WOLF CREEK
NUCLEAR OPERATING CORPORATION

June 7, 2007

Matthew W. Sunseri
Vice President Oversight

WM 07-0051

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 WM 07-0049, dated June 1, 2007, from M. W. Sunseri, WCNOC, to USNRC

Subject: Docket No. 50-482: 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 two audits at WCGS. The LRA Aging Management Program audit was conducted during the week of March 26, 2007 and the LRA Aging Management Review during the week of May 7, 2007. During the course of these audits the NRC staff also audited Time Limited Aging Analyses (TLAA).

Based on the results of the TLAA audit, it was determined that an Amendment to Section 4.1 and 4.3 of Reference 1 would facilitate the NRC Staff review. Reference 2 provided that Amendment.

Enclosure 1 provides the question and answer database that was complied during the TLAA audits. Each entry consists of a numbered question, reference to the applicable section of the LRA and the WCNOC response.

Attachment I provides a summary of the commitment made in this response.



A121
NRO

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

Sincerely,

M W Sunseri

Matthew W. Sunseri

TJG/rlt

Attachment I - List of Commitments
Enclosure 1 - TLAA Question and Answer Database

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 )**

Matthew W. Sunseri, of lawful age, being first duly sworn upon oath says that he is Vice President Oversight 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** M W Sunseri
Matthew W. Sunseri
Vice President Oversight

SUBSCRIBED and sworn to before me this 7 day of June, 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

## LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation in this document. Any other statements in this letter are provided for information purposes and are not considered regulatory commitments. Please direct questions regarding these commitments to Mr. Kevin Moles, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4126.

| | COMMITMENT SUBJECT | LRA Section | COMMITMENT DESCRIPTION |
|---|---|---|---|
| 34 | LRA Amendment | N/A | License Renewal Application changes discussed in WM 07-0051 will be submitted in an amendment to the application.<br><br>**Reference: WM 07-0051**<br>**Due: August 31, 2007** |

# Time Limited Aging Analyses (TLAA) Question and Answer Database

## Wolf Creek TLAA Audit Questions and Responses

| Question No | LRA Sec | Audit Question | Final Response |
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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."<br><br>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.<br><br>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 LRA Section 4.3, to read:<br><br>"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 | 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 actual plant experience remains bounded by design assumptions and calculations reflected in the USAR.<br>**1. Describe how the fatigue aging management program tracks usage factor.**<br><br>**TLAAA002 (Follow-up)**<br>In its response, the applicant indicated that design basis transient data were used for the fatigue usage factor tracking and | **1. Describe how the fatigue aging management program tracks usage factor.**<br>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:<br><br>For the period of extended operation, the 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."<br><br>Cycle-based monitoring assumes the alternating stress range 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 methods similar to those of the ASME code to define transient stress ranges between events. |

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| | | that this is conservative because it assumes that each actual transient is as severe as a design basis transient. In addition, the response states that the calculation adds a normalization factor to account for transients not tracked by the program.<br><br>During the audit, the staff reviewed basis document FP-WOLF-304 which 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 CUF prior to the available data.<br><br>The applicant's response is not consistent with the plant basis document.<br><br>**(1)a. Clarify the inconsistency and b. provide further discussion of the transient data.**<br><br>**(2) Discuss the transient severity during the period from 1983 through 1996 to ensure that backward-projected initial CUFs are reasonable.**<br><br>**(3) Provide the referenced calculation for the normalization factor.** | For the period of extended operation the WCGS fatigue monitoring program will use stress-based monitoring 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 two 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).<br><br>Stress-based monitoring uses actual plant transient profile data to determine 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 stress range is determined from pairs of events as they actually occur. Fatigue usage accumulation is then calculated from this stress range, for each cycle.<br><br>However, the WCGS fatigue monitoring program will use only cycle-based results for the locations monitored by the stress-based method, unless stress-based results are required to demonstrate acceptable fatigue usage.<br><br>**TLAAA002 (Follow-up) Response**<br>**(1)a. Clarify the inconsistency**<br>The LRA has been amended to cite fatigue usage factor predictions only to support disposition, consistent with NUREG/CR-6260 methods, of the Section 4.3.4 "Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)." The LRA has been amended to clarify the distinction between the fatigue predictions, which use both actual plant data and design basis transient data to demonstrate that fatigue usage should not exceed the code allowable of 1.0 during the extended licensed operating period; and the WCGS fatigue management program, which ensures that fatigue usage will not exceed the code allowable of 1.0 during the extended licensed operating period. For the six NUREG/CR-6260 locations with significant projected usage factors the amended LRA Section 4.3.4 disposition is the WCGS fatigue management program, in accordance with 10CFR54.21(c)(1)(iii).<br><br>LRA Table 4.3-2 shows the 28 locations monitored by the program. Fatigue in the line 1 through 4 locations in the table is tracked with cycle-based fatigue (CBF) monitoring. Fatigue in the line 5 through 13 locations in the table is tracked with stress-based fatigue (SBF) monitoring. The methods used by the program are described in FP-WOLF-304, and use (1) actual plant cycle count data, (2) the design basis fatigue effect of each cycle transient for CBF calculations of CUF, and (3) actual plant transient profile data to calculate fatigue usage factors for locations tracked by SBF monitoring.<br><br>**(1)b. provide further discussion of the transient data** |

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| | | | For the cycle-based locations (line numbers 1-4 in LRA Table 4.3-2), the WCGS fatigue management program uses (1) actual plant cycle count data and (2) the design basis fatigue effect of each cycle transient pair. Since all of the cycles contributing to fatigue have been accounted for since the beginning of plant life, the CUF results already represent the baseline usage to date for these locations and no back-projection was necessary [FP-WOLF-304 §8.2.1].<br><br>For the remaining 24 locations (line numbers 5-13 in LRA Table 4.3-2) the WCGS fatigue management program uses the stress-based fatigue model. This model uses both (1) actual plant data for the number of cycles to date and (2) actual plant transient profile data, to calculate usage factor accumulation over the period for which the profile data are available, i.e. from January 13, 1996 through December 31, 2005. The results for this nearly-10-year period were then also used to back-calculate the usage factor accumulation for the earlier periods [FP-WOLF-304, §8], assuming that the severities of the transients during the monitored period are typical of the period before monitoring was implemented.<br><br>**(2) Discuss the transient severity during the period from 1983 through 2004 to ensure that backward-projected initial CUFs are reasonable.**<br>The back-calculation is described in SIA Calculation Package FP-WOLF-304,"Baseline Evaluation and 60 Year Projection for Wolf Creek" (the "basis document"). This document contains proprietary data but is available at WCGS and was supplied during the audit. See also the response to TLAAA005.<br><br>**(3) Provide the referenced calculation for the normalization factor.**<br>The normalization factor is described in SIA Report SIR-95-043, "Cycle Counting and Cycle-Based Fatigue Methodology Report, Transient and Fatigue Monitoring System for Callaway/Wolf Creek." This document was supplied during the audit.<br><br>Discussions during the May 2007 audit meeting established that further written information on the normalization factor was not required. No information on the normalization factor or on the use of the related use of preloaded cycles has been included in this response or the LRA.<br><br>References:<br>1. 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.<br><br>2. 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. 3. December 2004. |

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| 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.<br>**1. Clarify what is the time frame for the second period.**<br><br>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 transient data acquisition system in March 1992.<br>**2. Explain how the effects were estimated taking in consideration the operating history.**<br>**3. Provide transient history cycle counting data prior to the installation of the automated system.**<br>**4. Clarify whether the program uses cycle counting only or if it performs online stress evaluation and CUF calculations.**<br><br>**TLAAA003 (Follow-up)**<br>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. | **1. Clarify what is the time frame for the second period.**<br>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.<br><br>**2. Explain how the effects were estimated taking in consideration the operating history.**<br>The estimates of event cycles examine the detailed operating history, and apply scaling rules where the history indicates a change in the rate of accumulation, if exact historical counts are not available. The notes to Table 4.3-1 describe details of particular cases.<br><br>**3. Provide transient history cycle counting data prior to the installation of the automated system.**<br>The details of the extraction of the current cycle counts from the operating history, and the detailed projections for the extended operating period, are available in supporting SIA, Inc. proprietary document FP-WOLF-304 (Ref. 2), and in the Westinghouse ICE-ICAT(97)-012 proprietary report of pre-1992 estimates (Ref. 1), both available at WCGS.<br><br>**4. Clarify whether the program uses cycle counting only or if it performs online stress evaluation and CUF calculations.**<br>For record purposes the WCGS fatigue management program uses cycle counting only.<br><br>**TLAAA003 (Follow-up) Response**<br>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 not a description of the program as implemented.<br><br>For record purposes the present WCGS fatigue management program uses cycle counting only.<br><br>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. For the period of extended operation the WCGS fatigue management program will use stress-based monitoring for the remainder of the locations in LRA Table 4.3-2, including the two 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 charging nozzles). |

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| | | | However, the WCGS fatigue management program will use only cycle-based results for the locations monitored by the stress-based method, unless stress-based results are required to demonstrate acceptable fatigue usage.<br><br>See also the response to TLAAA005.<br><br>References:<br>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.<br><br>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.<br>**1. Provide a summary of these transients and any revisions made to the design specifications.**<br><br>**TLAAA004 (Follow-up)**<br>In its response, the 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.<br>**2. Discuss the fatigue impact on the pressurizer nozzles due to the application of weld** | **1. Provide a summary of these transients and any revisions made to the design specifications.**<br>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 design specification and 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. See also the response to TLAAA011, on Section 4.3.2.4.<br><br>Surge line stratification effects for Wolf Creek Generating Station (WCGS) and Callaway were initially evaluated in WCAP-12893 based on data acquired before WCGS adopted modified operating procedures (MOP) to reduce the stratification cycling when there is a large temperature difference between the pressurizer and the RCS. MOP establish a continuous outflow from the pressurizer during plant heatup and cooldown, which creates a stable stratified condition in the surge line minimizing stratification cycling and stress cycles. Therefore, the frequency of stratification cycling assumed for the WCAP-12893 analysis is conservative for current WCGS operations.<br><br>The results of the fatigue usage analyses documented in WCAP-12893 have been incorporated into the design stress report summary for the primary auxiliary piping (WCAP-9728, Vol. IV, Rev. 2). Surge line stratification does not significantly increase the fatigue usage results for the pressurizer surge line nozzle (in the pressurizer stress report, rather than the surge line stress report) because the location in the nozzle affected by the pipe loads is not the most limiting for fatigue usage. Maximum fatigue usage for the pressurizer surge line nozzle occurs in the thick part of the nozzle at the |

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| | | **overlays.** | nozzle to vessel transition. The stresses causing fatigue usage at that location are from temperature gradients, not from pipe loads.<br><br>The WCGS site specific piping 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 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.)<br><br>Most of the design specifications and code analyses of these components are proprietary. Current specifications and analyses of record are available for review at WCGS.<br><br>**TLAAA004 (Follow-up) Response**<br>**2. Discuss the fatigue impact on the pressurizer nozzles due to the application of weld overlays.**<br>The surge line weld overlay installed during Refueling Outage 15 covers only the nozzle-to-safe-end weld, the safe end, and the safe-end-to-pipe weld. The maximum fatigue usage in the surge nozzle is at a location remote from this overlay and is unaffected by the increased thickness of the weld overlay. The preliminary analysis of this overlay demonstrates that the peak stresses 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 affected by the overlays is no greater than was calculated for the original welds.<br><br>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.<br><br>The information on this weld overlay modification and analysis was incomplete at the time the LRA was originally filed. This information has been included in LRA Amendment 1 Section 4.3.2.4 (page 4.3-20).<br><br>References:<br>**Pressurizer Surge Nozzle – Section 4.3.2.4**<br>1. Westinghouse Specification 955285 Rev. 0. "Pressurizer Series 84F." Westinghouse Proprietary. Pittsburgh: Westinghouse Nuclear Energy Systems, 1 May 1981 [WCGS DocNo M-713-00020 W01].<br><br>2. Westinghouse Specification 952575 Rev. 6. "Pressurizer, Addendum to Design |

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| | | | Specification 955285 Rev. 0, Standardized Nuclear Unit Power Plant System (SNUPPS)." Westinghouse Proprietary. Pittsburgh: Westinghouse Nuclear Energy Systems, 21 December 1992 [WCGS DocNo M-713-00004 W06]. Includes changes for rerating, steam generator tube plugging, and Thot reduction.<br><br>3. Westinghouse Design Report WNET-138 (SAP) Vol. 1 Rev. 3. J. K. Visaria. "Model F Series 84 Pressurizer Stress Report for Standardized Nuclear Unit Power Plant Systems (SNUPPS), Wolf Creek Nuclear Power Station Unit No. 1." Westinghouse Proprietary Class 2. Pensacola, FL: Westinghouse Electric Nuclear Services Division, January 1993 [WCGS DocNo BB-S-020 Rev. 0].<br><br>4. Westinghouse Design Report WNET-138 (SAP) Vol. 1 Add. 1. "Pressurizer Site Stress Report Addendum for Standardized Nuclear Unit Power Plant Systems (SNUPPS), Wolf Creek Unit 1." 4 January 1983 [With Ref. 3 in WCGS DocNo BB-S-020 Rev. 0].<br><br>5. Westinghouse Design Report WNET-138 (SAP) Vol. 1 Add. 2. "Pressurizer Site Stress Report Addendum for Standardized Nuclear Unit Power Plant Systems, Wolf Creek Unit 1." Westinghouse Proprietary Class 2. Cheswick, PA: Westinghouse Electro-Mechanical Division, January 1989 [With Ref. 3 in WCGS DocNo BB-S-020 Rev. 0]. Includes effect of constrained thermal expansion cycles for one relief nozzle due to a pinned support on the relief line.<br><br>6. Westinghouse Design Report WNET-138 (SAP) Vol. 1 Add. 3. "Pressurizer Site Stress Report Addendum for Standardized Nuclear Unit Power Plant Systems." Westinghouse Proprietary Class 2. Cheswick, PA: Westinghouse Electro-Mechanical Division, 25 January 1993 [With Ref. 3 in WCGS DocNo BB-S-020 Rev. 0]. Results of Ref. 2 changes for rerating, steam generator tube plugging, and Thot reduction.<br><br>7. WCNOC Calculation BB-S-009. , "Pressurizer Thermal Gradient Stress Analysis," For PMR CCP 05477, 22 October 1994. Accepts and incorporates Dominion Engineering Letter Report L-4321-01-1, David J. Gross, Dominion; to Arthur P. L. Turner, WCNOC; "Pressurizer Thermal Gradient Stress Analysis," 11 October 1994. Not a code design report.<br><br>8. Westinghouse Nuclear Safety Advisory Letter NSAL-04-5. "Pressurizer Insurge-Outsurge Transients." Pittsburgh: Westinghouse Electric, 26 August 2004.<br><br>9. WCAP-14950. M. A. Gray et al. Westinghouse Report. Mitigation and Evaluation of Pressurizer Insurge-Outsurge Transients. Westinghouse Proprietary Class 2C. February 1998. |

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| | | | **Surge Line and Hot Leg Surge Nozzle – In Sections 4.3.2.7 and 4.3.2.8:**<br>10. Westinghouse Design Specification G 667458 Rev. 4. "General Piping Design Specification, ANS Safety Class 1." Westinghouse Proprietary. Pittsburgh: Westinghouse Electric Corporation Nuclear Energy Systems, 20 June 1978 [WCGS DocNo M-730-00001 W03]. For Interim Rev. 1 to this specification see Ref. 11.<br><br>11. Westinghouse Design Specification 955238 Rev. 1. Contains G-667458 Rev. 4 [Ref. 10] plus Interim Rev. 1 to Rev. 4. "Standardized Nuclear Unit Power Plant System (SNUPPS), Piping Design Specification, ANS Safety Class 1." Westinghouse Proprietary. Pittsburgh: Westinghouse Electric Corporation Nuclear Energy Systems, 27 August 1981 [WCGS DocNo M-730-00003 W06]. Amended by Reference 12.<br><br>12. 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). The complete copy requires both this and Reference 11.<br><br>13. WCAP-9728 Volume 1 Rev. 3. C. K. Ng et al. "Westinghouse Design Report. Structural Analysis of the Reactor Coolant Loop for Standard Nuclear Unit Power Plant System." Westinghouse Proprietary. Pittsburgh: Westinghouse, August 1994.<br><br>14. WCAP-9728 Volume 3 Rev. 4. R. L. Brice-Nash et al. "Westinghouse Design Report. ASME Section III Class 1 Reactor Coolant Loop Branch Nozzle Stress Analysis for the Standard Nuclear Unit Power Plant System, Volume III." Westinghouse Proprietary Class 2C. Pittsburgh: Westinghouse, December 1995.<br><br>15. WCAP-9728 Volume 4 Rev. 2. R. L. Brice-Nash et al. "Westinghouse Design Report. ASME Section III Class 1 Auxiliary Piping Stress Analysis for the Standard Nuclear Unit Power Plant Systems, Volume IV." Westinghouse Proprietary Class 2C. Pittsburgh: Westinghouse, December 1995.<br><br>16. WCAP-12639. B. J. Coslow et al. "Westinghouse Owners Group Pressurizer Surge Line Thermal Stratification Generic Detailed Analysis Program MUHP-1091." Westinghouse Proprietary Class 2. Pittsburgh: Westinghouse, June 1990.<br><br>17. 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.<br><br>18. WCAP-12893 Supplement 1. L. M. Valasek. "Structural Evaluation of the Wolf |

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| | | | Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal *Stratification, Supplement* 1." *Westinghouse Proprietary Class 2C.* Pittsburgh: Westinghouse, December 1995. Includes effects of snubber reduction. Superseded by Ref. 15. Not a code design report. |
| TLAAA005 | 4.3.1 | LRA Table 4.3-1, footnote 3, states that "These estimates are also the bases for the evaluation of effects of the reactor coolant on fatigue life in Section 0."<br><br>**1. Provide detailed information showing the estimates for each transient.**<br>For example:<br>**2.· demonstrate and justify how the recorded 55 cycles reactor trips in 20 year operation was used to project 99 cycles for 60 years operation**<br>**3.· demonstrate and justify why the cycle numbers for reactor coolant leak test is 3 for the next 40 years (2005-2045)**<br><br>**4. Provide a technical justification for eliminating those design transients, which was then projected to 0 cycle in 60 year EOL, in the system design and fatigue analyses.**<br><br>**TLAAA005 (Follow-up)**<br>The staff does not agree that using near term operating experience provides a conservative basis because by using the near term operating experience only (or by providing more weight to the near term operating experience) the design basis transient cycles will be significantly reduced in the 60 | **Introduction**<br>This question and other audit questions have prompted filing of LRA Amendment 1 that addresses these original and followup questions. For this reason, the response to the original and followup questions cannot easily be separated, and have therefore been combined in this revised Response to TLAAA005.<br><br>The heads or text of the "Disposition" paragraphs of LRA subsections 4.3.2.1, 4.3.2.3, 4.3.2.6, 4.3.2.7, 4.3.2.10 and 4.3.2.11 stated or implied that projection of cycles or usage factors validate the code analyses, in accordance with 10 CFR 54.21(c)(1)(i). These dispositions in fact depend on the WCGS fatigue management program, in accordance with 10 CFR 54.21(c)(1)(iii), not on validation. These paragraphs in the LRA Amendment 1 now state only that the WCGS fatigue management program will ensure that the code analyses will remain valid for the extended licensed operating period, in accordance with 10 CFR 54.21(c)(1)(iii). For further information on these changes please see the response to the TLAAA002 followup request.<br><br>The projections are not the basis for the validation of these TLAAs. The basis for disposition of these TLAAs for the extended licensed operating period is the WCGS fatigue management program, which ensures that fatigue usage does not exceed the code limit of 1.0 without corrective action, regardless of the actual rate of fatigue usage accumulation at any particular time.<br><br>**Responses**<br>**LRA Table 4.3-1, footnote 3, states that "These estimates are also the bases for the evaluation of effects of the reactor coolant on fatigue life in Section 0."**<br>The column containing projected end-of-life cycles, and this footnote, has been omitted from the amended LRA Table 4.3-1. The similar columns of projected usage factor in Table 4.3-2 have also been omitted. The following responses to these audit questions on cycle projections have, however, been retained.<br><br>**1. Provide detailed information showing the estimates for each transient.**<br>Space does not permit including this detailed information in this response, other than in the general description of the method, below. The detailed projections and their basis are contained in SIA, Inc. Calculation Package FP-WOLF-304 (Ref. 1). This report contains proprietary data but is available at WCGS and was provided during the audit.<br><br>The algorithm used by FP-WOLF-304 to predict future rates of accumulation for various transient cycles weights recent history more heavily than experience from startup and |

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| | | year projection. Recent operating experience at older nuclear power plants demonstrates that, as components age or degrade, more thermal transient cycles could occur.<br><br>**5. Please provide justification for reducing (or for not considering) the impact of components degradation in the cycle projection for the 60 years.**<br><br>**6. In addition, WCGS stated that the design transient would not occur or that cycles can be reduce significantly, provide a justification to ensure that those transients can be eliminated or cycles can be significantly reduced.** | early years of operation for predicting the future rate of accumulation. However, in most cases the effect on the model of a more-rapid accumulation of cycles early in the operating life increases the projected future rate of accumulation.<br><br>For reactor trips and most other transients, this future linear accumulation rate E' is greater than the recent experience, and is computed as follows (from FP WOLF 304 Section 7.4.1):<br><br>[NOTE: The data base will not accept subscripts. Subscripts are therefore preceded by a down arrow (↓).]<br><br>$E = (R_1 * LTW + R_2 * STW) / (LTW + STW)$,<br><br>where:<br><br>$R_1$ = total average rate of cycle accumulation to date = $(N_{now} - N_{init}) / (t_{now} - t_{init})$,<br>$R_2$ = short-term average rate of cycle accumulation = $(N_{now} - N_{ago}) / (t_{now} - t_{ago})$,<br>$t_{init}$ = date/time associated with the start of cycle counting (Y0),<br>$t_{now}$ = date/time of latest day that has been processed,<br>$t_{ago}$ = a time that was NY years prior to $t_{now}$,<br>$N_{init}$ = configured initial cycle count for this event (usually 0),<br>$N_{now}$ = computed current cycle count for this event,<br>$N_{ago}$ = computed cycle count at time $t_{ago}$,<br>LTW (long-term weighting factor), STW (short-term weighting factor),<br>Y0 (first year of plant operation) and NY (number of years for short term) are configurable values that can be adjusted by the user).<br><br>The weighting factors were adjusted to give more importance to short-term rates than long-term rates. This reflects the assumption that the future rate of accumulation is better represented by recent history than the entire history. The following parameters were used:<br><br>Y0 = 1983 (i.e., first year of plant operation)<br>LTW = 1.0<br>STW = 3.0 (placing more importance on short term weight than long term)<br>NY = 9 (just less than the number of years of available monitoring program data: 1996 – 2005)<br><br>This method was used for all of the 60-year projections, with changes to some of these parameters in a few special cases, and with exceptions for zero accumulation in a few others. |

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| | | | **2. For example: demonstrate and justify how the recorded 55 cycles reactor trips in 20 year operation was used to project 99 cycles for 60 years operation.**<br>The 99 cycles of reactor trip (without cooldown) were projected using the model described above. The model is based on the plant operating history; using the cycles accumulated more rapidly during initial operation, plus those accumulated more recently, at a lower rate. The model assumes a conservative linear accumulation rate, somewhat greater than the most recent, for the remainder of plant life (Ref. 1, Figure 7-1, p. 108).<br><br>The normal reactor trip events were also calculated by this method and were not reduced to zero. However some reactor trip transients ([Trip with Inadvertent] "Cooldown with Safety Injection" and "...without Safety Injection") are special cases. No reactor trips have occurred with inadvertent cooldown, and none are expected to occur. Therefore, zero cycles of these subclasses of reactor trips were projected. The "NS" for those "...with Turbine Overspeed" reflects their inclusion in the number of transients with "No Inadvertent Cooldown."<br><br>**3. For example: demonstrate and justify why the cycle numbers for reactor coolant leak test is 3 for the next 40 years (2005-2045).**<br><br>Reactor coolant leak tests are a special case. The basis for the limited number of expected reactor coolant leak tests is the same as the basis for reduction in the number analyzed from 200 to 50: the absence of any planned leak tests at the pressure assumed by the definition of this transient. Therefore three additional tests is a conservative estimate.<br><br>**4. Provide a technical justification for eliminating those design transients, which was then projected to 0 cycle in 60 year EOL, in the system design and fatigue analyses.**<br>Transients that have an end-of-life predicted number of occurrences equal to 0 have not been eliminated or removed from cycle count tracking, nor from the basis for the fatigue analyses, but are unusual events that are not expected to actually occur. If such an event does occur, it will be counted and the fatigue usage attributable to the transient will be added to the cumulative usage. The justifications for projecting zero or significantly reduced cycles were described in the notes to Table 4.3-1 (now being omitted, with the omission of the projections), and in FP-WOLF-304 (Ref. 1); and are based on the 20 years of operating history plus operating guidance and limitations, as described.<br><br>**5. Provide justification for reducing (or for not considering) the impact of components degradation in the cycle projection for the 60 years.**<br>In most cases, the net effect of the FP-WOLF-304 model is a greater number of |

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| | | | projected cycles than would be predicted from the recent accumulation rate, less than would be predicted by the early accumulation rate, and slightly less than would be predicted by the average over the plant life to date. The inclusion of the early accumulation rate should be adequate to account for any increase that might occur late in the licensed operating period. See FP-WOLF-304 Figure 7.1. There are a few exceptions, for example refueling cycles, which accumulate nearly exactly linearly for the entire projected life.<br><br>**6. In addition, WCGS stated that the design transient would not occur or that cycles can be reduce significantly, provide a justification to ensure that those transients can be eliminated or cycles can be significantly reduced.**<br>Any statements concerning reduction in the number of transients in the context of these projections did not refer to reduction in the number of "design transients," that is, as listed in code design specifications for purposes of calculating expected lifetime usage factor; but referred only to projection of cycles for estimating probable usage factor, based on actual operating experience. As stated above, these projections are not the basis for disposition of these fatigue TLAAs.<br><br>Any reduction in the number of design transients for purposes of demonstrating acceptable design usage factors as calculated by the code analyses requires a revision to the governing code design specification.<br><br>Reference<br>1. 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. |
| TLAAA006 | 4.3.1 | LRA Table 4.3-1, footnote 4, states that the recorded 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.<br>**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.** | **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.**<br>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.<br><br>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 |

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| | | ***TLAAA006 (Follow-up)***<br>In its response, the applicant states that a temperature change of less than 150°F at any rate produces no fatigue usage.<br><br>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 [EαΔT/2(1-μ)=(30E6)(7E-6)(150)/2(1 0.3)]. Thus, a ±150°F step 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. | 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.<br><br>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 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.<br><br>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.<br><br>**TLAAA006 (Follow-up) Response**<br>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.<br><br>Reference:<br>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, respectively. However, LRA Table 4.3-5 lists the estimated CUF for the surge | 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.<br><br>The apparent difference arises because of the generic nomenclature used by NUREG 6260 to describe the locations evaluated in Table 4.3-5, 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 |

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| | | line hot leg nozzle as 0.05849. Clarify the difference. | appear in Table 4.3-5.<br><br>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.<br><br>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.<br><br>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."<br><br>Explain in detail the meaning of this statement. Clarify if WCGS is certain that the CUF will not exceed 1.0 or if this is an assumption. | 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.<br><br>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.<br><br>The cycle count projections have been eliminated from Table 4.3-1, and the text has been amended accordingly. |
| 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."<br>Justify if there will be sufficient time to take appropriate and timely corrective actions if periodic evaluation prescribed by | 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.<br><br>(The following text is copied from LRA Amendment 1, Section 4.3.1.3, Enhanced Corrective Action Limits and Corrective Actions.)<br><br>**Enhanced Corrective Action Limits and Corrective Actions**<br>The WCGS fatigue management program provides for periodic evaluation (once per operating 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 |

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| | | 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. | limits will not be exceeded.<br><br>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.<br><br>***Cycle Count Action Limit and Corrective Actions***<br><br>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.<br><br>If this action limit is reached, acceptable corrective actions include:<br><br>1. Review of fatigue usage calculations.<br>• To determine whether the transient in question contributes significantly to CUF.<br>• To identify the components and analyses affected by the transient in question.<br>• 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.<br><br>2. Evaluation of remaining margins on CUF based on cycle-based or stress-based CUF calculations using the WCGS fatigue management program software.<br><br>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).<br><br>***Cumulative Fatigue Usage Action Limit and Corrective Actions***<br><br>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 operating cycles.<br><br>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. |

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| | | | If this action limit is reached acceptable corrective actions include:<br><br>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.<br><br>2. Enhance fatigue monitoring to confirm continued conformance to the code limit.<br><br>3. Repair the component.<br><br>4. Replace the component.<br><br>5. Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded.<br><br>6. Modify plant operating practices to reduce the rate of fatigue usage accumulation rate.<br><br>7. Perform a flaw tolerance evaluation and impose component-specific inspections.<br><br>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. |
| 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:<br><br>(a) For the first set, the applicant states that "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 | 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.<br><br>**(a) Explain why the primary loop LBB analysis is related to the worst-location usage factor.**<br>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.<br><br>The LBB evaluation applies only to the primary coolant loop piping. The LBB evaluation |

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| | | basis no longer valid..."<br><br>**Explain why the primary loop LBB analysis is related to the worst-location usage factor.**<br><br>(b) For the second set, the 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."<br><br>**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.** | 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 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.<br><br>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).<br><br>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]<br><br>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).<br><br>**(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.**<br>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 their expected mean usage factors per cycle (or 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.<br><br>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 |

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| | | | 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.)<br><br>Reference<br>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.<br><br>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).<br><br>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.<br><br>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.<br><br>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. |

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| TLAAA012 | 4.3.2 | LRA Section 4.3.2.5 describes that the applicant uses both 10 and 15 percent steam generator tube plugging in its steam generator fatigue analyses. Explain the difference. | For steam generators, the 10 percent tube plugging assumed by the power rerate evaluations has been superseded by the current analysis of record, which includes up to 15 percent plugging.<br><br>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].<br><br>Reference:<br>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" charging nozzles have equal calculated usage factors. However, LRA Table 4.3-5 lists two different expected CUF values. Clarify the inconsistency. | For some years the two charging paths were used unequally, resulting in a faster accumulation of usage factor in the normal nozzle. These 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.<br><br>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 |

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| | | | 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.<br><br>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.<br><br>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 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." | 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 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.<br><br>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" bear no meaningful relation to the same terms as usually understood in mechanical design.<br><br>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.<br><br>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 that "Table 4.3-1, Item 1 shows only 27 startup cycles in the 19 years through 2004, and | LRA Amendment 1 removed the 60-year projection column from Table 4.3-1.<br><br>Section 4.6.2 will be amended to provide the following estimate, and analysis based, in part, on the estimate. |

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| | | projects about 62 in 60 years." Provide a technical justification for these projections. | 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 (Condition V events) should be adequate.<br><br>The number of assumed 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<br><br>$0.270 + 25.0 \times 0.028 = 0.970, <1.0.$ |
| TLAAA022 | 4.3.2.11 | Leak-before-break (LBB) technique was applied for WCGS primary Reactor Coolant Loop piping in current licensing period. The original NUREG 0800 Standard Review Plan states that LBB cannot be applied to piping subject to Stress Corrosion Cracking (SCC).<br><br>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.<br><br>Please provide technical justification/discussion to demonstrate that LBB analysis remains valid for the period of extended operation. | 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.<br><br>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 | In this case "N/A" means not applicable, because no specified number of these events |

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| | | 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 "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. | was defined as a design limit.<br><br>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.<br><br>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.<br><br>**LRA Table 4.3-1, item 13, COMS (LTOP)**<br>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.<br><br>**LRA Table 4.3-1, item 12, Low Head Safety Injection (LHSI)**<br>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.<br><br>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.<br><br>"N/A" is therefore correct for LHSI Actuation.<br><br>Reference:<br>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 [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:<br><br>**a. In LRA Section 4.3.1.3, the** | **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.**<br>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. Some fatigue related design limits, such as criteria for selection of |

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| | | **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.**<br><br>**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.**<br><br>**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.**<br><br>**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.**<br><br>**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."** | break locations for HELB, are based on cumulative usage factors (CUF) less than 1.0. 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.<br><br>The description of corrective action limits and corrective actions has been included in section 4.3.1.3 of LRA Amendment 1.<br><br>**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.**<br>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.<br><br>The description of corrective action limits and corrective actions has been included in section 4.3.1.3 of LRA Amendment 1.<br><br>**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.**<br>These cross-references have been clarified in LRA Amendment 1.<br><br>**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.**<br>These statements have been clarified in LRA Amendment 1. Please note that these former page numbers have changed.<br><br>**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.** |

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| | | **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.**<br><br>**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.** | **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.**<br>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.<br><br>**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.**<br>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.<br><br>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.<br><br>"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.<br><br>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.<br><br>This noise event has been observed since Refueling Outage 5. Indicated severity has not been uniform between occurrences. This variation is expected due to several factors:<br>• The system operating sequence varies prior to each occurrence.<br>• Monitoring equipment and methods have changed due to upgrades. |

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| | | | • Data from some events has been partially lost due to monitoring equipment failures.<br>• Equipment has been modified, notably primary loop restraint changes and snubber removal, and reactor vessel head modification.<br><br>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.<br><br>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 | 4.3 | Ref. WCAP-14173 Global to Local & Transfer functions Rev. 3 - Nov. 1996.<br><br>1. During the audit, the staff reviewed basis document "WCAP-14173", which listed stresses transfer functions.<br>In general, the stress vector consists of 6 stress components ([sigma]xx, [sigma]yy, [sigma]zz, [tau]xy, [tau]yz, [tau]zx).<br>**a. Please explain why the stress transfer function, (e.g., Table E.2-1) contains only one value and the meaning of this stress.**<br><br>**b. Please justify how one stress component could be used to evaluate fatigue CUF.**<br><br>**2.** The report defines stress | **1a. Please explain why the stress transfer function, (e.g., Table E.2-1) contains only one value and the meaning of this stress.**<br>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 than vectors. This is acceptable because $\lvert A+B+C\rvert \le \lvert A\rvert+\lvert B\rvert+\lvert C\rvert$ 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.<br><br>**1b. Please justify how one stress component could be used to evaluate fatigue CUF.**<br>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 $S_{pure}$.<br><br>Further, in practice most components have a single dominant stress direction with |

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| | | transfer functions as stress intensity. **Please explain how stress intensity value could be used as input for the transfer function methodology.**<br><br>**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?**<br><br>**4. Please describe how the stress transfer functions were benchmarked for the components of Wolf Creek Generating Station.**<br><br>**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.)** | 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.<br><br>**2. Please explain how stress intensity value could be used as input for the transfer function methodology.**<br>Stress intensity is not used as input for the transfer functions. The transfer functions take as input:<br>(a) instrument data (and/or calculated system parameters),<br>(b) peak stress intensity ranges from design stress reports,<br>(c) total stress response for a thermal step transient (Green's Function)<br><br>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.<br><br>**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?**<br>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.<br><br>**4. Please describe how the stress transfer functions were benchmarked for the components of Wolf Creek Generating Station.**<br>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.<br><br>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 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. |

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| | | | 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.<br><br>**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].)**<br>For WCGS, the Transfer Functions for the pressurizer locations were developed by Westinghouse. 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.<br><br>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.<br><br>(A) The DSR would include consideration of pressure stress in its fatigue evaluation.<br>Let **P** = the maximum pressure from the DSR analysis, in psig, and<br>**S** = the corresponding pressure stress at the critical location,<br>Then:<br>$S_{pr}$ = **S/P** = (xxx)/(yyy) = 3.71<br><br>(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.<br>Let $\mathbf{T_{hot}}$ = the operating temperature, (est. 650°F)<br>$\mathbf{T_{cold}}$ = the stress-free temperature (usually 70°F), and<br>$\mathbf{M_x}$, $\mathbf{M_y}$, $\mathbf{M_z}$ = the moments computed for hot (operating) condition.<br>Then:<br>$S_{momxz} = \mathrm{sqrt}(\mathbf{M_x}^2 + \mathbf{M_z}^2)/(\mathbf{T_{hot}}-\mathbf{T_{cold}}) = (xxx)/(650-70) = 9.40$<br>$S_{momy} = \mathrm{abs}(\mathbf{M_x}^2)/(\mathbf{T_{hot}}-\mathbf{T_{cold}}) = (0.)/(650-70) = 0.0$<br><br>(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 |

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| | | | or a range of flow rates. The Green's Function is taken as the extracted stress response (vs. time) at the critical location. |