WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President, Engineering

July 26, 2007
ET 07-0031

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

Reference:
1) Letter ET 06-0038, dated September 27, 2006, from T. J. Garrett, WCNOC, to USNRC
2) Letter ET 07-0020, dated May 25, 2007, from T. J. Garrett, WCNOC, to USNRC
3) Letter ET 07-0051, dated June 7, 2007, from M.W. Sunseri, WCNOC, to USNRC

Subject: Docket No. 50-482: Revision to the Aging Management Program, Aging Management Review and Time-Limited Aging Analysis Questions and Responses Related to Wolf Creek Generating Station License Renewal Application

Gentlemen:

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) License Renewal Application (LRA) for the Wolf Creek Generating Station (WCGS). As part of the review for license renewal, the Nuclear Regulatory Commission (NRC) staff conducted three audits at WCGS. The LRA Aging Management Programs (AMP) audit was performed during the week of March 26, 2007 and the Aging Management Reviews (AMR) audit during the week of May 7, 2007. During the course of these audits and during the week of July 9, 2007 the NRC staff also audited Time Limited Aging Analyses (TLAA).

After the May 7, 2007 audit, the question and answer databases complied during the audits were submitted in References 2 and 3. NRC staff review of WCGS responses resulted in additional questions and the need for clarifications to some WCNOC responses. The databases were revised to include the additional information. This submittal provides only the additional questions and responses for the AMR and AMP database. The TLAA database is included in its entirety. Enclosure 1 provides the AMR and AMP database additional questions and Enclosure 2 the TLAA database. Each entry consists of a numbered question, reference to the applicable section of the LRA and the WCNOC response.

The attachment provides a comprehensive commitment list including all commitments made in Reference 1 and subsequent submittals. Prior to this letter, WCNOC had made thirty-five commitments. An additional commitment, number thirty-six, has been added.

A121
NRR

This commitment requires amending the LRA as described in the responses provided in the TLAA question and answer database. WCNOC previously committed to amending the LRA to reflect responses provided in the AMR database in Reference 2, commitment number thirty-one. This commitment has been revised to include this letter as a reference. Commitment twenty-one has been revised to include specific details concerning action limits and corrective actions.

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment – List of Commitments

Enclosures 1- Wolf Creek AMP and AMR Audit Questions and Responses (Additional)
2- Wolf Creek TLAA Audit Questions and Responses

cc: J. N. Donohew (NRC), w/a, w/e
V. G. Gaddy (NRC), w/a, w/e
B. S. Mallett (NRC), w/a, w/e
V. Rodriguez (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By ______________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 26th day of July, 2007.

NOTARY PUBLIC OFFICIAL SEAL STATE OF KANSAS
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

LICENSE RENEWAL APPLICATION - LIST OF REGULATORY COMMITMENTS

The following table identifies a summary of those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in the License Renewal Application (LRA) and subsequent LRA correspondence. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Kevin Moles at (620) 364-4126.

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
1	Boric Acid Corrosion Program (RCMS 2006-198)	A1.4	Prior to the period of extended operation, procedures will be enhanced to state that susceptible components adjacent to potential leakage sources will include electrical components and connectors. **Reference: ET 06-0038** **Due: March 11, 2025**
2	Nickel-Alloy Penetration Nozzles Welded To The Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (RCMS 2006-199)	A1.5	Prior to the period of extended operation, procedures will be enhanced to indicate that detection of leakage or evidence of cracking in the vessel head penetration nozzles or associated welds will cause an immediate reclassification to the "High" susceptibility ranking, commencing from the same outage in which the leakage or cracking is detected. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
3	Closed-Cycle Cooling Water System (RCMS 2006-200)	A1.10	Prior to the period of extended operation, a new periodic preventive maintenance activity will be developed to specify performing inspections of the internal surfaces of valve bodies and accessible piping while the valves are disassembled for operational readiness inspections to detect loss of material and fouling. The acceptance criteria will be specified in this Preventive Maintenance activity. **Reference: ET 06-0038** **Due: March 11, 2025** **Revised ET 07-0020**
4	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (RCMS 2006-201)	A1.11	Prior to the period of extended operation, procedures will be enhanced to: (1) identify industry standards or Wolf Creek Generating Station (WCGS) specifications that are applicable to the component, and (2) specifically inspect for loss of material due to corrosion or rail wear. **Reference: ET 06-0038** **Due: March 11, 2025**
5	Fire Protection (RCMS 2006-202)	A1.12	Prior to the period of extended operation: (1) fire damper inspection and drop test procedures will be enhanced to inspect damper housing for signs of corrosion, (2) fire barrier and fire door inspection procedures will be enhanced to specify fire barriers and doors described in USAR Appendix 9.5A, "WCGS Fire Protection Comparison to APCSB 9.5-1 Appendix A", and WCGS Fire Hazards Analysis, and (3) training for technicians performing the fire door and fire damper visual inspection will be enhanced to include fire protection inspection requirements and training documentation. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
6	Fuel Oil Chemistry (RCMS 2006-203	A1.14	Prior to the period of extended operation: (1) the emergency fuel oil day tanks will be added to the ten year drain, clean, and internal inspection program, and (2) procedures will be enhanced to provide for supplemental ultrasonic thickness measurements if there are indications of reduced cross sectional thickness found during the visual inspection of the emergency fuel oil storage tanks. A one time ultrasonic (UT) or pulsed eddy current (PEC) thickness examination on the external surface of engine driven fire pump fuel oil tank (1DO002T) will be performed to detect corrosion related wall thinning. If UT is used, the examination will be on a 4 inch grid. The examination will be performed once between 10 and 2 years prior to the period of extended operation. **Reference: ET 06-0038** **Due: March 11, 2025** **Revised ET 07-0020**
7	One-Time Inspection (RCMS 2006-204)	A1.16	The One-Time Inspection program conducts one-time inspections of plant system piping and components to verify the effectiveness of the Water Chemistry program (A1.2), Fuel Oil Chemistry program (A1.14), and Lubricating Oil Analysis program (A1.23). This new program will be implemented and completed within the ten-year period prior to the period of extended operation. **Reference: ET 06-0038** **Due: March 11, 2025**
8	Selective Leaching of Materials (RCMS 2006-205)	A1.17	The Selective Leaching of Materials program is a new program that will be implemented prior to the period of extended operation. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
9	Buried Piping and Tanks Inspection (RCMS 2006-206)	A1.18	The Buried Piping and Tanks Inspection program is a new program that will be implemented prior to the period of extended operation. Within the ten-year period prior to entering the period of extended operation, an opportunistic or planned inspection will be performed. Upon entering the period of extended operation a planned inspection within ten years will be required unless an opportunistic inspection has occurred within this ten-year period. **Reference: ET 06-0038** **Due: March 11, 2025**
10	One-Time Inspection of ASME Code Class 1 Small-Bore Piping (RCMS 2006-207)	A1.19	The fourth interval of the ISI program at WCGS will provide the results for the one time inspection of ASME Code Class 1 small-bore piping. **Reference: ET 06-0038** **Due: March 11, 2025**
11	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (RCMS 2006-208)	A1.22	The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program that will be implemented prior to the period of extended operation. For those systems or components where inspections of opportunity are insufficient, an inspection will be conducted prior to the period of extended operation to provide reasonable assurance that the intended functions are maintained. **Reference: ET 06-0038** **Due: March 11, 2025**
12	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (RCMS 2006-209)	A1.24	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will be implemented prior to the period of extended operation. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
13	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (RCMS 2006-210)	A1.25	A review of the calibration surveillance test results will be completed before the period of extended operation and every 10 years thereafter. **Reference: ET 06-0038** **Due: March 11, 2025**
14	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (RCMS 2006-211)	A1.26	The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will be implemented prior to the period of extended operation. **Reference: ET 06-0038** **Due: March 11, 2025**
15	ASME Section XI, Subsection IWL (RCMS 2006-212)	A1.28	Prior to the period of extended operation, procedures will be enhanced to include two new provisions regarding inspection of repair/replacement activities. The 2001 Edition with 2002 and 2003 addenda of ASME Section XI, Subsection IWL, Article IWL-2000, includes two provisions that are not required by the 1998 edition. IWL-2410(d) specifies additional inspections for concrete surface areas affected by a repair/replacement activity, and IWL-2521.2 specifies additional inspections for tendons affected by a repair/replacement activity. In accordance with 10 CFR 50.55a, WCGS will revise their CISI program prior to the next inspection interval to incorporate the ASME Code edition and addenda incorporated into 10 CFR 50.55a at that time. **Reference: ET 06-0038** **Due: March 11, 2025** **Revised ET 07-0020**
16	Masonry Wall Program (RCMS 2006-213)	A1.31	Prior to the period of extended operation, procedures will be enhanced to identify un-reinforced masonry in the Radwaste Building within the scope of license renewal that requires aging management. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
17	Structures Monitoring Program (RCMS 2006-214)	A1.32	Prior to the period of extended operation, procedures will be enhanced to add inspection parameters for treated wood and to monitor groundwater for pH, sulfates, and chlorides. Two samples of groundwater will be tested every five years. **Reference: ET 06-0038** **Due: March 11, 2025** **Revised ET 07-0020**
18	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (RCMS 2006-215)	A1.33	Prior to the period of extended operation, procedures will be enhanced: (1) so that the main dam service spillway and the auxiliary spillway will be inspected in accordance with the same specification, (2) to clarify the scope of inspections for the spillways, (3) to add the 5 year inspection frequency for the main dam service spillway, and (4) to add cavitation to the list of concrete aging effects for surfaces other than spillways. **Reference: ET 06-0038** **Due: March 11, 2025**
19	Reactor Coolant System Supplement (RCMS 2006-216)	A1.35	WCNOC will: A. Reactor Coolant System Nickel Alloy Pressure Boundary Components Implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines, and B. Reactor Vessel Internals (1) Participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, WCNOC will submit an inspection plan for reactor internals to the NRC for review and approval. **Reference: ET 06-0038** **A, B(1), B(2) Due: March 11, 2025** **B(3) Due: March 11, 2023**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
20	Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements (RCMS 2006-217)	A1.36	Prior to the period of extended operation, the infrared thermography testing procedure will be enhanced to require an engineering evaluation when test acceptance criteria are not met. This engineering evaluation will include identifying the extent of condition, the potential root cause for not meeting the test acceptance, and the likelihood of recurrence. **Reference: ET 06-0038** **Due: March 11, 2025**
21	Metal Fatigue of Reactor Coolant Pressure Boundary (RCMS 2006-218)	A2.1	Prior to the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include: (1) Action levels to ensure that if the fatigue usage factor calculated by the code analysis is reached at any monitored location, appropriate evaluations and actions will be invoked to maintain the analytical basis of the leak-before-break (LBB) analysis and of the high-energy line break (HELB) locations, or to revise them as required, (2) Action levels to ensure that appropriate evaluations and actions will be invoked to maintain the bases of safety determinations that depend upon fatigue analyses, if the fatigue usage factor at any monitored location approaches 1.0, or if the fatigue usage factor at any monitored NUREG/CR6260 location approaches 1.0 when multiplied by the environmental effect factor F_{EN}, (3) Corrective actions, on approach to these action levels, that will determine whether the scope of the monitoring program must be enlarged to include additional affected reactor coolant pressure boundary locations in order to ensure that additional locations do not approach the code limit without an appropriate action, and to ensure that the bases of the LBB and HELB analyses are maintained, (4) 10 CFR 50 Appendix B procedural and record requirements.

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
			Cycle Count Action Limit and Corrective Actions An action limit will be established that requires corrective action when the cycle count for any of the critical thermal and pressure transients is projected to reach a high percentage (e.g., 90%) of the design specified number of cycles before the end of the next operating cycle. If this action limit is reached, acceptable corrective actions include: 1. Review of fatigue usage calculations. • To determine whether the transient in question contributes significantly to CUF. • To identify the components and analyses affected by the transient in question. • To ensure that the analytical bases of the leak-before-break (LBB) fatigue crack propagation analysis and of the high-energy line break (HELB) locations are maintained. 2. Evaluation of remaining margins on CUF based on cycle-based or stress-based CUF calculations using the WCGS fatigue management program software. 3. Redefinition of the specified number of cycles (e.g., by reducing specified numbers of cycles for other transients and using the margin to increase the allowed number of cycles for the transient that is approaching its specified number of cycles).

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
			Cumulative Fatigue Usage Action Limit and Corrective Actions An action limit will be established that requires corrective action when calculated CUF (from cycle based or stress based monitoring) for any monitored location is projected to reach 1.0 within the next 2 or 3 fuel cycles. If this action limit is reached acceptable corrective actions include: 1. Determine whether the scope of the monitoring program must be enlarged to include additional affected reactor coolant pressure boundary locations. This determination will ensure that other locations do not approach design limits without an appropriate action. 2. Enhance fatigue monitoring to confirm continued conformance to the code limit. 3. Repair the component. 4. Replace the component. 5. Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded. 6. Modify plant operating practices to reduce the fatigue usage accumulation rate. 7. Perform a flaw tolerance evaluation and impose component-specific inspections, under ASME Section XI Appendices A or C (or their successors), and obtain required approvals by the NRC.

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
			Prior to the period of extended operation, changes in available monitoring technology or in the analyses themselves may permit different action limits and action statements, or may re-define the program features and actions required to address fatigue time-limited aging analyses. (TLAAs) **Reference: ET 06-0038** **Due: March 11, 2025** **Revised ET 07-0031**
22			Deleted
23	Concrete Containment Tendon Prestress (RCMS 2006-220)	A2.3	Prior to the period of extended operation, procedures will be revised to: (1) extend the list of surveillance tendons to include random samples for the year 40, 45, 50, and 55 year surveillances, (2) explicitly require a regression analysis for each tendon group after every surveillance, (3) invoke and describe regression analysis methods used to construct the lift-off trend lines, (4) extend surveillance program predicted force lines for the vertical and hoop tendon groups to 60 years, and (5) conform procedure descriptions of acceptance criteria action levels to the ASME Code, Subsection IWL 3221 descriptions. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
24	ASME III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals (RCMS 2006-221)	A3.2.2	WCNOC will obtain a design report amendment to either quantify the increase in high-cycle fatigue effects, or to confirm that the increase will be negligible. WCNOC will complete this action before the end of the current licensed operating period. **Reference: ET 06-0038** **Due: March 11, 2025**
25	Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in B31.1 and ASME III Class 2 and 3 Piping (RCMS 2006-222)	A3.2.4	WCNOC will complete the reanalysis of the reactor coolant sample lines and any additional corrective actions or modifications indicated by them, before the end of the current licensed operating period. **Reference: ET 06-0038** **Due: March 11, 2025**
26	USAR Supplement (RCMS 2006-223)	A0	Following issuance of the renewed operating license in accordance with 10 CFR 50.71(e), WCNOC will incorporate the USAR supplement into the WCGS USAR as required by 54.21(d). **Reference: ET 06-0038** **Due: USAR update following issuance of the renewed operating license in accordance with 10CFR 50.71(e).** **Revised ET 07-0020**
27	Pressure-Temperature (P-T) Limits (RCMS 2006-224)	A3.1.3	WCNOC will revise the Pressure and Temperature Limits Report for a 60-year licensed operating life. **Reference: ET 06-0038 Due: March 11, 2025**
28	Implementation of New Programs (RCMS 2006-225)	N/A	Implementation of new programs may require additional action items not included in this list. WCGS is committed to including new program elements in the corrective action program. **Reference: ET 06-0038** **Due: March 11, 2025**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
29	LRA Amendment (RCMS 2007-250)	N/A	License Renewal Application changes discussed in ET 07-0011 will be submitted in an amendment to the Application. **Reference: ET 07-0011** **Revised ET 07-0025** **Due: August 31, 2007**
30	Nickel Alloy Aging Management Program (RCMS 2007-251)	A1.34	The WCGS Nickel Alloy Aging Management inspection plan will be submitted for NRC review and approval at least 24 months prior to entering the period of extended operation **Reference: ET 07-0016** **Due: March 11, 2023**
31	LRA Amendment (RCMS 2007-252)	N/A	License Renewal Application changes discussed in ET 07-0020 and ET 07-0031, Enclosure 1, will be submitted in an amendment to the Application. **Reference: ET 07-0020, ET 07-0031** **Due: August 31, 2007**
32	Closed-Cyle Cooling Water System (RCMS 2007-253)	N/A	WCNOC Procedure QCP-20-518, "Visual Examination of Heat Exchangers and Piping Components", will be revised to define cracking, provide additional guidance for detection of cracking and specific acceptance criteria relating to "as found" cracking. **Reference: ET 07-0020** **Due: March 11, 2025**
33	LRA Amendment (RCMS 2007-254)	N/A	License Renewal Application changes discussed in WM 07-0050 will be submitted in an amendment to the Application. **Reference: WM 07-0050** **Due: August 31, 2007**
34	LRA Amendment (RCMS 2007-255)	N/A	License Renewal Application changes discussed in WM 07-0051 will be submitted in an amendment to the Application. **Reference: WM 07-0051** **Due: August 31, 2007**

	COMMITMENT SUBJECT	LRA, Appendix A, Section	COMMITMENT DESCRIPTION
35	LRA Amendment (RCMS 2007-269)	N/A	License Renewal Application changes discussed in ET 07-0028 will be submitted in an amendment to the Application. **Reference: ET 07-0028** **Due: August 31, 2007**
36	LRA Amendment (RCMS 2007-270)	N/A	License Renewal Application changes discussed in ET 07-0031, Enclosure 2, will be submitted in an amendment to the Application. **Reference: ET 07-0031** **Due: August 10, 2007**

Wolf Creek AMP Audit Questions and Responses (Additional)

Wolf Creek AMR Audit Questions and Responses (Additional)

Question No	LRA Sec	Audit Question	Final Response
AMPA135	B.2.1.8	The Steam Generator Tube Integrity Program does not indicate that the steam generator feedwater ring is within the scope of the AMP. In LRA Section 3.1.2.2.14 the applicant states that the issue discussed in SRP LR Section 3.1.2.2.14 is not applicable to WCGS; however, the applicant credits the Water Chemistry and Steam Generator Tube Integrity Programs to manage loss of material due to flow accelerated corrosion in the steam generator feedwater inlet rings and supports. Please clarify how the scope of this AMP appropriately provides for inspections of this component and how the inspection method, sample size, and frequency performed will be capable of managing loss of material (wall thinning) due to flow accelerated corrosion in the steam generator feedwater inlet ring.	The scope of the Wolf Creek Generating Station (WCGS) Steam Generator Tube Integrity Program is consistent with NEI 97-06, Steam Generator Program Guidelines. NEI 97-06 recommends that secondary side components that are susceptible to degradation be monitored if their failure could affect the intended function of the steam generator. The WCGS Steam Generator Tube Integrity Program includes the feedwater ring and J tubes as part of the secondary side inspections. The WCGS Steam Generator Tube Integrity Program provides instructions for visual inspections of the upper steam drum, including the feedwater ring and the J tubes, on at least one steam generator each outage.
AMPA136	B.2.1.8	Clarify if the Steam Generator Tube Integrity Program has been augmented or enhanced to include inspections of the following commodity groups: ·steam generator feedring and feedring J-tubes ·steam generator flow distribution baffle ·steam generator internal structures - non pressure	The WCGS Steam Generator Tube Integrity Program includes a Secondary Side Condition Monitoring and Operational Assessment that are used to document the secondary side integrity plan. Elements of this integrity plan include secondary side cleanings and secondary side visual inspections. Eddy current analysis is performed to determine sludge and scale build-up. The Eddy current analysis supplements visual inspections to determine the overall condition of the steam generator. Wolf Creek has performed a thorough baseline visual inspection of the secondary side of the steam generators. Wolf Creek operating experience has not identified any aging related failures that resulted in a loss of intended functions for steam generator secondary side

Question No	LRA Sec	Audit Question	Final Response
		boundary miscellaneous parts ·steam generator secondary blowdown apparatus If so, justify why the augmented or enhanced inspections bases for these commodity groups are considered to be capable of managing the applicable aging effect in the applicable AMR items (i.e., cracking, wall thinning, or loss of material). If the Steam Generator Tube Integrity Program has not been enhanced or augmented to cover these commodity groups, clarify which program will be used to inspect for the applicable aging effects in these commodity groups, and justify why the inspection methods are considered to be capable of managing the applicable effects of concern.	components. Aging management activities for the following commodity groups are included in the secondary side integrity plan: (1) steam generator feedring and feedring J-tubes The WCGS Steam Generator Tube Integrity Program manages the aging of the feedring and the feedring J-tubes. See response to AMPA135. The justification for managing the aging effect of wall thinning of the feedring is provided in the response to the Audit Question AMRA015. The management of the aging effects of cracking and loss of material for the J-tubes is consistent with the steam generator anti-vibration bars evaluated by GALL lines IV.D1-14 and IV.D1-15. (2) steam generator flow distribution baffle According to Updated Safety Analysis Report (USAR) 5.4.2.2, the flow distribution baffle serves to minimize the sludge deposit by directing flow to the center of the tube bundle. Eddy current analysis is performed to determine sludge and scale build-up. Sludge accumulation is monitored for areas of concern, including tube support plates and the quatrefoils of the support plates. (3) steam generator internal structures - non pressure boundary miscellaneous parts Overall condition of the internal structures is monitored by visual inspection of the secondary side. This includes areas of tube support plates and upper steam drum. The management of the aging effect of loss of material is consistent with the steam generator tube wrapper evaluated by GALL line IV.D1-9. (4) steam generator secondary blowdown apparatus Foreign object search and visual inspection are performed for the tubesheet annulus and the blowdown lane. This is performed

Question No	LRA Sec	Audit Question	Final Response
			following sludge lancing or as eddy current inspection data indicates. The management of the aging effects of cracking and loss of material is consistent with the steam generator anti-vibration bars evaluated by GALL lines IV.D1-14 and IV.D1-15.
AMPA137	B.2.1.6	In its response to audit question AMPA008, the applicant provided details of the actions taken to address the erroneous wear rate predictions by the CHECWORKS model as reported in PIR 20002032. The response states that the possible cause of the erroneous predictions could be the backing rings installed during construction. a)Explain if the EPRI specialist who reviewed the CHECWORKS model agreed with the WCGS position that the erroneous predictions were caused by the backing rings. Provide supporting information. b)Describe if similar erroneous predictions have been experienced after the modifications to the CHECWORKS model were performed. Explain the specific actions taken by WCGS to assure that the wear rate predictions at other locations with backing rings will be correctly interpreted.	a) The Electric Power Research Institute (EPRI) specialist brought on site in August-September of 1999 agreed that the backing rings that were installed at these locations could be the causes for wear in this line and for the erroneous wear rate predictions in the CHECWORKS model. EPRI was requested to provide technical assistance to flow-accelerated corrosion (FAC) engineering in the evaluation of the WCGS inspection results, at the location equivalent to Callaway's rupture location, and potential applicability to other locations at WCGS. The key objective of the EPRI/WCNOC joint effort evaluation was to identify piping locations for emergent, on-line inspections, in order to detect unexpected pipe wall thinning due to conditions similar to the Callaway failure. Secondary objectives were to look for improvements to the CHECWORKS FAC model and identify additional locations for inspection in future outages. b) Recent inspections (RF13, 14 & 15) have not identified other locations with high wear. In August of 1999, the Callaway station had a rupture of a 6-inch drain line from the moisture separator reheater drain tank. Since Wolf Creek is similar to Callaway in design, Wolf Creek performed inspection of the equivalent location of the Callaway failure. In order to determine locations for future inspections, a review of the FAC model in CHECWORKS was performed by EPRI. EPRI completed this review and found no major problem with the Wolf Creek model in the CHECWORKS program. EPRI also provided a list of additional inspection locations. From the inspection performed of the piping in the equivalent location to Callaway's failure, Wolf Creek replaced these components immediately with like-for-like components (Aug. of 1999). In Refueling Outage 12 (RF12, 2002), Wolf Creek replaced the complete line from the control valve to the inlet nozzle of the high-pressure heater; this is

Question No	LRA Sec	Audit Question	Final Response
			approximately one hundred and fifteen feet of pipe, with FAC resistant chrome-moly pipe and without backing rings. Therefore, this portion of piping is no longer considered to be susceptible to FAC. The CHECWORKS program is an EPRI developed computer program that is used by the nuclear industry for the prediction of wear in susceptible piping. This program is continually being evaluated and updated by EPRI. EPRI uses input from plants from all over the United States for updates to the program. Wolf Creek upgraded the CHECWORKS program in March 2006, to use the latest version of CHECWORKS (SFA version 2.1) from version 1.0G. The model was verified/updated again at that upgrade. Adjustments to the CHECWORKS model based on inspection data can be used to assure that the wear rate predictions at other locations with backing rings will be correctly interpreted. The EPRI CHECWORKS software program uses an empirical model to predict FAC wear rates on a component-by-component basis. Once inspection data becomes available the empirical model (CHECWORKS) adjusts its predictions to calibrate the predictions to the field data and determine absolute wear rates. This adjustment method is known as applying the line correction factor (LCF). Therefore, as a good correlation (low LCF) is established, it can serve as the basis for determining the wear status of those components that have not been inspected and indicate that the actual versus predict wear correlate. After each outage the inspection data is incorporated into CHECWORKS program and new wear rate predictions are determined. In Refueling Outages 13, 14 and 15, Wolf Creek inspected on average about 40 to 50 locations each outage that were incorporated into the CHECWORKS model. In general, Wolf Creek has LCF values that indicate a good correlation between the actual and predicted wear. A good correlation is typically indicated by the LCF being within the range of 0.5 to 2.5.

Question No	LRA Sec	Audit Question	Final Response
AMRA071	3.1	Three line items in LRA Table 3.1.2-1, crediting Notes "I" and "2," address crack growth in carbon steel with stainless steel cladding in reactor coolant. Note 2 refers to LRA Section 3.1.2.2.5 for further evaluation. LRA Section 3.1.2.2.5 references Westinghouse WCAP 15338-A to support the statement (generic Note I) that the aging effect for this component, material and environment combination is not applicable at WCGS. The staff approved the use of WCAP-15338-A as a reference in LRAs for Westinghouse 4 loop plants in an SER, dated September 25, 2002 (ML022690375). The subject SER includes the following license renewal action items: ·The license renewal applicant is to verify that its plant is bounded by the WCAP-15338 report. Specifically, the renewal applicant is to indicate whether the number of design cycles and transients assumed in the WCAP-15388 analysis bounds	The Nuclear Regulatory Commission (NRC)safety evaluation of WCAP-15338-A notes that "Underclad cracks … have been reported …only in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes." In the WCGS vessel only the carbon steel forgings are SA-508 Class 2 or 3. The clad is stainless steel weld metal, Analysis A8; and Ni-Cr-Fe Weld Metal, F-Number 43. Although the vessel contains these SA-508 forgings clad by high-heat-input processes, the qualification of clad welding processes to avoid cracking is documented in WCGS USAR Section 5.3.1.2.g and Appendix 3A section on Reg. Guide 1.43, Revision 0, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components". No underclad flaws have been detected or analyzed for the WCGS reactor vessel therefore WCAP-15338-A was not invoked. See License Renewal Application (LRA) Section 4.7.2 for additional details.

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		the number of cycles for 60 years of operation of its RPV. ·Section 54.21(d) of 10 CFR requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those applicants for license renewal referencing the WCAP-15338 report for the RPV components shall ensure that the evaluation of the TLAA is summarily described in the FSAR supplement. a) Clarify if WCGS has completed the action items as described in the WCAP 15338-A SER. b)Provide documentation for staff review to support that these action items have been completed.	
AMRA072	3.1	SRP-LR Section 3.1.3.2.16.2 recommends a one-time inspection of the pressurizer spray head. However, the staff	As indicated in LRA Section 3.1.2.2.16.2, the pressurizer spray head is not included in scope of license renewal. LRA Table 3.1.1 item 3.1.1-36 includes Reactor Coolant System (RCS) stainless steel pipes and valves in the scope of license renewal with a

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		was unable to identify an AMR result line in LRA Table 3.1.2-2 related to the pressurizer spray head. a)Identify where is the AMR result line for the pressurizer spray heads. b)Clarify if there will be a one time inspection as recommended in the SRP LR.	structural integrity (attached) intended function (Criterion (a)(2)). The RCS stainless steel components with the structural integrity (attached) intended function were evaluated as consistent with NUREG-1801 item IV.C2-17 which is included in LRA Table 3.1.1 item 3.1.1-36. The discussion column of LRA Table 3.1.1 item 3.1.1-36 and its associated further evaluation LRA section 3.1.2.2.16-2 (SRP-LR section 3.1.2.2.16-2) reflects the evaluation of these RCS Criterion (a)(2) components. The pressurizer spray head is a non-pressure boundary subcomponent. The function of the spray head is to disperse the flow for maximizing condensation of the steam bubble. Failure of the spray head would not prohibit the spray water from entering the pressurizer for condensing the steam. The spray water would be still available as a stream instead of a fine spray. The intended function of pressurizer spray would not be impaired by the failure of the spray head. Thus, the pressurizer spray head is not relied on to provide the intended function for 10CFR54.4(a)(1). According to Post-Fire Safe Shutdown Analysis, the pressurizer spray is isolated to prevent RCS pressure reduction. The steam generator atmospheric relief valves are used to control RCS cooldown. Thus the pressurizer spray head does not provide any intended function for 10CFR54.4(a)(3). If the pressurizer spray head were to degrade and shed one or more pieces of the head, the postulated loose parts may affect the operation of the PORV or the code safety valve during pressurization transients. However, based on the operating experience and industry experience, the possibility of its hypothetical failure is not sufficient to include the pressurizer spray head in scope for 10CFR54.4(a)(2).

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AMRA073	3.3	In its response to audit question AMRA039, the applicant stated that the LRA would be amended to add two new generic items for flex hoses in the fire protection system for elastomer materials in dry gas internal and plant indoor air external environments with an aging effect of hardening and loss of strength. The Fire Protection Program will be credited to manage these aging effects. LRA Table 3.3.1, item 3.3.1-11, was also referenced. However, the response did not indicate if the discussion column of LRA Table 3.3.1-11 would be amended to add the Fire Protection Program. The response also stated that the Fire Protection Program would be amended, but did not indicate the changes. a)Provide the proposed LRA changes for the Fire Protection Program. b)Clarify if LRA Table 3.3.1, item 3.3.1-11, will be amended to add the Fire Protection Program in the discussion column. In	a) Proposed LRA Changes: LRA Appendix A1.12 - Amended first sentence of first paragraph: The Fire Protection program manages loss of material for fire rated doors, fire dampers, diesel-driven fire pump, and the halon fire suppression system; cracking, spalling, and loss of material for fire barrier walls, ceilings, and floors; hardness and shrinkage due to weathering of fire barrier penetration seals; and hardness-loss of strength for halon fire suppression system flexible hoses. LRA Appendix A1.12 - Added paragraph (at the end of A1.12): Prior to the period of extended operation, halon fire suppression system inspection procedures will be enhanced to include visual inspections of halon tank flexible hoses for hardening-loss of strength. Visual inspections would not be required for flexible hoses that have scheduled periodic replacement intervals. LRA Appendix B2.1.12 - Amended first sentence of first paragraph: identical to Appendix A1.12 first sentence first paragraph above. LRA Appendix B2.1.12 - Subsection "Enhancements" - Added as enhancements to Elements 3 and 4: Prior to the period of extended operation, halon fire suppression system inspection procedures will be enhanced to include visual inspections of halon tank flexible hoses for hardening-loss of strength. Visual inspections would not be required for flexible hoses that have scheduled periodic replacement intervals. LRA Table 3.3.2-14 will be amended to change existing standard Notes for Flexible Hoses in an environment with the ambient temperature of less than 95^0 F, from "J" to "G".

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		addition, the response to audit question AMRA029 also amends the discussion column of item 3.3.1-11. Clarify how these two responses would change the discussion column. c)Explain how the Fire Protection Program will manage the internal surfaces of the flexible hoses.	LRA Table 3.3.2-14 will be amended to add the following two new lines for flexible hoses in an environment with ambient temperature greater than 95^0 F: Component Type: Flexible Hoses Intended Function: PB Material: Elastomer Environment: Dry Gas (internal) Aging Effect: Hardening and Loss of Strength AMP: Fire Protection (B2.1.12) NUREG-1801 Vol 2 Item: None Table 1 Item: None Note: G Component Type: Flexible Hoses Intended Function: PB Material: Elastomer Environment: Plant Indoor Air (external) Aging Effect: Hardening and Loss of Strength AMP: Fire Protection (B2.1.12) NUREG-1801 Vol 2 Item: VII.F2-7 Table 1 Item: 3.3.1.11 Note: E b) The discussion column for Table 3.3.1, Item 3.3.1.11 will be amended to add the Fire Protection program (B2.1.12)(note: exception in 3.3.1.11 for Control Bldg flex connectors was deleted in response to AMRA029). LRA Section 3.3.2.2.5.1 will be amended to insert the following as a new third paragraph: The Fire Protection program (B2.1.12) will manage the hardening and loss of strength from elastomer degradation for halon fire suppression system flexible hoses not periodically replaced in locations where the ambient temperature cannot be shown to be less than 95^0 F.

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			c) Halon Tank weight and pressure check surveillance tests are performed every 18 months and provide inspection opportunities for visual inspection of the halon tank flexible hoses. Procedures STN FP-404A and STN FP-404B will be enhanced to include visual inspection of halon tank flexible hoses.
AMRA074	3.3	In its response to audit question AMRA021, the applicant stated that a new commitment would be added to the LRA commitment list. Clarify if the Closed Cycle Cooling Water System Program and its FSAR supplement will be amended to add this enhancement and commitment. Provide the proposed LRA changes.	WCNOC letter ET 07-0020, dated May 25, 2007 added commitment number 32 (RCMS 2007-253) to the License Renewal Application - List of Regulatory Commitments. Commitment number 32 states "WCNOC Procedure QCP-20-518, "Visual Examination of Heat Exchangers and Piping Components", will be revised to define cracking, provide additional guidance for detection of cracking and specific acceptance criteria relating to "as-found" cracking." LRA Appendix A1.10 and Appendix B2.1.10 will be amended as follows: Appendix A1.10 will be amended to include: "Visual inspection procedures used for identification of stress corrosion cracking (SCC) will be enhanced to define cracking, provide additional guidance for detection of cracking and identify specific acceptance criteria relating to "as-found" cracking." Appendix B2.1.10 will be amended to include: "Detection of Aging Effects - Element 4 Visual inspection procedures used for identification of stress corrosion cracking (SCC) will be enhanced to define cracking, provide additional guidance for detection of cracking and identify specific acceptance criteria relating to "as-found" cracking."
AMRA075	3.4	In its response to audit question AMRA041, the applicant stated	NUREG 1801 does conclude that there are no aging effects that require management for aluminum and stainless steel in plant

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		that the GALL Report concludes that there are no aging effects that require management for stainless steel (sheathing) and aluminum (sheathing) in plant indoor air. a)LRA Tables 3.4.2-2 and 3.4.2-3, assign Note J to the line items pertaining to the stainless steel and aluminum jacketing exposed to plant indoor air. This note implies that neither the component nor the material and environment combination is addressed in the GALL Report. In light of the response provided in AMRA041, the staff believes that the Note needs to be revised to make these AMR line items consistent with the GALL Report. Clarify if the LRA will be amended to reflect this change. b)The response provided in AMRA041 also states that all the insulation within the scope of license renewal is jacketed with either stainless steel or aluminum. Therefore, LRA Table 3.4.2-5 should include AMR line items for both stainless steel and	indoor air as stated in the response to AMRA041. However, the NUREG 1801 lines are not specific to aluminum jacketing or stainless steel jacketing but apply to other component types. Therefore "Standard Note C" would be applicable to address the different component types. a) LRA Tables 3.4.2-2 (main steam system) and 3.4.2-3 (feedwater system) assigned Note J to aluminum jacketing exposed to plant indoor air. The LRA will be amended to specify NUREG 1801 line V.F-2, Table 1 Item 3.2.1.50, and Note C for aluminum jacketing in plant indoor air in LRA Tables 3.4.2-2 and 3.4.2-3. b) LRA Table 3.4.2-5 (steam generator blowdown system) does not include the aluminum jacketing and stainless steel jacketing that is used to protect the blowdown system piping insulation materials. The LRA will be amended to specify NUREG 1801 line V.F-2, Table 1 Item 3.2.1.50, and Note C for aluminum jacketing in plant indoor air and specify NUREG 1801 line VIII.I-10, Table 1 Item 3.4.1.41, and Note C for stainless steel jacketing in plant indoor air in LRA Table 3.4.2-5.

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		aluminum exposed to plant indoor air similar to those in LRA Tables 3.4.2-2 and 3.4.2-3. Clarify this inconsistency.	
AMRA076	3.5	LRA Table 3.5.1, item 3.5.1-59, states that there is no aging effect for the component type of stainless steel support members, welds, bolted connections, and support anchorage to building structure. Identify the environment or location of this line item (e.g., exposed to air indoor uncontrolled).	Line item 3.5.1.59 is linked to six lines in Table 3.5.2-22. Three of those lines have an environment of "Plant Indoor Air (Structural)". The GALL lines for these refer to "Air - Indoor uncontrolled". These components are: Component Type: Supports ASME 2 & 3 NUREG 1801 Item: III.B1.2-7 Component Type: Supports Mech Equip Non ASME NUREG 1801 Item: III.B4-8 Component Type: Supports Non ASME NUREG 1801 Item: III.B2-8 Three of the linked lines have an environment of "Borated Water Leakage". The GALL lines for these refer to "Air with borated water leakage". These components are: Component Type: Supports ASME 1 NUREG 1801 Item: III.B1.1-10 Component Type: Supports ASME 2 & 3 NUREG 1801 Item: III.B1.2-8 Component Type: Supports Non ASME NUREG 1801 Item: III.B2-9

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AMRA077	3.5	LRA Section 3.5.2.2.1.4 states that the Structure Monitoring Program will identify and manage any cracks in the concrete or degradation of the moister barrier that could potentially provide a pathway for water to reach inaccessible portions of the steel containment liner. SRP-LR Section 3.5.2.2.1.4 states that the existing program relies on ASME Code Section XI, Subsection IWE, and 10 CFR Part 50, Appendix J to manage loss of material. LRA Table 3.5.2.1, item 3.5.1-06, assigns Note "B" and credits the ASME Section XI, Subsection IWE Program to manage loss of material in the containment liner. It seems that there is an inconsistency between the AMR result line and the text provided in the further evaluation, as one is addressing cracking and the other loss of material. Please clarify.	LRA Section 3.5.2.2.1.4 will be amended to note that the WCGS program relies on ASME Code Section XI, Subsection IWE, and 10 CFR Part 50, Appendix J to manage loss of material. LRA Table 3.5.2-1 will be amended to add 10 CFR Part 50, Appendix J (B2.1.30) as one of the AMP's for component type "Liner Containment" in Plant Indoor Air (Structural). The text in LRA Section 3.5.2.2.1.4 is intended to specifically address the conditions given in NUREG 1801, Item II.A1-11, in the AMP column. These conditions, if satisfied, allow the presumption that, for inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant. Therefore, further evaluation for corrosion in inaccessible areas of the steel containment liner is not required. WCGS meets these criteria as follows: 1.Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner. LRA Section 3.5.2.2.1.4 response: Reinforced concrete structures at WCGS were designed, constructed, and inspected in accordance with applicable ACI and ASTM standards, which provide for a good quality, dense, well cured, and low permeability concrete. Design practices and procedural controls ensured that the concrete was consistent with the recommendations and guidance provided by ACI 201.2R. The mixes were designed with entrained air content between 3% and 6%, and the concrete slumps were controlled throughout the batching, mixing, and placement processes. USAR Section 3.8 discusses the design requirements for each major structure.

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			2. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. LRA Section 3.5.2.2.1.4 response: The Structures Monitoring Program (B2.1.32) will identify and manage any cracks in the concrete (or degradation of the moister barrier) that could potentially provide a pathway for water to reach inaccessible portions of the steel containment liner. 3. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. LRA Section 3.5.2.2.1.4 response: The Structures Monitoring Program (B2.1.32) will identify and manage any (cracks in the concrete or) degradation of the moister barrier that could potentially provide a pathway for water to reach inaccessible portions of the steel containment liner. 4. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. LRA Section 3.5.2.2.1.4 response: Procedural controls will ensure that borated water spills are not common, and when detected are cleaned up in a timely manner.
AMRA078	3.1	LRA Table 3.1.1, item 3.1.1.24, states that "Water Chemistry will be augmented with ASME Section XI Inservice Inspection,	a) ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD volumetrically inspect piping welds in the reactor coolant system. WCGS reactor coolant loop piping is CASS and the reactor coolant piping welds are ultrasonic tested (UT) as required

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		Subsections IWB, IWC, and IWD because the CASS in the reactor coolant system piping at WCGS meets the NUREG-0313 requirements for ferrite content but not for carbon content." a) Provide an expanded discussion on the use of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program for detection of cracking due to SCC in CASS piping at WCGS. Address which examinations specified in ASME Section XI are credited and their capability to detect cracking due to SCC in CASS piping before a through-wall leak occurs. b) Clarify what is the carbon content used for CASS piping in the reactor coolant system at WCGS.	by Table IWB-2500-1 Examination Category B-J, Items B9.10 and B.9.11. UT inspection is a proven industry ASME Code technique for detection of weld and adjacent base metal cracking caused by SCC. b) The WCGS Certified Material Test Reports of CASS Class 1 piping indicate that the carbon content of the reactor coolant system CASS piping is 0.05% – 0.08%.
AMRA079	3.1	In its response to audit question AMRA002, WCGS stated that the reactor vessel closure head (O-ring leak monitoring tubes) described in LRA Table 2.3.1-1 is made of nickel alloy and is not associated with GALL Report	(a) Reactor Vessel Flange O-ring Leak Monitoring Tubes The reactor vessel flange O-ring leak monitoring tubes are made of nickel alloy and are evaluated with GALL Report, Volume 2, items IV.A2-14 and IV.A2-18, which are referenced to LRA Table 3.1.1, items 3.1.1.83 and 3.1.1.65, respectively. For LRA Table 3.1.1, items 3.1.1.83 and 3.1.1.65, the GALL recommends no further

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		Volume 2, item IV.A2-5, which is based on the material made of stainless steel. The response also stated that the reactor vessel closure head (O-ring leak monitoring tubes) is evaluated with GALL Report, Volume 2, items IV.A2-14 and IV.A2-18 and is referenced to LRA Table 3.1.1, items 3.1.1.83 and 3.1.1.65, respectively. The title of LRA Section 3.1.2.2.7.1, "PWR stainless steel reactor vessel flange leak detection line," and its associated description appear to be inconsistent with the response to AMRA002 because they imply that the WCGS vessel flange head detection lines are made of stainless steel. In addition, the title of LRA Section 3.1.2.2.7.1 does not mention the stainless steel bottom mounted instrument guide tubes (high pressure conduits) which are discussed in SRP-LR Section 3.1.2.2.7.1 and which are referenced to LRA Table 3.1.1, item 3.1.1.23. Explain the discrepancy between	evaluations of aging management. Thus, the evaluation of LRA Section 3.1.2.2.7.1 is not applicable to the reactor vessel flange O-ring leak monitoring tubes. (b) LRA Section 3.1.2.2.7.1 LRA Section 3.1.2.2.7.1 will be amended to change the title to "PWR stainless steel reactor vessel instrument tubes and bottom-mounted flux thimble guide tubes." The Discussion column of LRA Table 3.1.1, item 3.1.1.23 will be amended to state that the reactor vessel O-ring leak monitoring tubes are made of nickel alloy. (c) Bottom-Mounted Guide Tubes Aging Management The components that reference LRA Section 3.1.2.2.7.1 and LRA Table 3.1.1, item 3.1.1.23 are stainless steel reactor vessel instrument tubes in the upper internals and the bottom-mounted flux thimble guide tubes (High Pressure Conduits), which are described in USAR 3.9(N).5.1. The aging management evaluations of these components are addressed in LRA Table 3.1.2-1, page 3.1-52. The WCGS ASME Section XI ISI AMP (B2.1.2) manages aging of bottom-mounted flux thimble guide tubes and they receive a VT-2 visual inspection as specified in ASME Section XI, Table 2500-1, Category BP.

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		the description in LRA Subsection 3.1.2.2.7.1 and the response to Question AMRA002. Clarify which components referencing SRP-LR Section 3.1.2.2.7.1 are addressed by this LRA Section. Revise the LRA Table 3.1.1-23 item accordingly.	
AMRA080	3.1	In its response to audit question AMRA002, the applicant described the function and configuration of the high pressure conduits. The line in LRA Table 3.1.2-1 for reactor vessel penetrations (high pressure conduits), GALL Report, item IV.A2-1, references LRA Table 3.1.1, item 3.1.1.23, and LRA Section 3.1.2.2.7.1. The LRA credits the Water Chemistry and ASME Section XI, Subsections IWB, IWC and IWD Programs to manage the aging effect of cracking due to SCC in the bottom mounted instrument guide tubes (high pressure conduits). a) Clarify whether there are any nickel alloy welds associated with the bottom mounted instrument guide tubes. If there are nickel alloy welds, identify where the	(a) There are Alloy 82/182 welds associated with the bottom mounted instrument guide tubes. It includes J-groove welds of the Flux Thimble Guide Tube Penetrations to the vessel bottom and the welds of the Flux Thimble Guide Tube Penetrations to the thimble guide tubes (high pressure conduits). The aging management of these welds is evaluated as part of the Flux Thimble Guide Tube Penetrations that are made of nickel alloys and are addressed in LRA Table 3.1.2-1, pages 3.1-50 and 3.1-51 as follows: <u>Component Type:</u> RV Penetrations (Flux Thimble Guide Tube Penetrations) <u>Intended Function:</u> PB <u>Material:</u> Nickel Alloys <u>Environment:</u> Reactor Coolant (Int) <u>Aging Effect:</u> Cracking <u>AMP:</u> Nickel Alloy Aging Management Program (B2.1.34) ASME Section XI ISI, Subsections IWB, IWC, and IWD (B2.1.1) Water Chemistry (B2.1.2) Comply with applicable NRC Orders and FSAR Commitment (B2.1.35) <u>NUREG-1801 Vol 2 Item:</u> IV.A2-9 <u>Table 1 Item:</u> 3.1.1.31 (No further evaluation required)

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		AMR results for those welds are presented in the LRA. b) Provide an expanded discussion on the use of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program for the detection of cracking due to SCC in the bottom mounted instrument guide tubes. Address which examinations specified in ASME Code Section XI are credited and their capability to detect cracking due to SCC in the bottom mounted instrument guide tubes before a through-wall leak occurs.	Note: E and Plant Specific Note 1 Component Type: RV Penetrations (Head vent pipe, flux Thimble Guide Tube Penetrations) Intended Function: PB Material: Nickel Alloys Environment: Reactor Coolant (Int) Aging Effect: Loss of Material AMP: Water Chemistry (B2.1.2) NUREG-1801 Vol 2 Item: IV.A2-14 Table 1 Item: 3.1.1.83 (No further evaluation required) Note: B (b) The discussion on the aging management for SCC in the bottom mounted instrument guide tubes is provided in the response to AMRA079.
AMRA081	3.1	In its response to audit question AMRA016, the applicant stated that LRA Section 3.1.2.2.16.1 will be revised to read, "These control rod drive mechanism housings are stainless steel for WCGS, therefore no additional commitments or further evaluation is required." Clarify whether there are any nickel alloy welds associated with the control rod drive mechanism housings. If there are nickel alloy	There are Alloy 82/182 welds of the control rod drive mechanism (CRDM) housings to the CRDM penetration tubes. Also, there are Alloy 82/182 J-groove welds of the CRDM penetration tubes to the lower surface of the vessel head. The aging management of these welds is evaluated as part of the CRDM penetration tubes that are made of nickel alloys and are addressed in LRA Table 3.1.2-1, page 3.1-44 as follows: Component Type: RV Control Rod Drive Head Penetration (CRDM tubes) Intended Function: PB Material: Nickel Alloys Environment: Reactor Coolant (Int) Aging Effect: Cracking

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		welds associated with the control rod drive mechanism housings, identify where the AMR results for those welds are presented in the LRA.	AMP: ASME Section XI ISI, Subsections IWB, IWC, and IWD (B2.1.1) Water Chemistry (B2.1.2) Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of PWR (B2.1.5) NUREG-1801 Vol 2 Item: IV.A2-9 Table 1 Item: 3.1.1.65 (No further evaluation required) Note: B Component Type: RV Control Rod Drive Head Penetration (CRDM tubes) Intended Function: PB Material: Nickel Alloys Environment: Reactor Coolant (Int) Aging Effect: Loss of Material AMP: Water Chemistry (B2.1.2) NUREG-1801 Vol 2 Item: IV.A2-14 Table 1 Item: 3.1.1.83 (No further evaluation required) Note: B
AMRA082	3.2	LRA Section 3.2.2.2.3.4 states that the Lubricating Oil Analysis and the One-Time Inspection Programs will manage loss of material due to pitting and crevice corrosion for copper alloys, copper nickel, and stainless steel components exposed to lubricating oil, except for the RCP lube oil leakage collection system. Explain how is loss of material managed in the RCP lube oil leakage collection	The Reator Coolant Pump (RCP) lube oil leakage collection system is part of Floor & Equipment Drains System. The aging management evaluation for Loss of Material in the RCP lube oil leakage collection system is addressed in LRA Table 3.3.2-17. The environment is oil leakage from the RCP motor that may be contaminated oil. As indicated in the Plant Specific Note 2 of LRA Table 3.3.2-17, Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22) manages Loss of Material on internal component surface exposed to contaminated oil environment instead of Lubricating Oil Analysis program (B2.1.23).

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		system.	The inspection of Internal Surfaces in Miscellaneous Piping and Ducting Component AMP (B2.1.22) will perform visual inspection during maintenance activities to manage Loss of Material of the internal surface of the RCP lube oil collection components.

Wolf Creek TLAA Audit Questions and Responses

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TLAAA001	4.3	LRA Section 4.3 states that, "The design number of each transient was selected to be somewhat larger than expected to occur during the 40 licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs." Clarify if the projections of future operations are based on innovations of system designs. Any "innovation" that has been included in the design basis became the CLB. Explain how WCGS can project future operation based on innovations in the system design that may or may not ever be developed.	The statement "The design number of each transient was selected to be somewhat larger than expected to occur during the 40 licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs." has been clarified in License Renewal Application (LRA) Section 4.3, to read: "The number of occurrences of each transient for use in the fatigue analyses was specified to be somewhat larger than the number of occurrences expected during the 40-year licensed life of the plant, based on engineering experience and judgment. This provides a margin of safety and an allowance for future changes in design or operation that may affect system design transients."
TLAAA002 **Closed to RAI 4.3-3.**	4.3.1	LRA Section 4.3.1 states that the present fatigue aging management program uses cycle counting and usage factor tracking to ensure that	**1. Describe how the fatigue aging management program tracks usage factor.** LRA Section 4.3.1.3 has been amended to describe how the fatigue usage factor at the monitored locations is tracked by one of two methods:

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TLAAA001	4.3	LRA Section 4.3 states that, "The design number of each transient was selected to be somewhat larger than expected to occur during the 40 licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs." Clarify if the projections of future operations are based on innovations of system designs. Any "innovation" that has been included in the design basis became the CLB. Explain how WCGS can project future operation based on innovations in the system design that may or may not ever be developed.	The statement "The design number of each transient was selected to be somewhat larger than expected to occur during the 40 licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs." has been clarified in License Renewal Application (LRA) Section 4.3, to read: "The number of occurrences of each transient for use in the fatigue analyses was specified to be somewhat larger than the number of occurrences expected during the 40-year licensed life of the plant, based on engineering experience and judgment. This provides a margin of safety and an allowance for future changes in design or operation that may affect system design transients."
TLAAA002 **Closed to RAI 4.3-3.**	4.3.1	LRA Section 4.3.1 states that the present fatigue aging management program uses cycle counting and usage factor tracking to ensure that	**1. Describe how the fatigue aging management program tracks usage factor.** LRA Section 4.3.1.3 has been amended to describe how the fatigue usage factor at the monitored locations is tracked by one of two methods:

Question No	LRA Sec	Audit Question	Final Response
		actual plant experience remains bounded by design assumptions and calculations reflected in the USAR. **1. Describe how the fatigue aging management program tracks usage factor.** **TLAAA002 (Follow-up #1)** **In its response, the applicant indicated that design basis transient data were used for the fatigue usage factor tracking and that this is conservative because it assumes that each actual transient is as severe as a design basis transient.** **During the audit, the staff reviewed basis document FP-WOLF-304 that indicates that actual plant transient data was used for the fatigue usage factor calculation from January 13, 1996 through December 31, 2005, and that the value was used to derive backward-projected initial**	For the period of extended operation, the Wolf Creek Generating Station (WCGS) fatigue monitoring program will use cycle-count-based monitoring for the first four locations listed in Table 4.3-2. These four locations are included among the six sample locations that will be monitored for the additional effect of the reactor coolant environment on fatigue usage, as discussed in Section 4.3.4, "Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components." Cycle-based monitoring assumes the alternating stress range (as defined in the ASME Code) of every cycle of a transient is equal to that of the design basis, worst-case events assumed by the code fatigue analysis. Accumulated fatigue usage is then the sum of the number of transient cycles times the per-cycle, design basis fatigue usage of each. The method uses event pairing as described in the ASME Code to define the bounding alternating stress range for the cycle comprised of the paired events. Events are paired in the same way as they are in the design stress report except that conservative adjustments are required when only part of an event pair has actually occurred. As cycles are accumulated, the cumulative fatigue usage (CUF) calculated by the fatigue monitoring program will be a conservative upper bound relative to the design analysis. If all transients specified in the design were to occur, then event pairing would be precisely as was done in the design analysis and calculated CUF would be the same as for the design analysis. For the period of extended operation the WCGS fatigue monitoring program will gather stress-based monitoring CUF data for the remaining 12 locations in Table 4.3-2 (24 when the steam generator feedwater nozzle locations are counted separately). These 12 locations include the three remaining locations monitored for the additional effect of the reactor coolant environment on fatigue usage, the hot leg nozzle connecting to the surge line, and the two charging nozzles. (These three nozzles comprise two locations in LRA Table 4.3-5 and in NUREG/CR-6260,

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		CUF prior to the available data. **The applicant's response is not consistent with the plant basis document.** **(1)a. Clarify the inconsistency and b. provide further discussion of the transient data.** **(2) Discuss the transient severity during the period from 1983 through 1996 to ensure that backward-projected initial CUFs are reasonable.** **TLAAA002 (Follow-up #2) (07-10-2007)** **In its response to TLAAA002, follow-up #1, question 2, the applicant defined Period 1 (without monitoring data) and Period 2 (with monitoring data). The applicant concluded, especially for the pressurizer surge line nozzles and components and components, that the**	which evaluate the two charging nozzles as a single location). Stress-based monitoring uses actual plant transient profile data to conservatively determine an estimate of the alternating stress range of monitored cycles between event pairs, from recorded pressure, temperature, flow, and rate-of-change data; using models based on the code fatigue analysis. The transfer function methodology used for the stress based monitoring calculations utilize a one dimensional stress parameter to conservatively estimate the stress range of cycles comprised of pairs of events that have actually occurred. The pairing of events is done in a way that creates the maximum amplitude cycles from the transient event history (pagoda rain flow analogy method). The one dimensional stress parameter model is defined such that the estimated alternating stress calculated by the monitoring program is greater than or equal to the stress range that would be calculated from the six components of the stress tensor by the methods outlined in the ASME Code. Fatigue usage accumulation is then calculated from this estimated stress range, for each cycle. The WCGS fatigue monitoring program will use cycle-based fatigue usage calculations for all locations until such time as an action level for accumulated fatigue usage is reached for a particular location. If this occurs, an acceptable corrective action is to enhance fatigue usage calculations using stress based data to confirm continued conformance to the Code limit. **TLAAA002 (Follow-up #1) Response** WCGS Responses to the TLAA002 follow-up questions to be addressed in RAI 4.3-3 response

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		use of the WCGS Period 2 data is conservative for Period 2, but realistic for Period 1. Provide quantitative data to justify this statement. Specifically, identify the portion of the Period 1 transient contribution to the CUF of 0.0584 described in LRA Table 4.3-5.	
TLAAA003	4.3.1	LRA Section 4.3.1.2 states that the usage factors calculated by the program include the effects of cycles incurred before the program was installed, in two periods. The LRA only describes one period, February 1982 through March 1992. **1. Clarify what is the time frame for the second period.** The applicant also states that effects were counted or estimated from the operating history for the period between initial cold hydro in 1982 to the installation of automated	**1. Clarify what is the time frame for the second period.** The first paragraph of LRA Section 4.3.1.2 describes two periods, (1) "...between initial cold hydro in February 1982 to the installation of the automated transient data acquisition system in March 1992," and (2) "...thereafter, up to the implementation of the fatigue management program." The fatigue management program was implemented in 1997. Transient cycles for the period before March 1992 was counted from historical plant records. Transient cycles from March 1992 until implementation of the fatigue management program in 1997 were counted by an early version of FatiguePro that started operating in March 1992. Cycle counts are available from data analysis reports compiled by this early version of FatiguePro. **2. Explain how the effects were estimated taking in consideration the operating history.** **3. Provide transient history cycle counting data prior to the installation of the automated system.** **4. Clarify whether the program uses cycle counting only or if it performs online stress evaluation and CUF calculations.**

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		transient data acquisition system in March 1992. **2. Explain how the effects were estimated taking in consideration the operating history.** **3. Provide transient history cycle counting data prior to the installation of the automated system.** 4. **Clarify whether the program uses cycle counting only or if it performs online stress evaluation and CUF calculations.** **TLAAA003 (Follow-up)** In its response, the applicant indicated that the program uses cycle counting only. However, the basis document indicates that actual plant transient data was used to track CUF. Please clarify this inconsistency.	Cycle counting data from February 1984 until implementation of the fatigue management program in 1997 was reconstructed from historical plant records and from an earlier version of FatiguePro. Fatigue cycle counts for the period February 1984 through March 1992 were reconstructed by Westinghouse (Westinghouse ICE-ICAT (97)-012 proprietary report Reference 1) by review of historical plant records. These records included control room logs, recorded instrument data from the plant computer system, event reports, and startup test reports. The numbers of the design transient cycles experienced in the period covered by the historical records review were tabulated (Reference 1) and imported into the fatigue management software program as baseline cycles) An early version of FatiguePro was operating for the period March 1992 until implementation of the Fatigue Management program in 1997. Although raw data files collected by FatiguePro from 1992 through 1995 were not preserved, the FatiguePro analyses, including cycle count results, are available and were used to determine fatigue cycles experienced during the period 1992 through 1995. Raw data files and FatiguePro analyses are available for the period 1996 until implementation of the Fatigue Management program and were used to determine fatigue cycles experienced during that period. The cycle counts for the period March 1992 until implementation of the fatigue monitoring program were added to the cycles counted from historical records to create a complete baseline cycle count as of the time of implementation of the fatigue monitoring program. With the exception of the NUREG/CR 6260 locations, which will be monitored for fatigue usage including the environmental effects of the reactor coolant during the period of extended operation, all original design basis fatigue analyses remain valid until such time as the specified

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			number of at least one type of transient is exceeded. Therefore, only cycle count data is needed to verify compliance with the component design bases until the allowed cycles are exceeded. Thus, for record purposes the WCGS fatigue management program uses cycle counting only until a corrective action limit on cycles is reached. Usage factors calculated by either cycle based or stress based methods, may be used as part of a corrective action plan that responds to reaching an action level for cycles. **TLAAA003 (Follow-up) Response** The FP-WOLF-304 basis document is a "Baseline Evaluation and 60-Year Projection." It demonstrates that the WCGS fatigue management program should be successful, but is does not include a detailed description of the program as going to be implemented. For the period of extended operation the WCGS fatigue management program will use cycle-count-based monitoring for the locations of LRA Table 4.3-2 line numbers 1 through 4, all of which are included among the six sample locations that will be monitored for the additional effect of the reactor coolant environment on fatigue usage, as discussed in Section 4.3.4. The WCGS fatigue management program will collect stress-based monitoring data for the remainder of the locations in LRA Table 4.3-2, including the three remaining locations monitored for the additional effect of the reactor coolant environment on fatigue usage (the hot leg nozzle connecting to the surge line, and the two charging nozzles). For the period of extended operation, the stress based fatigue monitoring data may be used to verify that accumulated fatigue usage, including the effect of the reactor coolant environment, remains below the Code limit of 1.0. The WCGS fatigue monitoring program will use cycle counts and cycle-based fatigue usage calculations for all locations until such time as an action level for accumulated fatigue usage is reached for a particular

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			location. If this occurs, an acceptable corrective action is to enhance fatigue usage calculations using stress based data to confirm continued conformance to the Code limit. References: 1. Miller, Teresa A. Westinghouse Report ICE-ICAT (97)-012. "Transient and Fatigue Cycle Monitoring Transient and Fatigue History Evaluation Report of Wolf Creek Nuclear Operation Corporation, Wolf Creek Plant." Proprietary. April 1998. 2. Structural Integrity Associates (SIA) Calculation Package FP-WOLF-304. "Baseline Evaluation and 60 Year Projection for Wolf Creek." Rev. 0. Contains Proprietary Westinghouse data. 25 May 2006.
TLAAA004	4.3	LRA Section 4.3 discusses thermal stratification transients that were not foreseen in the original design. Subsequently to these transients, the applicant performed significance evaluations and design specifications and analyses revisions. **1. Provide a summary of these transients and any revisions made to the design specifications.** **TLAAA004 (Follow-up)** **In its response, the**	**1. Provide a summary of these transients and any revisions made to the design specifications.** The two types of transients of concern that were not foreseen in the original design are related to thermal stratification of the fluid in the pressurizer surge line NRC Bulletin 88-11), and inflow and outflow (insurge and outsurge) of cooler RCS fluid through the surge line to and from the lower portion of the pressurizer (Westinghouse Nuclear Safety Advisory Letter NSAL-04-5). LRA Section 4.3.2.8, "Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification," discusses the changes to the pressurizer surge line analysis in response to Bulletin 88-11. Section 4.3.2.7, "ASME Section III Class I Piping and Piping Nozzles," notes that these effects are included in the current code analysis of the hot leg surge nozzle. Section 4.3.2.4, "Pressurizer and Pressurizer Nozzles," discusses these effects in the pressurizer surge nozzle, and includes a discussion of the continuous-spray operating changes, which affects surge line stratification and

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		applicant states that surge line weld overlays were installed during Refueling Outage 15. Pressurizer nozzles have high CUFs. The application of weld overlays increases the wall thickness; therefore, increasing the fatigue usage factor. **2. Discuss the fatigue impact on the pressurizer nozzles due to the application of weld overlays.**	insurge/outsurge transients. See also the response to TLAAA011, on Section 4.3.2.4. The WCGS site piping design specification, Westinghouse 955238, Rev. 2, Appendix D – "Fluid System Transients," includes the statement "Due to thermal stratification consideration for the surge line, the associated thermal transients are shown in Section 2.1." Section 2.1 is references, which include 2.1.8 WCAP-12893 to define the transients. (A typographical error gives the WCAP number as "12873." However, the title and date of issue are correct for WCAP-12893 so the intent is clear.) NRC Bulletin 88-11 is also included as a reference document (specification paragraph 2.2.5) to define the surge line stratification concern. The most recent revisions of the Class 1 stress reports for the surge line and nozzles have been conformed to this specification. An additional concern regarding stratified fluid conditions in the surge line and pressurizer is insurge/outsurge transients (Westinghouse Nuclear Safety Advisory Letter NSAL-04-5). These transients occur when colder water from the reactor coolant system (RCS) flows into the pressurizer (insurge) and are subsequently expelled by hotter water from the pressurizer flowing out through the surge line (outsurge). These transients have a more severe effect on the pressurizer surge line nozzle and pressurizer lower head than on the surge line pipe. Because water entering the pressurizer through the surge line does not mix with the water in the pressurizer, an insurge creates a stratified condition in the lower head of the pressurizer with cooler water at RCS temperature below warmer water at pressurizer saturation temperature. The insurge/outsurge combination produces a temperature cycle in the portion of the pressurizer wall that is cooled by the insurging RCS fluid and then heated by the outsurging pressurizer fluid. The effects of insurge/outsurge transients are not significant during power operation when the temperature difference between the pressurizer and the RCS is

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			small. The significant fatigue effects of insurge/outsurge transients occur during plant heatup and cooldown when the temperature difference between the pressurizer and the RCS can be large. Westinghouse has performed a generic analysis of the fatigue effects of insurge/outsurge transients (WCAP- 14950), which shows that fatigue usage for the specified number of heatup/cooldown transients, including the effects of postulated insurge/outsurge events is less than the ASME Code allowable. A design document change notice (DDCN) has been issued for the Wolf Creek site pressurizer specification (952575, Rev. 6) requiring insurge/outsurge transients, as defined in WCAP-14950, to be included in the pressurizer design analysis. The pressurizer design stress report has been amended to conform to the amended specification. **TLAAA004 (Follow-up) Response** **2. Discuss the fatigue impact on the pressurizer nozzles due to the application of weld overlays.** The surge line weld overlay installed during Refueling Outage 15 covers the nozzle-to-safe-end weld, the safe end, and the safe-end-to-pipe weld. The weld overlay extends beyond the nozzle to safe end weld toward the pressurizer until it blends into the tapered thickness transition of the nozzle. The overlay extends beyond the pipe to safe end weld onto the pipe for a distance of several pipe wall thicknesses. Thus, the ends of the overlay are sufficiently far from the original welds to be unaffected by the stress intensification of the weld. The preliminary analysis of this overlay demonstrates that the maximum peak stresses in the portions of the pipe and nozzle immediately adjacent to the overlay are at the ends of the overlay and are no greater than the peak stresses previously calculated for the nozzle-to-safe-end and safe-end-to-pipe welds. Therefore, the calculated fatigue usage at the current highest stress locations adjacent to the overlays is no greater than was calculated for the original welds.

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			The maximum fatigue usage for the pressurizer surge line nozzle occurs in the thick part of the nozzle at the nozzle to vessel transition (surge nozzle knuckle). The stresses causing fatigue usage at that location are principally from temperature gradients. The fatigue usage factors of the nozzle-to-safe-end and safe-end-to-pipe welds are no longer the basis of a safety determination, because the reliability of these welds will be verified by periodic inspections and by flaw propagation analyses that are not TLAAs. The information on this weld overlay modification and analysis was incomplete at the time the LRA was originally filed. This information was included in LRA Amendment 1 Section 4.3.2.4 (page 4.3-20). The LRA Amendment 1 paragraph "Effect of a Pressurizer-Surge-Nozzle-to-Safe-End Weld, Safe End, and Safe-End-to-Surge-Line Weld Overlay" will be amended to conform to the TLAAA004 (Follow-up) Response, to read: A weld overlay was installed over the surge-nozzle-to-safe-end weld, safe end, and safe-end-to-pipe weld during Refuel 15. The overlay extends beyond the nozzle-to-safe-end weld toward the pressurizer until it blends into the tapered thickness transition of the nozzle. The overlay extends beyond the safe-end-to-pipe weld onto the pipe for a distance of several pipe wall thicknesses. Therefore, the ends of the overlay are sufficiently far from the original welds to be unaffected by the stress intensification of the weld. The fatigue usage factors of the nozzle-to-safe-end and safe-end-to-pipe welds are no longer the basis of a safety determination, because the reliability of these welds will be verified by periodic inspections and by flaw propagation

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			analyses that are not TLAAs. The maximum fatigue usage in the surge nozzle is at a location inside the nozzle inner radius. The overlay did not require a revision to the fatigue analysis at this location. The fatigue analysis of this location remains a TLAA, and fatigue in this location will continue to be monitored. References: 1. Westinghouse Nuclear Safety Advisory Letter NSAL-04-5. "Pressurizer Insurge-Outsurge Transients." Pittsburgh: Westinghouse Electric, 26 August 2004. 2. Westinghouse Design Specification 955238 Rev. 2. "Standardized Nuclear Unit Power Plant System (SNUPPS), Piping Design Specification, ANS Safety Class 1 ... RCS, SIS, RHRS, CVS." Westinghouse Proprietary. Pittsburgh: Westinghouse Electric Corporation Nuclear Energy Systems, 8 December 1995. Amends Reference 11 (See Section 1.0). 3. WCAP-12893. M. A. Gray et al. "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification." Rev. 0. Westinghouse Proprietary Class 2. Pittsburgh: Westinghouse, March 1991. Not a code design report. 4. WCAP-14950. M. A. Gray et al. Westinghouse Report. Mitigation and Evaluation of Pressurizer Insurge-Outsurge Transients. Westinghouse Proprietary Class 2C. February 1998. 5. Westinghouse Specification 952575 Rev. 6. "Pressurizer, Addendum to Design Specification 955285 Rev. 0, Standardized Nuclear Unit Power

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			Plant System (SNUPPS)." Westinghouse Proprietary. Pittsburgh: Westinghouse Nuclear Energy Systems, 21 December 1992 [WCGS DocNo M-713-00004 W06 and DDCN M-713-00004-W06-01]. Includes changes for rerating, steam generator tube plugging, Thot reduction, and insurge/outsurge transients.
TLAAA005	4.3.1	**Revised TLAAA005 (6/27/2007)** LRA Table 4.3-5 was evaluated based on the estimated cycles to 60-year EOL described in the original LRA Table 4.3-1. The LRA Table 4.3-1 estimated cycles assumes that some design transients will never occur. However, those design transients have occurred in other nuclear power plants. For example, in its LRA, Shearon Harris, Unit 1, stated that it had experienced the following design transients within the first 18 years of operation: · inadvertent reactor coolant system depressurization · reactor trip cooldown with	**a) Explain what actions will be taken if any of these design transients not considered do occur.** The fatigue monitoring programs, both cycle counting and stress based fatigue data collection, track all significant specified design transients by either automated data acquisition or manual entry of events. Therefore, if any of the low probability events such as inadvertent RCS depressurization, reactor trip with safety injection, excessive feedwater flow, or other severe transients do occur they will be counted, fatigue usage evaluated, and added to cycle counts and accumulated fatigue usage. LRA Amendment 1, Section 4.3.1.3 discusses action limits and corrective actions to be established on cycle counts and calculated cumulative fatigue usage to ensure limits will not be exceeded. For cycle counts, an action limit will be established that requires corrective action when the cycle count for any of the critical thermal or pressure transients is projected to reach a high percentage (e.g., 90%) of the design-specified number of cycles before the end of the next operating cycle. In order to assure sufficient margin to accommodate occurrence of a low probability transient, corrective actions must be taken before the remaining number of allowable occurrences for any specified transient becomes less than 1. For example, the specified number of accumulator safety injection events is 4 so corrective actions would be required when 75% (3) of the specified cycles have occurred.

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		safety injection · reactor trip cooldown without safety injection · inadvertent safety injection · excessive feedwater flow The CUFs for the NUREG/CR-6260 locations at WCGS were evaluated without considering these transients and other anticipated operational occurrences. Eliminating anticipated operational occurrences do not provide conservative margin to ensure that the CUFs for the reactor coolant pressure boundary are not exceeded during the period of extended operation. **a) Explain what actions will be taken if any of these design transients not considered do occur.** **TLAAA005 (Follow-up)** b) **In its response to item (a) the applicant explained, with several examples, the**	For calculated cumulative fatigue usage an action limit will be established that requires corrective action when calculated cumulative usage factor (CUF) for any monitored location is projected to reach 1.0 within the next 2 or 3 operating cycles. In order to assure sufficient margin to accommodate occurrence of a low probability transient, corrective actions must be taken while there is still sufficient margin to accommodate at least one occurrence of the worst case (highest fatigue usage per cycle) design transient event. For example, if inadvertent RCS depressurization, when adjusted for the environmental effects of the reactor coolant system, at a NUREG/CR-6260 location, causes 20% of the total allowable fatigue usage, corrective action for that location would be required before calculated usage (including environmental effects factor, F_{en}) reached 0.8. Fatigue management program procedure revisions to implement corrective actions will include requirements that action limits be established such as to assure that corrective actions are taken while there is still sufficient remaining margin to experience at least one cycle of the worst case specified design transient without exceeding a CUF value of 1.0. For NUREG/CR-6260 locations, CUF calculation will be done using the appropriate F_{en} environmental factor. **TLAAA005 (Follow-up) Response** The surge line hot leg nozzle (SL HL nozzle) and the accumulator/RHR cold leg safety injection nozzles (SI nozzles) are locations that are evaluated for the environmental effects of the reactor coolant in accordance with NUREG/CR-6260. Fatigue usage for the SL HL nozzle is tracked by stress based monitoring. Fatigue usage for the SI nozzles are tracked by cycle based monitoring. Cumulative fatigue usage with the application of appropriate environmental factors (F_{EN}) must remain less than the ASME Code limit of 1.0 unless another fatigue management approach is used for the location.

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		action limits and corrective actions to be established on cycle counts and CUF calculations to ensure that the ASME Code limits will not be exceeded. • **Provide specific steps or data on the surge line hot leg nozzle and the accumulator/RHR cold leg safety injection nozzles, described in LRA Table 4.3-5, to demonstrate ASME Code compliance.** • **Provide the action limits on these tow components and specify whether the action limits are design or projection based.**	Because F_{EN} factors are applied for these locations, cycle count action limits are not sufficient to assure that ASME Code limits are satisfied, because CUF (with F_{EN}) will exceed 1.0 well before all of the specified design transient cycles have been experienced. Therefore, corrective action limits based on CUF are applicable to these locations. The WCGS fatigue management program provides for periodic evaluation (once per fuel cycle) of actual accrued fatigue usage. This actual accrued usage is based on the historical plant experience; it is neither a design value nor a projection. For the SL HL nozzle, this CUF is calculated from data for actual plant transients using the stress based models. For the SI nozzles, the CUF is calculated from the accrued transient cycles and the fatigue usage per cycle calculated by the design stress report fatigue analysis of record assuming that the transient severity is as specified in the component design specifications. In order to apply the fatigue usage action limits, CUF must be projected at the ends of one, two, and three additional operating cycles to determine if the action limit has been reached. These short term predictions will be based on extrapolation of CUF accumulation to the date of evaluation starting from a reliable baseline CUF. If an action limit has been reached, corrective actions will be taken in accordance with the Wolf Creek Fatigue Management and Corrective Action Programs. The reason for establishing the corrective action limit at 2 or 3 fuel cycles before the CUF limit is reached is to allow time for appropriate corrective action to be accomplished. An additional consideration that must be applied in the evaluation of whether a corrective action limit has been reached is that margin must be maintained to allow one cycle of the highest fatigue usage per cycle transient to occur without exceeding CUF (with F_{EN}) = 1.0. This consideration may require that corrective action be taken more than 2 or 3

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			fuel cycles before CUF (F_{EN}) is projected to exceed 1.0. This is because the projections will be based on historical experience, which is not expected to include many of the low probability design transients. To implement this addition to the corrective action limit development, fatigue usage, with $F_{EN,}$ per cycle (ΔCUF (with F_{EN})) must be calculated for each of the low probability transients, for each location (SL HL nozzle and SI nozzles). For each location, 1.0 – CUF (with F_{EN}) at the time of evaluation must be greater than the largest ΔCUF (with F_{EN}) calculated for the low probability transients. For this evaluation, the low probability design transients to be used in the evaluation will include: • Aux. Spray Actuation, Spray Water Diff.>320F • Excessive Feedwater Flow • Reactor Trip – Cooldown with no SI • COMS • Reactor Trip – No Inadvertent Cooldown with Turbine Over-speed • Reactor Trip - Cooldown with SI • Inadvertent RCS Depressurization • Accumulator Safety Injection • Operating Basis Earthquake The above list includes only transients specified by component design specifications and not transients more severe than postulated in the design basis or licensing basis, or transients that are more severe than allowed by plant procedures (e.g., a surge line stratification transient with a pressurizer to RCS temperature difference greater than that allowed by plant procedures).
TLAAA006	4.3.1	LRA Table 4.3-1, footnote 4, states that the recorded	**1. Clarify what is the definition of the terms "very slow cooldowns" and "significant cooldown." Explain why very slow cooldowns do**

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		transients include successive heatups without intervening cooldowns, indicating that the difference between the number of heatup and cooldown cycles is due to very slow cooldowns not counted as significant cooldown transients. 1. Clarify what is the definition of the terms "very slow cooldowns" and "significant cooldown." Explain why very slow cooldowns do not count as cooldown cycles. **TLAAA006 (Follow-up)** In its response, the applicant states that a temperature change of less than 150°F at any rate produces no fatigue usage. The staff understands that the stress change for a carbon steel component with a temperature step change of 150°F could be as high as 22.50 ksi [$E\alpha\Delta T/2(1-\mu)=(30E6)(7E-6)(150)/2(1\ 0.3)$]. Thus, a ±150°F step	**not count as cooldown cycles.** The LRA footnote has been amended to indicate that the difference in the number of heatup and cooldown cycles occurs because either additional heatup or cooldown cycles can be counted to account for special circumstances, such as prolonged holds at a constant intermediate temperature. See LRA Amendment 1, Table 4.3-1, Note 1. For the computerized cycle counting system, "A Reactor Coolant System (RCS) heatup has occurred if the cold leg water temperature (CLETEMP) increases by more than 150°F from the previous cooldown condition and stays above that value for more than five minutes." Similarly, "An RCS cooldown has occurred if the cold leg water temperature CLETEMP decreases by more than 150°F from the previous heatup condition and remains below that value for more than five minutes." The reason for the 150°F criterion is to prevent counting as heatup/cooldown events partial heatups that are aborted after a small temperature increase has occurred. A temperature change of 150°F at a controlled rate no greater than 100°F/hr produces no fatigue usage by itself, and it is unlikely that another transient that needs to be combined with the aborted heatup will occur before completion of a normal heatup to hot standby because most specified transients start from a power operation condition. In addition to the computer-generated heatup and cooldown events, there have been a few cases where events were added manually to the cycle counting database to conservatively account for special circumstances. One of the manual additions was made for a heatup from 380°F to normal operating temperature (NOT) starting 1/09/1992 following a 3-day hold at 380°F. Because there was no cooldown during the hold, the computer algorithm would count the entire heatup from ambient to normal operating temperature as a single heatup. Because of the prolonged hold at one temperature, thermal gradients and induced thermal stresses produced in the piping and components by the heatup to 380°F would have largely

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		change will cause a 45.00 ksi stress difference. The staff understands that screening out transients with a stress difference of 22.50 ksi in the CUF evaluation is not acceptable because most transients have a temperature difference less than 150°F. If screening out 150°F would be acceptable, then there would not be a need to monitor most of the transients. Please revise the response related no fatigue usage.	disappeared. Thus, resumption of the heatup constitutes an additional cycle, which is conservatively included in the cycle counting database as a full RCS heatup. This manual addition created an extra heatup cycle not associated with a cooldown cycle. Both the computer algorithm and the manual additions to the cycle counting database can result in an imbalance between the numbers of heatup and cooldown cycles when heatups or cooldowns are interrupted for long periods due to unusual circumstances. In general, both the computer algorithm and manual reviews will conservatively add cycles to the database. **TLAAA006 (Follow-up) Response** A step function temperature change would produce a skin stress on the wetted surface in excess of the fatigue endurance limit. The 150°F temperature change criteria only apply to heatup/cooldown cycles, which by definition are not temperature step functions. The FatiguePro program criteria for automated identification and counting of transient cycles are generally specific to the type of transient being considered. Reference: 1. Structural Integrity Associates (SIA) Report SIR 95 043. "Cycle Counting and Cycle-Based Fatigue Methodology Report, Transient and Fatigue Monitoring System for Callaway/Wolf Creek." Rev. 2, 21 January 1997.
TLAAA007	4.3.1	LRA Table 4.3-2 lists estimated 60-years CUF values for the pressurizer surge line nozzle and pressurizer surge line as 0.01168 and 0.00003,	The pressurizer surge line nozzle is at the pressurizer end of the surge line. The surge line hot leg nozzle is at the RCS hot leg end of the surge line. The apparent difference arises because of the generic nomenclature used by NUREG 6260 to describe the locations evaluated in Table 4.3-5,

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		respectively. However, LRA Table 4.3-5 lists the estimated CUF for the surge line hot leg nozzle as 0.05849. Clarify the difference.	versus the more-exact Wolf-Creek-specific descriptions in Table 4.3-2. The pressurizer surge line nozzle and pressurizer surge line locations (Table 4.3-2 items 12 and 13, respectively) do not appear in Table 4.3-5. The "Hot Leg Surge Line Nozzle," Item 7 of Table 4.3-2, U60 = 0.0585, is the same location as the "Surge Line Highest CUF Location, Hot Leg Nozzle," in Table 4.3-5, U60 = 0.05849. These U (60) values are consistent. Although Table 4.3-5 identifies this location as the "Surge Line Highest CUF Location, Hot Leg Nozzle," it is not in fact included in the surge line Class 1 analysis, but in the Class 1 main loop nozzle analysis. The usage factor projections have been eliminated from Table 4.3-2 in LRA Amendment 1.
TLAAA008	4.3.1	LRA Section 4.3.1.2 states that "Since these locations were chosen to represent the highest usage factors in the Class 1 components and piping systems, these estimates demonstrate that the 60-year period of extended operation should not produce fatigue usage factors greater than 1.0." Explain in detail the meaning of this statement. Clarify if WCGS is certain that the	This is an assumption of the WCGS fatigue management program, supported by the selected sample of monitored locations. The sample includes locations specified by the licensing basis, USAR Table 3.9(N)-13, as cited by Technical Specification 5.5.5 and USAR 3.9(N).1.1. The LRA has been amended to explain that the monitored locations were chosen to represent limiting usage factor locations in the Class 1 components and piping systems, and that (with the one exception explained in Section 4.3.4) they include those under the NUREG/CR-6260 program to monitor fatigue usage factors including effects of the reactor coolant environment. The cycle count projections have been eliminated from Table 4.3-1, and the text has been amended accordingly.

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		CUF will not exceed 1.0 or if this is an assumption.	
TLAAA009	4.3.1	The enhanced corrective action limits described in LRA Section 4.3.1.3 state that "Corrective action will be initiated if the periodic evaluation prescribed by the program indicates that a cumulative usage factor (CUF) limit might be exceeded in the next operating cycle." Justify if there will be sufficient time to take appropriate and timely corrective actions if periodic evaluation prescribed by the program indicates that the CUF limit might be exceeded in the next operating cycle. Clarify what are the definitions of the CUF limits for initiating corrective actions.	Allowance of sufficient time for corrective action is a criterion for these action limits, which are under development for the extended licensed operating period. The time constraints and their bases cannot be described in detail in advance of these action limits. LRA section 4.3.1.3 has been amended to further describe the basis for these action limits, including time constraints. **Enhanced Corrective Action Limits and Corrective Actions** The WCGS fatigue management program provides for periodic evaluation (once per fuel cycle) of fatigue usage and cycle count tracking of critical thermal and pressure transients to verify that ASME Code CUF limit of 1.0 and other CUF design limits will not be exceeded. The program will be enhanced to specify acceptable corrective actions to be implemented to ensure that design limits are not exceeded. These enhancements will include action limits for accrued transient cycles or CUF that require initiation of corrective actions, allowing sufficient time to effectively address the issues. For WCGS locations identified in NUREG/CR-6260 and described in Section 4.3.4, "Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components," this action limit will be based on accrued fatigue usage calculated with the F_{EN} factors required for including effects of the reactor coolant environment for Period 1, Period 2, and beyond. **Cycle Count Action Limit and Corrective Actions** An action limit will be established that requires corrective action when the cycle count for any of the critical thermal and pressure transients is

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			projected to reach a high percentage (e.g., 90%) of the design specified number of cycles before the end of the next operating cycle. If this action limit is reached, acceptable corrective actions include: 1. Review of fatigue usage calculations. • To determine whether the transient in question contributes significantly to CUF. • To identify the components and analyses affected by the transient in question. • To ensure that the analytical bases of the leak-before-break (LBB) fatigue crack propagation analysis and of the high-energy line break (HELB) locations are maintained. 2. Evaluation of remaining margins on CUF based on cycle-based or stress-based CUF calculations using the WCGS fatigue management program software. 3. Redefinition of the specified number of cycles (e.g., by reducing specified numbers of cycles for other transients and using the margin to increase the allowed number of cycles for the transient that is approaching its specified number of cycles). **Cumulative Fatigue Usage Action Limit and Corrective Actions** An action limit will be established that requires corrective action when calculated CUF (from cycle based or stress based monitoring) for any monitored location is projected to reach 1.0 within the next 2 or 3 fuel cycles.

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			If this action limit is reached acceptable corrective actions include: 1. Determine whether the scope of the monitoring program must be enlarged to include additional affected reactor coolant pressure boundary locations. This determination will ensure that other locations do not approach design limits without an appropriate action. 2. Enhance fatigue monitoring to confirm continued conformance to the code limit. 3. Repair the component. 4. Replace the component. 5. Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded. 6. Modify plant operating practices to reduce the fatigue usage accumulation rate. 7. Perform a flaw tolerance evaluation and impose component-specific inspections, under ASME Section XI Appendices A or C (or their successors), and obtain required approvals by the NRC. These corrective actions are equally applicable to the WCGS NUREG/CR-6260 locations described in Section 4.3.4, "Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components," including consideration of the effects of the reactor coolant environment. An additional consideration in establishing corrective action limits is to

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			assure that corrective actions are taken while sufficient margin remains to allow at least one occurrence of the worst case (highest fatigue usage per cycle) low probability transient that is included in design specifications, without exceeding Code limits. (See response to TLAAA005)
TLAAA010	4.3.1	The enhanced corrective action limits described in LRA Section 4.3.1.3 state that the period of extended operation will require two sets of corrective action limits to maintain the basis of safety determinations supported by fatigue analyses: (a) For the first set, the applicant states "If the monitoring program indicates that these calculated values are exceeded, the worst-location usage factor assumed by the primary loop LBB analysis may be exceeded and its basis no longer valid..." **Explain why the primary loop LBB analysis is related to the worst-location usage factor.** (b) For the second set, the	The LRA Section 4.3.1.3 description of enhanced corrective action limits has been amended. The amended description discriminates between action limits based on the design number of transient event cycles (rather than a reduced usage factor allowable), and the less-restrictive limit based on the code fatigue usage allowable of 1.0. **(a) Explain why the primary loop LBB analysis is related to the worst-location usage factor.** In response to this question WCGS has reviewed the WCAP-10691 LBB evaluation and finds that the LRA description requires clarification, in that the application to WCGS of the conclusion of the generic-plant LBB evaluation does not depend on the calculated worst-case fatigue usage factor in the WCGS primary coolant loop. It does, however, depend on maintaining transient cycle severity, and the number of transient events, within the bounds of the WCGS design basis, and therefore within the bounds of the generic fatigue crack growth analysis. The LBB evaluation applies only to the primary coolant loop piping. The LBB evaluation is supported by an evaluation of fatigue crack growth effects applicable to the limiting-case generic plant. The limiting-case generic plant evaluation evaluates growth of cracks at "a typical location" assuming design basis applied loads for the assumed set of design basis cycles. For application to WCGS this LBB evaluation makes no direct comparison between the limiting-plant fatigue crack growth evaluation and results of the WCGS Class 1 analysis, other than as may be inferred from the description of the separate crack stability analysis [Ref. 1 Sections 3.0 and 4.0]. The description of the separate crack growth

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		applicant states that "The second is some fraction of the code acceptance criterion of 1.0 for each location." and that "The fraction of 1.0 used may vary from one monitored location to another, and should be consistent with the expected usage factor accumulation rate for each location." The applicant further stated that "... these second action limits will be reached no earlier than the first." **Provide a definition for the term "expected usage factor accumulation rate." Explain why the second action limits will be reached no earlier than the first.**	stability analysis showed that comparable stresses at the most limiting primary loop location are less at WCGS. Therefore, the fatigue crack growth results will be less at comparable locations throughout the primary loop, for the same set of design event cycles. Therefore, the first action limit for LBB is the point at which the WCGS applied loads and number of cycles indicate that the generic analysis might no longer bound the WCGS case; that is, when the WCGS fatigue management program determines that the design basis number of cycles for an event tracked by the program might be exceeded (within an acceptable time limit to allow for corrective action, such as an operating cycle, or within an equivalent percentage of the design basis cycle count limit). The statement in the original LRA, that the LBB fatigue crack growth evaluation was performed "...at a worst-case location (i.e., with the highest alternating stress range)..." has been amended to state that the generic LBB fatigue crack growth evaluation was performed "at a typical location." [—as in Ref. 1 Section 6.0] The Disposition has also been amended to omit "Validation, in accordance with 10 CFR 54.21(c)(1)(i)." The disposition now depends only on the WCGS fatigue aging management program, in accordance with 10 CFR 54.21(c)(1)(iii). **(b) Provide a definition for the term "expected usage factor accumulation rate." Explain why the second action limits will be reached no earlier than the first.** For cycle-based fatigue monitoring the "expected usage factor accumulation rate" (or rate of increase in usage factor) is based on the sum of the products of cycle (or event pair) accumulation rates from historical data, times they're expected mean usage factors per cycle (or

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			event pair). For stress-based fatigue monitoring the expected usage factor accumulation rate is based on historical data on measured event severity as well as transient event frequency. For details of the projection methods, see the responses to TLAAA002, TLAAA003 and TLAAA005. The statement that "...these second action limits will be reached no earlier than the first" has been omitted from the amended LRA Section 4.3.1.3. However, in almost all cases the stress-based action limits will be reached no earlier than the cycle count limits, because (1) they are based on the code limit of 1.0, (2) the cycle count action limits limit the cumulative usage to the calculated lifetime usage factor at monitored locations, which is always no more than 1.0, and (3) the same criterion to provide sufficient time for corrective action, discussed in the response to TLAAA009, applies equally to both cases. (This timeliness criterion may however be applied differently, as described in the amended description of corrective action limits and corrective actions in LRA Amendment 1 Section 4.3.1.3.) Reference 1. WCAP-10691. S. A. Swamy, Y. S. Yee, R. A. Holmes, and H. F. Clark, Jr. "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Callaway and Wolf Creek Plants." Westinghouse Proprietary Class 2. Pittsburgh: Westinghouse, October 1984.
TLAAA011	4.3.2	LRA Section 4.3.2.4 states that use of continuous spray during heatup and cooldown prevent thermal stratification. Provide operating data that demonstrate that thermal stratifications are eliminated.	Thermal stratification has not been entirely eliminated, but fluid instabilities that occur during thermal stratification, and the resulting cyclic thermal stresses, have been minimized. The current analysis of record was based on conditions monitored before continuous spray was adopted, and is therefore conservative, since the adoption of continuous spray further minimizes these cyclic effects.

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			The statement "to prevent thermal stratification" has been amended to "to minimize thermal stratification" in LRA Amendment 1, Section 4.3.2.4 (page 4.3-20). Monitoring of the pressurizer surge line was performed at WCGS using temporary sensors to support the WOG investigation of surge line thermal stratification effects. Data from these measurements were used to develop transients for surge line stratification for use in analyses reported in WCAP-12893 "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification." These measurements were made before WCGS adopted modified operating procedures (MOP) to maintain continuous outflow from the pressurizer during plant heatup and cooldown. The instrumentation used to make these measurements is no longer installed on the surge line. WCAP-12893 evaluated the effects of surge line stratification on piping and nozzle stresses and fatigue usage using the transients developed specifically for WCGS and Callaway, which were in part based on the monitoring results for heatup and cooldown cycles without MOP. MOP, which create a continuous outflow from the pressurizer whenever the temperature difference between the pressurizer and the RCS is large do not prevent stratification, but reduce or eliminate cycling of the stratified condition minimizing fatigue cycles. Thus, the fatigue usage calculated in WCAP-12893, which takes no credit for MOP, is conservative for the current operation of WCGS. The fatigue usage calculations from WCAP-12893 have been incorporated in the latest revision of the primary system auxiliary piping stress report (WCAP-9728, Vol IV, Rev. 2).
TLAAA012	4.3.2	LRA Section 4.3.2.5	For steam generators, the 10 percent tube plugging assumed by the

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		describes that the applicant uses both 10 and 15 percent steam generator tube plugging in its steam generator fatigue analyses. Explain the difference.	power rerate evaluations has been superseded by the current analysis of record, which includes up to 15 percent plugging. LRA Amendment 1 clarified these paragraphs of the LRA.
TLAAA013	4.3.2	LRA Section 4.3.2.11 states that an evaluation made by Westinghouse found a large increase in the crossover and cold leg stresses at the reactor coolant pump, but since original stresses were low the effects on stresses and usage factors would not affect code compliance or the conclusion of the LBB analysis. The staff understands that a large stress increase causes the allowable flaw length to decrease in the LBB analysis. Clarify if the updated LBB analysis considered this "large increase in stress" and if the LBB was redemonstrated.	"The loop leak-before-break (LBB) evaluation was reviewed for the additional loadings due to column tilt. The largest increase in moment loading was the RCP outlet nozzle. This location was not a critical location in the LBB evaluation and did not become a critical location even with the increase in loading. All 12 weld locations in the primary loop were reviewed for the new thermal loadings, and acceptable margins were maintained" [Westinghouse SAP-94-178]. Reference: 1. Westinghouse Letter Report SAP-94-178. Michael C. Bollingbach, Westinghouse Power Systems Field Sales; to K. S. Parthasarathy, WCNOC. "Wolf Creek Nuclear Operating Corporation, Wolf Creek Generating Station, Wolf Creek RCP Column Tilt Evaluation." 12 October 1994 [Copy available attached to WCNOC ITIP 02872].
TLAAA014	4.3.4	LRA Section 4.3.4 states that the "normal" and "alternate"	For some years the two charging paths were used unequally, resulting in a faster accumulation of usage factor in the normal nozzle. These

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		charging nozzles have equal calculated usage factors. However, LRA Table 4.3-5 lists two different expected CUF values. Clarify the inconsistency.	estimates are the result of changes to equalize usage for the remainder of the design life.
TLAAA015	4.3.4	The Fen value is a function of oxygen content. Clarify if the Water Chemistry Program controlled oxygen content in the past 20 years of operation.	The Wolf Creek Water Chemistry AMP controls oxygen in the reactor coolant system (RCS) and pressurizer to less than 5 ppb (AP 02-003 sections 6.29 & 6.31). WCGS relies on and is consistent with the EPRI guidelines for Primary Water Chemistry (see AMP B2.1.2). WCGS has controlled dissolved oxygen in the RCS to the 5 ppb level since plant startup. The F_{en} dependence on dissolved oxygen is a constant for both stainless and alloy steel for oxygen concentrations less than 50 ppb (0.05 ppm) (Ref. 1, Ref. 2). The only circumstance that would allow the dissolved oxygen level in the RCS during operation to exceed 50 ppb is loss of hydrogen overpressure. Loss of hydrogen overpressure has never occurred at WCGS during operation. 1. NUREG/CR5704, Argonne National Laboratory Report ANL-98/31. "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels." Washington: US NRC, April 1999. 2. NUREG/CR-6583, Argonne National Laboratory Report ANL-97/18. "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels." Washington: US NRC, March 1998.
TLAAA016	4.3.6	LRA Section 4.3.6 states that "Since the remaining plant life from the present to the end of	WCGS has experienced no earthquakes of detectable magnitude since start of plant operation. A significant OBE or significant earthquake would be defined as an earthquake producing sufficient ground acceleration to

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		the period of extended operation (2006 to 2045) is less than that of the original license to which the numbers of OBE and SSE events apply, and since no SSE or significant OBE has occurred, these analyses remain valid for the period of extended operation." Define the term "significant OBE."	trigger the free field Strong Motion Accelerometer (SMA). The trigger actuated level is adjustable over a minimum range of 0.01 g to 0.03g. (See USAR paragraph 3.7(B).4.1b). If the trigger level is exceeded, seismic switches are closed to activate a plant annunciator in the control room indicating a possible seismic event. USAR 3.7(B).4.3 states, "Following a seismic event, all accessible data will be processed for an initial determination of the earthquake level." No actuation of the SMA triggers attributable to an earthquake has occurred at WCGS to date. LRA Amendment 1 deleted the word "significant."
TLAAA017	4.3.7	LRA Section 4.3.7 states that a cumulative usage factor was calculated and compared to a fatigue curve and the usage factor was based on tests of typical designs to failure. Clarify which fatigue curve the LRA refers to. Explain how the usage factor was determined based on tests of typical designs to failure.	This "fatigue curve" and "usage factor" bears no meaningful relation to the same terms as usually understood in mechanical design. The "fatigue curve" used was the test-to-failure curve of the component described (Power Strut Welded-Fillet PS608 angle fittings), with cycles multiplied by 1.5 for conservatism. As explained in the last two paragraphs, the "usage factor" of 0.9 is simply the very conservative allowed 900 maximum-deflection cycles over the assumed allowable, 1000 cycles. In fact, as stated, the allowed deflection was less than the indicated endurance limit of the "fatigue curve," so that a much larger – or infinite – number of allowable cycles could have been used. LRA Amendment 1 changed the statement "900 actual/1000 allowable" to "900 assumed/1000 allowed."
TLAAA018	4.6	The Loading Condition V discussion in LRA Section 4.6 states "Table 4.3-1, Item 1 shows only 27 startup cycles in the 19 years through 2004,	LRA Amendment 1 removed the 60-year projection column from Table 4.3-1. LRA Section 4.6.2 will be amended to provide the following estimate, and analysis based, in part, on the estimate.

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		and projects about 62 in 60 years." Provide a technical justification for these projections.	The BC-TOP-1 "Containment Building Liner Plate Design Report" addresses cyclic loading of the main steam penetrations. BC-TOP-1 Loading Condition V is directly dependent on startup-shutdown cycles, which, from experience, are a constant multiplier of two per refueling cycle. WCGS currently refuels on 18-month cycles, and expects about 42 refuelings before the end of the extended period of operation, or about 85 startup-shutdowns cycles at two per refueling. In the 19 years of operation through 2004, WCGS has recorded 27 startup cycles, which also indicates that about 85 might occur in a 60-year operating life. Therefore, the design basis assumption of 100 full-range thermal cycles (BC-TOP-1 Condition V events) should be adequate. The number of assumed BC-TOP-1 Condition IV events does not change with licensed life. The design basis equivalent usage factor for the 10 assumed Condition IV events is 0.270. The design basis equivalent usage factor for the 100 assumed Condition V events is 0.028. Up to 2500 Condition V events would then result in an equivalent usage factor of only $0.270 + 25.0 \times 0.028 = 0.970, <1.0.$
TLAAA019 **(Withdrawn)**		Withdrawn	
TLAAA020 **(Withdrawn)**		Withdrawn	
TLAAA021 **(Withdrawn)**		Withdrawn	
TLAAA022	4.3.2.11	Leak-before-break (LBB) technique was applied for WCGS primary Reactor Coolant Loop piping in	LRA Section 4.3.2.11 presently demonstrates that aging effects affecting the LBB evaluation will be managed for the extended licensed operating period, so that the LBB analysis itself will remain valid for the extended licensed operating period.

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		current licensing period. The original NUREG 0800 Standard Review Plan states that LBB cannot be applied to piping subject to Stress Corrosion Cracking (SCC). The potential for primary water stress corrosion cracking (PWSCC) in Alloy 182-82 weld material has recently been recognized, and this material exists in the hot leg and cold leg welds to the RPV nozzles at Wolf Creek. Please provide technical justification/discussion to demonstrate that LBB analysis remains valid for the period of extended operation.	The question of whether the LBB is valid under the current license is being addressed, but no revision has yet been made to the LBB analysis. Therefore, this question will not be addressed in the license renewal application under 10 CFR 54, but under Part 50.
TLAAA023	4.3.1	In LRA Table 4.3-1, the design limits for line items 12 and 13 are marked as "N/A". In these specific instances, it is not clear what does the term "N/A" means. Clarify if it refers to "not available" or	In this case "N/A" means not applicable, because no specified number of these events was defined as a design limit. This table describes transients counted by the fatigue management program. Where applicable, the "design limits" column lists the number assumed by Westinghouse design specification documents.

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		"not applicable". If it refers to not available, explain why the design limits for these two items are not available. If it refers to not applicable, provide a justification for this conclusion.	The fatigue monitoring program tracks low head safety injection (LHSI Injection) and low-temperature overpressure protection actuation (COMS (LTOP) Actuation), even though they were not defined as independent events by the Westinghouse design specifications used to develop this table. **LRA Table 4.3-1, item 13, COMS (LTOP)** LTOP actuation prevents significant repressurization at low temperature. "RCS Cold Overpressure" has been added to the design specification set of design transients, and "N/A" has been replaced with 10 events of 600 relief valve operating cycles each, 6000 total (Westinghouse Design Specification 952575 Rev. 6, Appendix A) in LRA Amendment 1. **LRA Table 4.3-1, item 12, Low Head Safety Injection (LHSI)** LHSI Actuation is not expected to occur independently of other events, Table 4.3-1 item 20(b), Reactor Trip and Cooldown with Safety Injection, and item 22, Inadvertent RCS Depressurization, both result only in High Head Safety Injection (HHSI) actuation. A discussion during the audit suggested that "LHSI Actuation" might include use of LHSI pumps for residual heat removal (RHR), with a reactor coolant system temperature as high as 350 °F. These RHR operations (with system temperature as high as 350 °F) are included in the heatup, cooldown, and refueling transients. "N/A" is therefore correct for LHSI Actuation. Reference: 1. Westinghouse Design Specification 952575 Rev. 6. "Pressurizer, Addendum to Design Specification 955285 Rev. 0, Standardized Nuclear Unit Power Plant System (SNUPPS)." Westinghouse Proprietary. Pittsburg: Westinghouse Nuclear Energy Systems, 21 December 1992

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			[WCGS DocNo M-713-00004 W06].
TLAAA024	4.3	LRA Section 4.3 contains many terms, words, and statements that are not clearly defined. Clarify the following items: a. In LRA Section 4.3.1.3, the term "corrective action limits" is used without definition. Provide a definition and specific data or rules for each and all of these corrective action limits. b. In LRA page 4.3-11, the applicant states that additional locations will be included if the predicted CUF is approaching to 1.0. Specify which additional locations will be included and the rationale for selecting these particular locations. c. In LRA Section 4.3, items such as "Table 4.3-1 above" and "Appendix B.3.1" are referenced out of context. Clarify these references to	**a. In LRA Section 4.3.1.3, the term "corrective action limits" is used without definition. Provide a definition and specific data or rules for each and all of these corrective action limits.** One of the corrective action limits will be based on the accrued numbers of transient cycles. Fatigue analyses at different locations depend differently on the various types of transient cycles. The cumulative usage factor (CUF) criterion for selection of HELB break locations is a CUF of 0.1 or greater. Related analyses, such as the generic fatigue crack growth calculation done to justify LBB for the RCS main piping loop, assume a set of transient cycles that bound the WCGS specified transient cycles, but are not based on CUF. All of these analyses remain valid so long as the specified numbers of occurrences of the transients are not exceeded. The-cycle based corrective action limit will be set to assure that corrective action is taken to verify continuing validity of all potentially affected calculations before the specified numbers of occurrences of the design transients are exceeded. The description of corrective action limits and corrective actions has been included in section 4.3.1.3 of LRA Amendment 1. **b. In LRA page 4.3-11, the applicant states that additional components will be included if the predicted CUF is approaching to 1.0. Specify which additional components will be included and the rational for selecting these particular components.** The components to be added would depend on those components approaching the fatigue design limit, and therefore include others that might be affected by the same transient events. The added components would be determined as part of the Aging Management Program (AMP) corrective actions, specifically by an extent of condition evaluation.

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		make them more specific. d. The words "approximately," "might," and "more directly" are used in several places such as LRA pages 4.3-11, 4.3-13, 4.3-27, 4.3-27, and 4.3-45. Please examine the uses of these words across LRA Section 4 and clarify the meanings as appropriate. e. In LRA page 4.3-20, the applicant states: "With the basis set of transients, including the power rerate and Thot modification and other effects above, worst-case fatigue factors for the present design exceeded 0.9 in a few pressurizer components." **However, this statement did not provide any specific information on the names and locations of the components at which the "worst-case fatigue factors" for the present design exceeded 0.9. Provide specific**	The description of corrective action limits and corrective actions has been included in section 4.3.1.3 of LRA Amendment 1. **c. In LRA Section 4.3, items such as "Table 4.3-1 above" and "Appendix B.3.1" are referenced out of context. Clarify these references to make them more specific.** These cross-references have been clarified in LRA Amendment 1. **d. The words "approximately," "might," and "more directly" are used in several places such as LRA pages 4.3-11, 4.3-13, 4.3-27, 4.3-27, and 4.3-45. Please examine the uses of these words across LRA Section 4 and clarify the meanings as appropriate.** These statements have been clarified in LRA Amendment 1. Please note that these former page numbers have changed. **e. In LRA page 4.3-20, the applicant states: "With the basis set of transients, including the power rerate and Thot modification and other effects above, worst-case fatigue factors for the present design exceeded 0.9 in a few pressurizer components.** **However, this statement did not provide any specific information on the names and locations of the components at which the "worst-case fatigue factors" for the present design exceeded 0.9. Provide specific information on names and locations of the components at which the worse-case fatigue factors exceeded 0.9.** Calculated design basis usage factors exceed 0.9 at three pressurizer locations. LRA Section 4.3.2.4 has been amended to note the number of locations. The values and their locations are proprietary. The proprietary report is available for review at WCGS and was made available during the audit.

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		information on names and locations of the components at which the worse-case fatigue factors exceeded 0.9. f. In LRA page 4.3-32, the applicant states: "The analysis of data to date indicates no significant effects, and no increase and apparent declines." However, it is not clear how the terms "significant effects" and "apparent declines" are qualified. Provide definitions, and applicable data, for these terms and clarify the conclusions.	**f. In LRA page 4.3-32, the applicant states: "The analysis of data to date indicates no significant effects, and no increase and apparent declines." However, it is not clear how the terms "significant effects" and "apparent declines" are qualified. Provide definitions, and applicable data, for these terms and clarify the conclusions.** The noise event was first monitored to fulfill a commitment to the NRC, and subsequently for tracking and trending purposes. The commitment to the NRC has been met. The analysis of noise event monitoring data prior to Refueling Outage 15 (described in the preceding paragraphs of the LRA) indicated no effects on the vessel, piping, or components sufficient to cause a loss of safety function or to invalidate the design basis of a component, no increase in event severity, and apparent declines in event severity. "Significant effects" means sufficient to cause a loss of safety function or to invalidate the design basis of a component. "Apparent declines" means that although the measured severity (acceleration, velocity, and displacement) of noise events had not uniformly declined with each subsequent heatup, there was an apparent reduction in severity over time when allowances were made for changes in monitoring equipment and evaluation methods. Since the original WCGS LRA was filed, WCNOC has made a preliminary examination of Refuel 15 monitoring data. These results introduced some uncertainty in the statement that previously appeared in this section, that analysis of data to date indicates "apparent declines" in event severity. However, the additional data continue to indicate that the event severity remains bounded by earlier instances. This noise event has been observed since Refueling Outage 5. Indicated severity has not been uniform between occurrences. This variation is

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			expected due to several factors: • The system operating sequence varies prior to each occurrence. • Monitoring equipment and methods have changed due to upgrades. • Data from some events has been partially lost due to monitoring equipment failures. • Equipment has been modified, notably primary loop restraint changes and snubber removal, and reactor vessel head modification. All of these factors have contributed to and will continue to contribute to variability in the measured results; and effects of particular changes are not clearly discernable from the data. Thus, correlation of data from the various occurrences has involved considerable uncertainty. Raw data from Refueling Outage 15 indicate somewhat higher responses than those observed during Refueling Outage 13 and 14, and with these uncertainties, WCGS therefore no longer concludes that there have been "apparent declines" in event severity. LRA Amendment 1, Section 4.3.2.9, reflects this change. However, even with these uncertainties, and the Refuel 15 data, the measured magnitudes and characteristics of these events collected over the period from Refueling Outage 5 through Refueling Outage 15 continue to indicate that effects are very limited, and that the occurrence characteristics remain consistent. WCNOC therefore concludes that results of previous evaluations remain valid, and are expected to continue to remain valid.
TLAAA025 **(Closed to RAIs 4.3-1 and 4.3-2)**	4.3	Ref. WCAP-14173 Global to Local & Transfer functions Rev. 3 - Nov. 1996. 1. During the audit, the staff reviewed basis document "WCAP-14173", which listed	**1a. Please explain why the stress transfer function, (e.g., Table E.2-1) contains only one value and the meaning of this stress.** The FatiguePro Transfer Functions define a single-dimensional peak stress value intended to bound the range of actual stress-intensity cycling for the set of operating transients that contribute significant fatigue usage (i.e. S_{alt} greater than the endurance limit). This is done by modeling individual stress components, and then adding them as integers rather

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		stresses transfer functions. In general, the stress vector consists of 6 stress components ([sigma]xx, [sigma]yy, [sigma]zz, [tau]xy, [tau]yz, [tau]zx). **a. Please explain why the stress transfer function, (e.g., Table E.2-1) contains only one value and the meaning of this stress.** **b. Please justify how one stress component could be used to evaluate fatigue CUF.** **2.** The report defines stress transfer functions as stress intensity. **Please explain how stress intensity value could be used as input for the transfer function methodology.** 3. **Is the same methodology, i.e., using only one component of stress intensity vector to calculate the fatigue value, applied to all RCPB locations?**	than vectors. This is acceptable because $\lvert A+B+C \rvert \leq \lvert A \rvert + \lvert B \rvert + \lvert C \rvert$ for all vectors A, B, and C. Care is taken to sign the components (positive or negative) to maximize the stress range for the transient pairs that produce the most fatigue usage at the given location. Note that unlike stress intensity, this is a signed quantity, which can take on values less than zero. 1b. Please justify how one stress component could be used to evaluate fatigue CUF. As it is defined by Miner's rule and the ASME Code, Fatigue Usage is a function of stress amplitude, not of stress components. FatiguePro uses its one-dimensional (1D) peak stress to develop a stress range spectrum (S_{FP}) that bounds the theoretical spectrum that could be constructed based on perfect knowledge of the time history of the six-dimensional (6D) stress tensor (S_{pure}). Since usage factor is monotonically increasing function of stress range, the usage computed from S_{FP} will bound the usage computed from S_{pure}. Further, in practice most components have a single dominant stress direction with respect to fatigue. For instance, in nozzle safe-end regions, typically fatigue usage is controlled by thermal transients, and the dominant stress component is in the axial direction. This is determined on a case-by-case basis in the Green's Function calculation, which computes the transient thermal stress intensity range using finite element analysis. The uniaxial stress is then taken as the stress intensity response, signed according to the dominant stress component. **2. Please explain how stress intensity value could be used as input for the transfer function methodology.** Stress intensity is not used as input for the transfer functions. The transfer functions take as input: (a) instrument data (and/or calculated system parameters),

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		4. Please describe how the stress transfer functions were benchmarked for the components of Wolf Creek Generating Station. **5. Please explain how to determine the stress transfer function for S(pr), S(momxz), S(momy). (Please use Table E.2-1 of WCAP-14173 asan example to demonstrate S(pr)= 3.71, S(momyz)=9.40, S(momy)=0.0.)**	(b) peak stress intensity ranges from design stress reports, (c) total stress response for a thermal step transient (Green's Function) The term "stress intensity" is not used as a definition, it is used as a description. In fact, the transfer function report defines a virtual stress value that is designed to bound the actual stress intensity ranges for all fatigue-significant transients. This type of stress value does not have a name in the professional literature, so it is spoken of in general terms. **3. Is the same methodology, i.e., using only one component of stress intensity vector to calculate the fatigue value, applied to all RCPB locations?** This is an error of terminology. FatiguePro does not (in general) use just one component of the stress vector to calculate fatigue – it uses the 1D virtual stress described above. FatiguePro does use the same 1D approach for all monitored locations, at WCGS and at all other monitored plants. **4. Please describe how the stress transfer functions were benchmarked for the components of Wolf Creek Generating Station.** FatiguePro Transfer Functions are derived from the Design Stress Report (DSR) for the location in question (see Question 5 below). As such, they are only valid in so far as the DSR they are based on is valid. Structural Integrity Associates (SIA) has never benchmarked Transfer Functions to an independent standard. However, SIA has in the past benchmarked FatiguePro Transfer Functions against the basis DSR. SI does this by simulating the various design transients from the DSR, and then running those transients in FatiguePro. When this has been done, the stress and usage results have matched the DSR results very closely. SIA no longer routinely performs

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			this type of validation, as it is unnecessary. Since the Transfer Functions are derived from the DSR analysis, the results will match as a matter of course. SIA does perform a thorough validation of the FatiguePro software, to assure that it faithfully implements the Transfer Functions. This is executed in a series of verification and validation calculations, according to requirements of a Verification and Validation Plan prepared according to the SIA QA program. The results are summarized in a Software Verification and Validation Report, prepared for each specific FatiguePro version. The report for WCGS is "Software Verification and Validation Report for the FatiguePro Monitoring System for Wolf Creek;" SIR-96-085, Rev. 6, May 1997. **5. Please explain how to determine the stress transfer function for S(pr), S(momxz), S(momy). (Please use Table E.2-1 of WCAP-14173 as an example to demonstrate S(pr)= 3.71 [psi/psi], S(momyz)=9.40 [psi/in-kip], S(momy)=0.0 [psi/in-kip].)** For WCGS, Westinghouse developed the Transfer Functions for the pressurizer locations. Without getting into the proprietary details behind WCAP-14173, the spirit of the question can be answered by describing how those terms would be determined. As mentioned above, the purpose of the various stress components in the Transfer Functions is to bound the stress intensity range of that component during the various operating transients. Those stress intensity ranges are typically derived from the design stress report (DSR) for the location in question, rather than computed according to some formula. In this specific case, Westinghouse used a prior analysis performed to address 88-11 issues – Ref. [5] of the WCAP. (A) The DSR would include consideration of pressure stress in its fatigue

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			evaluation. Let **P** = the maximum pressure from the DSR analysis, in psig, and **S** = the corresponding pressure stress at the critical location, Then: S_{pr} = **S/P** = (xxx)/(yyy) = 3.71 (B) The DSR also provides piping moments for the Surge Line girth weld. Typically, these will be provided at either design or normal operating temperature, with an assumed zero stress at ambient conditions. Let $\mathbf{T_{hot}}$ = the operating temperature, (est. 650°F) $\mathbf{T_{cold}}$ = the stress-free temperature (usually 70°F), and $\mathbf{M_x}$, $\mathbf{M_y}$, $\mathbf{M_z}$ = the moments computed for hot (operating) condition. Then: $S_{momxz} = \text{sqrt}(\mathbf{M_x}^2 + \mathbf{M_z}^2)/(\mathbf{T_{hot}}-\mathbf{T_{cold}})$ = (xxx)/(650-70) = 9.40 $S_{momy} = \text{abs}(\mathbf{M_x}^2)/(\mathbf{T_{hot}}-\mathbf{T_{cold}})$ = (0.)/(650-70) = 0.0 (C) A finite element analysis was performed to compute the stress response of the location to a 1°F step increase in water temperature, either for a conservative flow rate or a range of flow rates. The Green's Function is taken as the extracted stress response (vs. time) at the critical location.
TLAAA026	4.1	There are several inconsistencies with the disposition category described in LRA Table 4.1-1. For example: • LRA Table 4.1-1, states that the disposition	The LRA Table 4.1-1 disposition categories for 'ASME Section III Class I valves' and for 'ASME Section III Class I piping and piping nozzles' are both 10 CFR 54.21(c)(1)(i) and –(iii). The table will be corrected by an amendment to the LRA. The remainder of the disposition categories has been reviewed and is consistent with the text.

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		category for 'ASME Section III Class I valves' is 10 CFR 54.21(c)(1)(iii); however, LRA Section 4.3.2.6 describes two different disposition categories. • LRA Table 4.1-1, states that the disposition category for 'ASME Section III Class I piping and piping nozzles' is 10 CFR 54.21(c)(1)(iii); however, LRA Section 4.3.2.7 describes two different disposition categories. a) Review all the items described in the table and clarify all inconsistencies.	
TLAAA027		LRA Section 4.3.1.1, fatigue design curve, states "The curves include adjustments for the elastic modulus and for departure from zero means stress; and a design margin for uncertainties including modest environmental effects (ASME	The last sentences of the LRA Section 4.3.1.1 description of the basis for the fatigue design curve will be clarified by an amendment to the LRA, to read: The curves include adjustments for the elastic modulus and for departure from zero mean stress; and a margin for uncertainties including modest environmental effects (ASME Section III - 1965, Par. N-415).

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		Section III - 1965, Par. N-415). The design margin is a factor of 2 on stress or a factor of 20 on cycles, whichever produced the lower, more conservative allowable for the data set." In the professional literature, these factors are used to account for differences and uncertainties in fatigue life that are associated with material and loading conditions. Clarify if these factors of 2 and 20 are design margins and how you intent to use them.	The design basis for the Wolf Creek ASME components is to meet all requirements of the Code. This includes use of the Code fatigue design curves.
TLAAA028	4.3	Clarify the following inconsistencies described in LRA Table 4.3-3: a) The LRA table states that the reactor pressure vessel, head, studs, shoes and shims, and supports are evaluated in LRA Section 4.3.2.1. However, this section does not address shoes	LRA Table 4.3-3 will be corrected by an amendment to the LRA, as follows: a) The first entry, first column will read only "Reactor Pressure Vessel, Head, and Studs." The shoes, shims, and supports are not TLAAs. b, c, d) The eighth entry will be changed to a table note applicable to Reactor Coolant Pumps, Pressurizer, and Valves, to read: Pressure-retaining bolting for the reactor coolant pumps, pressurizer, and valves is included in the component code

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		and shims, and supports. b) The LRA table states that pressure retaining bolting in the reactor coolant pumps is evaluated in LRA Section 4.3.2.3. However, this section does not address pressure retaining bolts. c) The LRA table states that pressure retaining bolting in the pressurizer is evaluated in LRA Section 4.3.2.4. However, this section does not address pressure retaining bolts. d) The LRA table states that pressure retaining bolting in valves is evaluated in LRA Section 4.3.2.6. However, this section does not address pressure retaining bolts.	analyses but is not described separately. The seventh entry will read: Steam Generators (Primary or Tube Side and Shell Side),(2) Including Closure Bolting | BB-EBB01A, B, C, D| 4.3.2.5
TLAAA029	4.3.2	LRA Section 4.3.2.5, primary manway studs, states "The replacement studs met code stress criteria, but high calculated usage factors would have required their	The current code design report [Ref. 1, incorporated in Ref. 2] is certified to the revised plant-specific specification with rerate and the T_{hot} reduction [Ref. 3], and cites [at Ref. 1 § 9.2 and Tables 9.2-1, 9.2-4, and 9.2-5] a qualification by test, for the design basis set of lifetime transients, from Westinghouse test reports [Refs. 4 and 5, cited as Refs. 24 and 26 by Ref. 1]. The results of these tests were evaluated to the requirements of

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		periodic replacement, at the rate of transient cycle accumulation implied by the original 40-year design life. The studs and nuts were qualified by test, with a sufficient number of test cycles to envelope the entire set of design basis transients." Explain in detail how these studs and nuts were qualified by tests.	ASME III Appendix II requirements [Ref. 6, cited as Ref. 27 by Ref. 1]. A detailed description of the actual test reports basis and results, from Reference 6, is included in the current design report [Ref. 1 § 9.2], including • Comparison of the Wolf Creek design basis duty cycles with those of the Model F standard specification (Table 9.2-2) • Comparison of the Wolf Creek design basis duty cycles with those of the Model F transient grouping, used to develop the test transient groupings (Table 9.2-3) • Development of test transient groupings and their strain (test deflection) ranges (Tables 9.2-4, -5, and –6) • Demonstration that the stress ranges achieved by the test bound those indicated by the Wolf Creek design report (Tables 9.2-10 through –15). **References:** 1. WCAP-16546-P, WNET-180(SAP) Volume 1, Revision 2. Westinghouse Design Report. P. A. Stancampiano. Model F Steam Generator Stress Report for Wolf Creek Nuclear Power Plant, Revision 2 of WNET-180(SAP) Volume 1, SCGT 2271, SCGT 2272, SCGT 2273, SCGT 2274. Westinghouse Proprietary Class 2. Madison, PA: Westinghouse Electric Company LLC Nuclear Services, March 2007. Incorporated into Ref. 2. 2. WCNOC Calculation BB-S-017. Arthur P. L. Turner. "Model F Steam Generator Stress Report for Wolf Creek Nuclear Power Plant." Contains Westinghouse Proprietary Class 2 information. Rev. 0. 16 March 2007. Incorporates Ref.1.

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			3. Westinghouse Specification 953291 Rev. 8. "Standardized Nuclear Power Plant System Model F Steam Generator." Westinghouse Proprietary Class 2. Westinghouse Electric Company LLC, 2 March 2007. Includes rerating and T_{hot} reduction. 4. Westinghouse Calculation Note SM-90-19. "Primary Manway Stud Extended Fatigue Evaluation for Wolf Creek." Rev.0. March 1990. 5. Westinghouse Report WNEP-8646. Fatigue Life Qualification Test of Steam Generator Primary Manway Closure Studs and Gear-Nuts, Test Report for Georgia Power Company Vogtle Plant A. August 1986. 6. Westinghouse Calculation Note CN-SGDA-01-46 "Wolf Creek MODs Stress Report Update Summary." Rev. 0. June 2001.
TLAAA030	4.3.2	LRA Section 4.3.2.5, studs barrels, states that if the number of load cycles assumed by the fatigue analysis is not exceeded; the predicted usage factor will remain within the allowable of 1.0 a) Clarify what is the number of load cycles assumed by the fatigue analysis b) Clarify what is the	LRA Section 4.3.2.5 will be amended to read: "The code stress report includes.... If the number of load cycles specified by the design specifications and evaluated by the fatigue analysis is not exceeded; the calculated usage factor will remain within the allowable of 1.0."

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		predicted usage factor	
TLAAA031	4.3	ICE-ICAT(97)-012, addressed in the basis document, states that the accumulated cycle from 1984 through March 1992 is 7 for the loss of offsite power and 2 for loss of load transients. LRA Table 4.3-1 indicates that no additional cycles occurred between March 1992 and December 2005. In order to demonstrate the validity of automatic cycle counting, verify that these transients did not occur.	The loss of offsite power transient is initiated by complete loss of offsite power (System Standard 1.3.F). It has been confirmed from records other than the fatigue monitoring program that no complete loss of offsite power events occurred during the period March 1992 to December 2005. Some, if not all, of the loss of offsite power events recorded in ICE-ICAT (97)-012, based on a review of historical plant records, were probably partial losses of offsite power (i.e., loss of offsite power to one train, but not the other). Counting these events as complete loss of off-site power is conservative. System Standard 1.3F defines "Loss of Load" as "This transient involves a step decrease in turbine load from full power (turbine trip) without immediate reactor trip" and "The reactor eventually trips as a consequence of a high pressurizer level trip." It was confirmed from records other than the fatigue monitoring program that no events meeting this description occurred during the period March 1992 to December 2005. One or both of the loss of load events reported in ICE-ICAT (97)-012 may be the result of conservative counts of loss of load events during which the expected immediate reactor trip did occur. It is concluded that the automated cycle counting module of the fatigue monitoring system correctly shows no accrued cycles for these transients during the period March 1992 to December 2005, because none have occurred. There is also evidence that the construction of the cycle count baseline by review of historical records reported in ICE-ICAT(97)-012 is conservative.