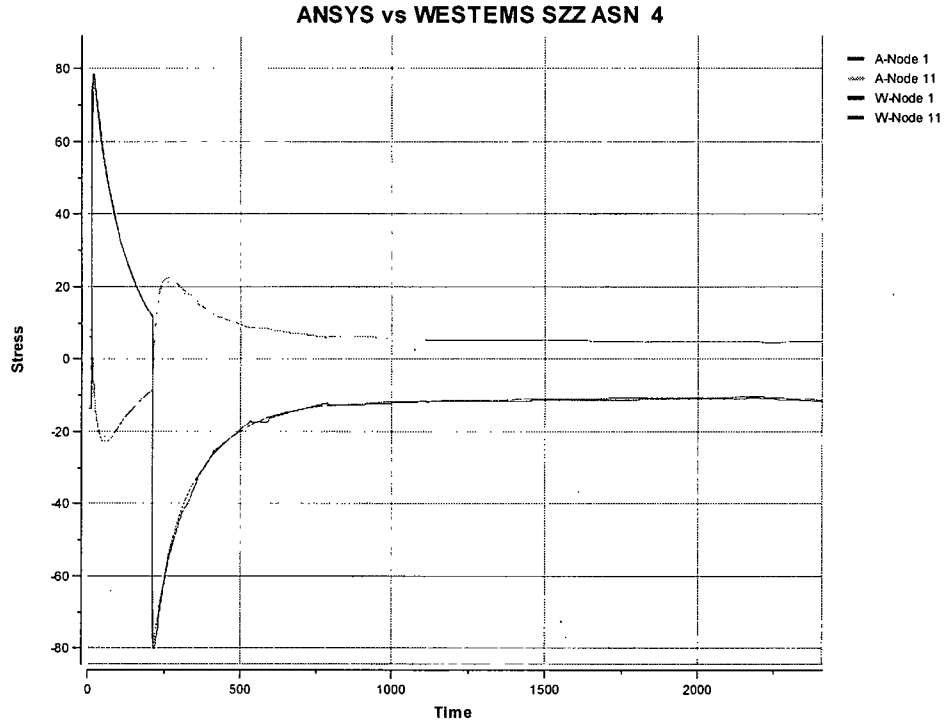

Figure B-8: ASN 4 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

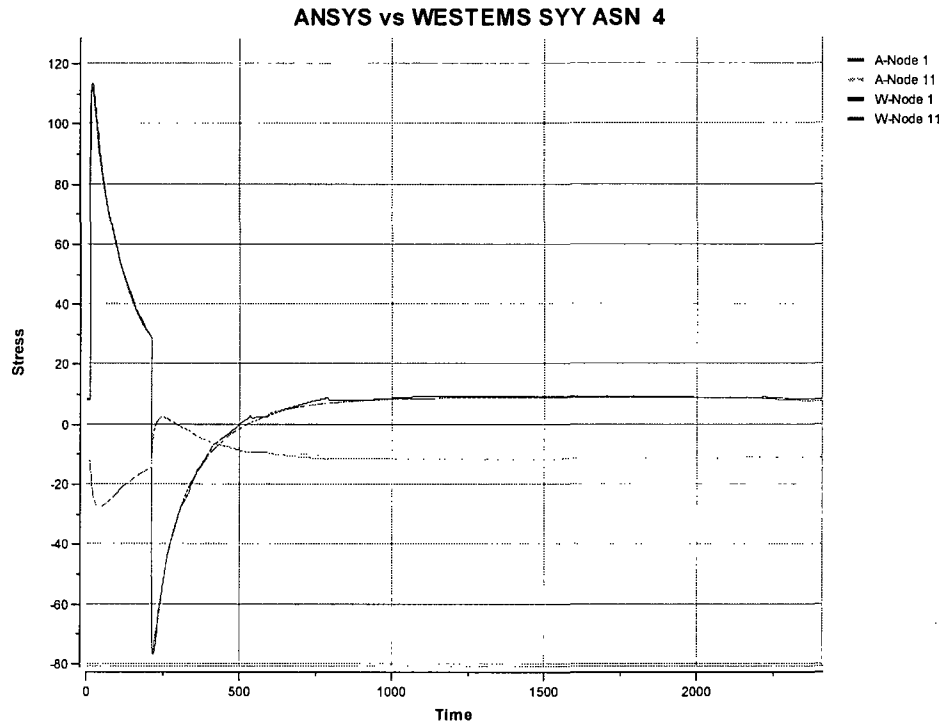

Figure B-9: ASN 4 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

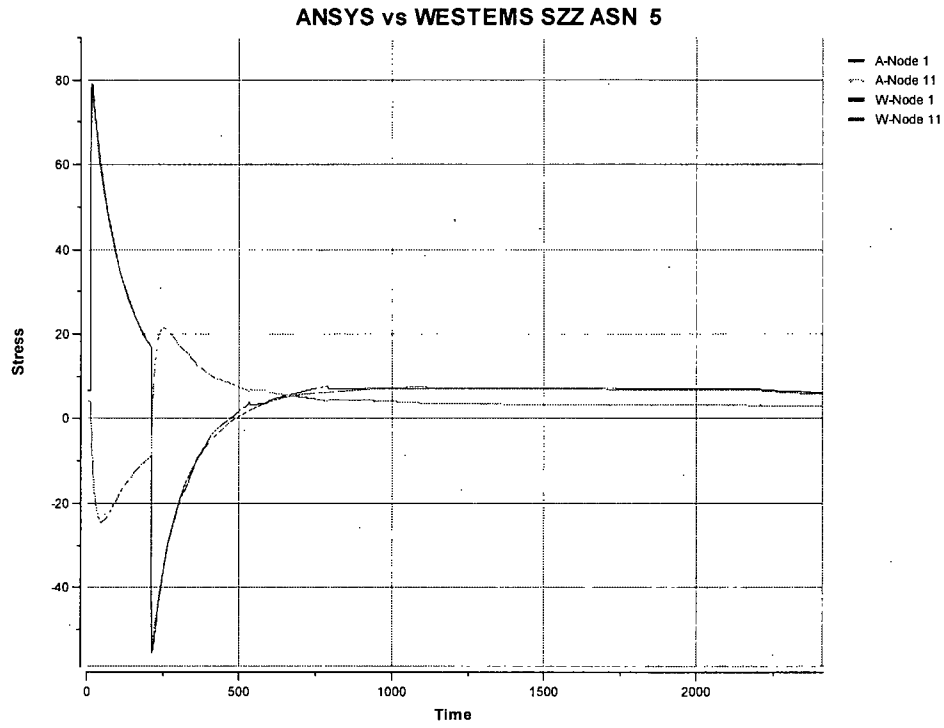

Figure B-10: ASN 5 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

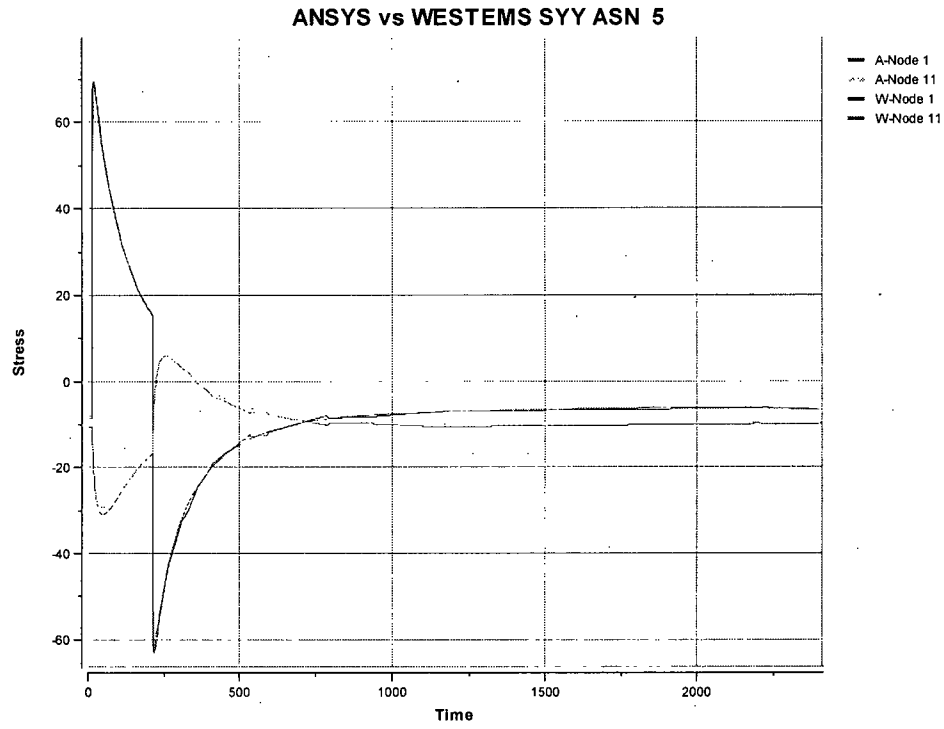

Figure B-11: ASN 5 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

The results shown below, obtained by a WESTEMS user in a Westinghouse European site, serve additional verification of the transfer function methodology..

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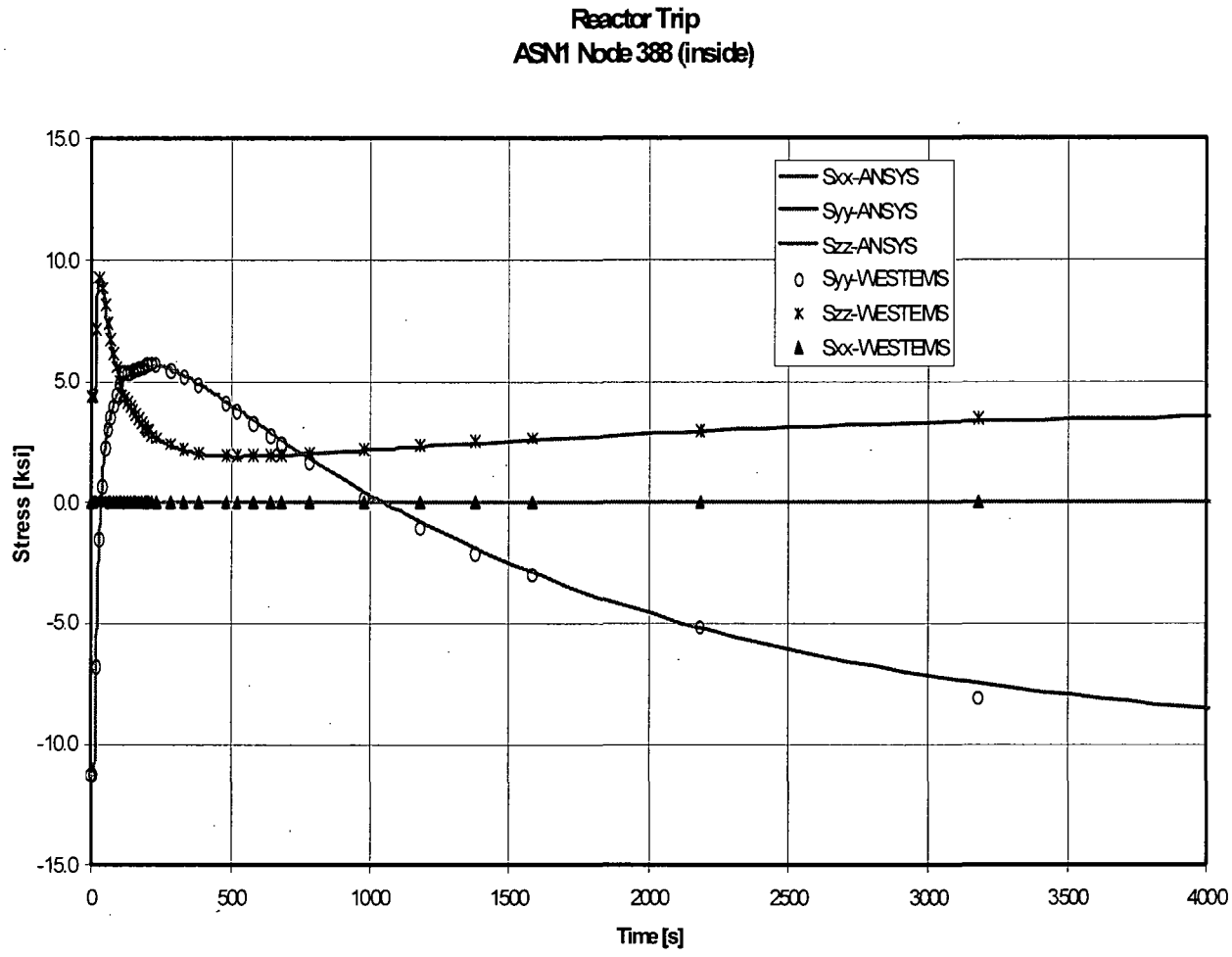

Additional Thermal Stress Benchmark Results, Sample 1.

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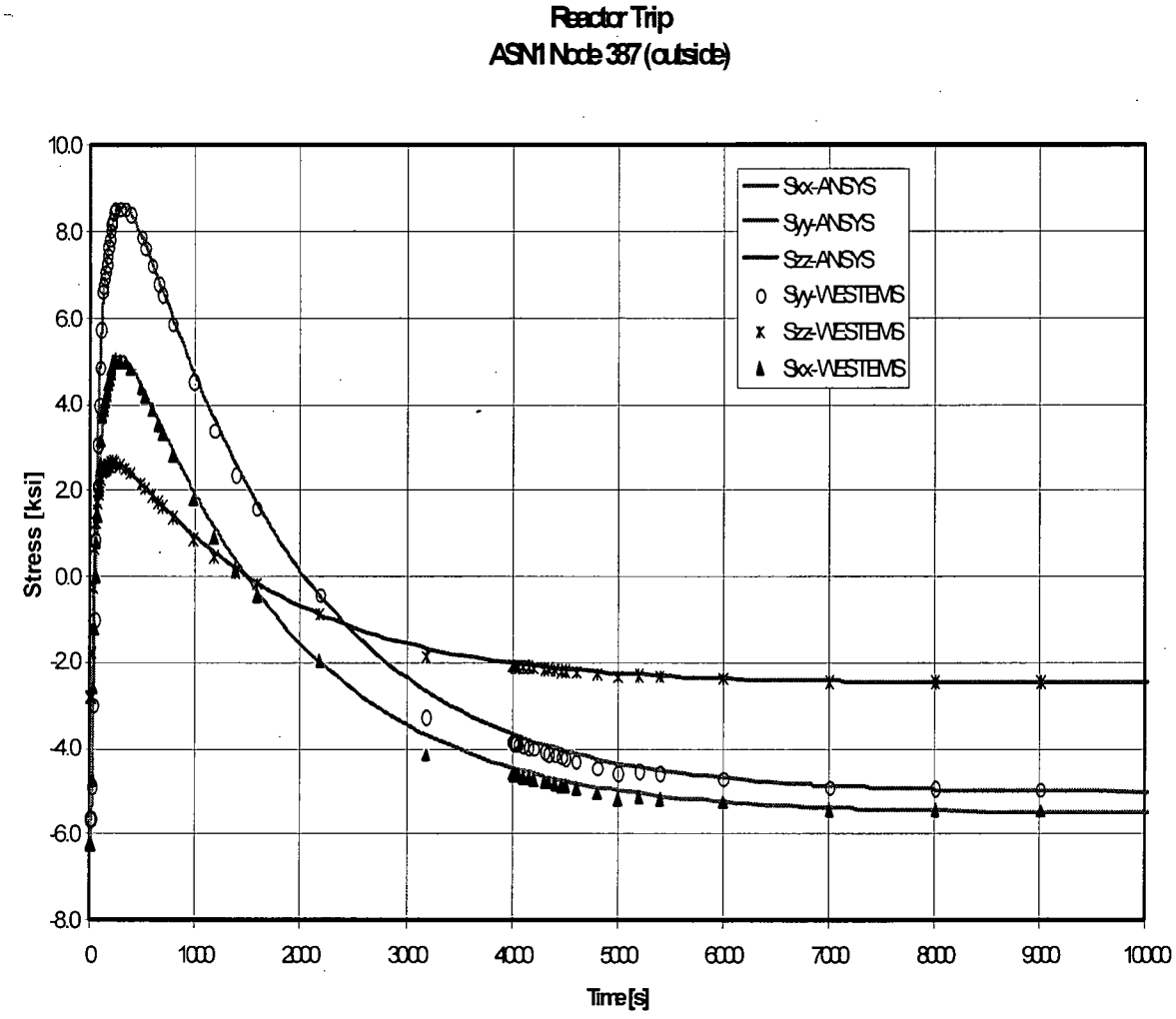

Additional Thermal Stress Benchmark Results, Sample 2.

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References:

1. "Transfer Function Method for thermal Stress and Fatigue Analysis: Technical Basis", WCAP-12315, Westinghouse Proprietary Class 2, C. Y. Yang, May 1990.

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Question No: LRA 4.3-18

NRC Request:
There are discrepancies in the Design CUF values between LRA and the basis document (WCAP-15398 Supplement 1, December 2001). For example, the CUF of main feedwater nozzle is 0.98 as shown in Table 5.7.1-2 of the WCAP-15398 (Supplement 1, which is different from the value (0.93) described in the LRA Table 4.3-2 (Design Fatigue Usage Factors). Please explain.

HNP Response:
The value of 0.93 in LRA Table 4.3-2 (List Number 31) is incorrect. It was taken from the original version of WCAP-15398 (September 2000). The correct value of 0.98 is from Supplement 1 of WCAP-15398. LRA Table 4.3-2 will be updated to incorporate the correct value.

A License Renewal Application amendment is required.

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Question No: LRA 4.6.1

NRC Request:
On page 4.6-2 of Section 4.6.1.1 of the LRA, it states that ILRT will be conservatively assumed to be performed every 5 years and this yields 12 cycles for 60 years. However, ILRT will be performed once every 10 years only after adopting 10 CFR 50, appendix J, Option B.

- When did HNP adopt Option B?
- necessary, provide a correction to the LRA based on your response.

HNP Response:
HNP adopted Option B in 1999. A letter from the NRC from Richard J. Laufer to James Scarola of Carolina Power and Light Company, dated September 17, 1999, Subject: Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment RE: Containment Integrated Leak Rate Testing (TAC No. MA5944), issued Amendment 91 to revise the Technical Specifications to incorporate performance-based 10CFR 50 Appendix J, Option B, for Type A containment integrated leakage rate testing. An ILRT has not been performed since issuance of Amendment 91. A preoperational Type A ILRT was performed February 25, 1986 and periodic Type A ILRTs were performed October 25, 1989, September 21, 1992 and May 23, 1997. The next ILRT (Option B Type A) is required no later than May 23, 2012 based on a one-time extension from once in 10 years to once in 15 years granted by the NRC in Amendment 122. This amendment was issued March 30, 2006 in a letter from the NRC from Chandu P. Patel to Cornelius J. Gannon of Carolina Power and Light Company, Subject: Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Containment Integrated Leak Rate Test (TAC No. MC6722).

The number of ILRTs performed to date is four counting the pre-operational test. The number of remaining ILRTs projected to the end of extended HNP plant life, from 2012 to 2046, on a 10-year interval is only an additional four, for a total of eight ILRTs. Therefore, for the remaining period of operation, including the license renewal period, a grand total of eight ILRTs is projected. The total of 12 ILRTs discussed in Section 4.6.1.1 of the LRA conservatively bounds the actual projected number of eight ILRTs. No change to the LRA is needed.

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Question No: LRA 4.7.4-1

NRC Request:
Section 4.7.4 HELB Location Postulation Based on Fatigue CUF

Question 4.7.4-1 (HELB location postulation based on fatigue CUF)

LRA section 4.7.4 indicates that current usage factors used for the postulation of break locations in Class 1 piping may be used for the 60 year operating term. However, Section 4.3.1.7 of the LRA indicates that the pressurizer surge line was not bounded by the original 40 year design transients and was subsequently reanalyzed for 60-year transients. Clarify whether there are any Class 1 piping locations where the cumulative usage factor may exceed 0.1 during the period of extended operation.

HNP Response:
The discussion in Section 4.3.1.7 was provided to indicate that a more detailed evaluation was required. The detailed evaluation included the effects of surge line transients considering the effects of thermal stratification and plant-specific cyclic data. The effects of reactor water environment on fatigue are described in LRA Section 4.3.3. A review of the analysis prepared for license renewal provided the values of cumulative usage factors before the application of the fatigue life correction factor (Fen). The unadjusted cumulative usage factor (U) at the hot leg nozzle safe end to pipe weld is 0.1138. The unadjusted cumulative usage factor (U) at the pressurizer nozzle safe end to pipe weld is 0.0288. However, these are based on 133 heatup/cooldown cycles projected for 60 years. For the remaining non-heatup/cooldown transients, the 40-year design number of cycles was included in the fatigue pairs. Extrapolating the unadjusted CUF to the design limit of 200 heatup/cooldown cycles can be accomplished as follows: $U_{200} = U_{133} * 200/133$. This yields CUFs of 0.1711 at the hot leg nozzle safe end to pipe weld and 0.0433 at the pressurizer nozzle safe end to pipe weld. These values can then be compared to the current licensing basis (CLB) values 0.85 and 0.2, respectively. The cumulative usage factors from the refined analysis prepared for license renewal, when adjusted for the full 40-year design cycles, are bounded by the existing CLB analysis. Therefore, the break locations postulated in the CLB remain applicable.