WOLF CREEK
NUCLEAR OPERATING CORPORATION

August 9, 2007

Matthew W. Sunseri
Vice President Oversight

WM 07-0067

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
3) Letter ET 07-0031, dated July 26, 2007, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket No. 50-482: Wolf Creek Generating Station License Renewal Application, Amendment 2

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 Program (AMP) audit was conducted the week of March 26, 2007 and the LRA Aging Management Review (AMR) audit was conducted 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).

Based on the results of the March 26 and May 7, 2007 TLAA audits, WCNOC modified Sections 4.1 and 4.3 of the LRA. WCNOC submitted these amended sections as Amendment 1 to the WCGS LRA in Reference 2. Following the July 9, 2007 TLAA audit, it was determined that an additional Amendment to the LRA was necessary. Amendment 2 is based on the responses provided in TLAA question and answer database compiled during the audits and submitted in Reference 3. Enclosure 1 provides the amended pages. For a complete copy of Chapter 4 of the WCGS LRA,

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please exchange the pages provided with the corresponding pages in References 1 and 2.

This letter contains no new commitments. Reference 3 provides the latest comprehensive commitment list in support of the WCGS LRA. This letter closes commitment number thirty-six on that list. If you have any questions concerning this matter, please contact me at (620) 364-4008, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

M W Sunseri

Matthew W. Sunseri

MWS /rlt

Enclosure 1- LRA, Amendment 2

cc: J. N. Donohew (NRC), w/e
V. G. Gaddy (NRC), w/e
B. S. Mallett (NRC), w/e
V. Rodriguez (NRC), w/e
Senior Resident Inspector (NRC), 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 9th day of Aug., 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

Wolf Creek Generating Station
License Renewal Application
Amendment 2

Amended Page(s)	Audit Question Number
4.1-5	TLAAA026
4.3-3	TLAAA027
4.3-11 through 4.3-12	TLAAA009
4.3-13 through 4.3-14	TLAAA028
4.3-20 through 4.3-37(1)	TLAAA004
4.3-23	TLAAA030
4.3-42(2)	RAI 4.3-3
4.6-4 through 4.6-5	TLAAA018

(1) The change for Audit Question TLAAA004 is reflected on pages 4.3-20 and 4.3-21. Pages 4.3-22 through 4.3-37 are provided due to repagination from the amended text.
(2) The change to Page 4.3-42 (paragraph above Table 4.3-5) was identified by WCNOC during development RAI 4.3-3 response.

4.1.4 Summary of Results

Sections 4.2 through 4.7 describe six general categories of TLAAs. They are listed in Table 4.1-1. They are presented in the order in which they appear in Sections 4.2 through 4.7 and following the order of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants."

NUREG-1800, Tables 4.1-2 and 4.1-3, list examples of analyses that could be TLAAs, depending on the applicant's current licensing basis (CLB). Table 4.1-2 summarizes the results of the WCNOC review of the analyses identified in NUREG-1800 Tables 4.1-2 and 4.1-3.

Table 4.1-1: List of TLAAs

TLAA Category	Description	Disposition Category(1)	Report Section
1.	Reactor Vessel Neutron Embrittlement		4.2
	Neutron Fluence, Upper Shelf Energy and Adjusted Reference Temperature (Fluence, USE, and ART)	ii, iii	4.2.1
	Pressurized Thermal Shock (PTS)	ii	4.2.2
	Pressure-Temperature (P-T) Limits	ii	4.2.3
	Low Temperature Overpressure Protection (LTOP)	ii	4.2.4
2.	Metal Fatigue		4.3
	Fatigue Management Program		4.3.1
	ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components		4.3.2
	Reactor Pressure Vessel, Nozzles, Head, and Studs	iii	4.3.2.1
	Control Rod Drive Mechanism (CRDM) Pressure Housings, Adapter Plugs, and Canopy Seals	i	4.3.2.2
	Reactor Coolant Pump Pressure Boundary Components	iii	4.3.2.3
	Pressurizer and Nozzles	iii	4.3.2.4
	Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses	iii	4.3.2.5
	ASME Section III Class 1 Valves	i, iii	4.3.2.6
	ASME Section III Class 1 Piping and Piping Nozzles	i, iii	4.3.2.7

Fatigue Design Curve with Margin for Uncertainties and Moderate Environmental Effects: The ASME Section III fatigue S-N curves (allowable alternating stress intensity versus number of cycles) are based on regression analysis of a large number of fatigue data points for samples strain-cycled in air. 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).

Bounding Parameters for Transients: Fatigue analyses assume a given number of cycles of each transient of a set of transient events, where each event is defined by limiting pressure and temperature transients and other load conditions. Since actual event cycles are seldom as severe as those considered in the analysis, the resulting stress ranges are lower. Therefore, the contributions to cumulative usage factor are also lower.

Actual Number of Event Cycles: The analytic limit for a fatigue analysis is a cumulative usage factor (CUF) of 1.0 at any location. The CUF is calculated as the sum of all contributing partial usage factors for the design basis number of each of the design basis cyclic loading events. Even if the analysis showed a calculated usage factor at the 1.0 limit for a location, and even if the design basis number of events were reached for one or more events of a set, but not for the remainder of the assumed events, some margin will remain to the 1.0 limit because not as many design basis events will have occurred as assumed by the analysis. Therefore they will not have contributed as much to the usage factor as the analysis assumed.

4.3.1.2 Present Status of Monitored Locations

The present WCGS fatigue management program was implemented in 1997. The usage factors calculated by the program include the effects of cycles incurred before the program was installed. The cycle count input to the program was accumulated from two periods. Effects were counted or estimated from the WCGS operating history for the period between initial cold hydro in February 1982 to the installation of the automated transient data acquisition system in March 1992. Data from the data acquisition system and from operating records were used thereafter, up to the implementation of the fatigue management program.

Cycle Counts

Table 4.3-1, "Significant Transient Cycle Limits Tracked by the WCGS Fatigue Management Program," includes the cycle counts to December 31, 2005. The cycle accumulations shown in this table indicate that the original design basis number of events should not be reached in a 60-year operating life, nor should the code usage factor limit of 1.0 be exceeded.

WCGS operating changes described in Section 4.3.2.4, "Pressurizer and Pressurizer Nozzles," have successfully mitigated surge line and pressurizer thermal stratification and

(e.g., 90%) of the design-specified number of cycles before the end of the next fuel 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.

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.

The enhancements described in this section will be required to address fatigue TLAAs in the period of extended operation. Wolf Creek Nuclear Operating Corporation will complete these program enhancements before the end of the current licensed operating period. Changes in available monitoring technology or in the analyses themselves may by that time permit different action limits and action statements, or may otherwise change the program features and actions required to address the fatigue TLAAs.

4.3.2 ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components

Fatigue analyses are performed for ASME Section III Division 1 Class 1 piping, vessels, heat exchangers, pumps, and valves and if applicable, their supports. Table 4.3-3 lists all Class 1 vessels, heat exchangers, pumps, piping and subcomponents subject to Class 1 analyses, and the subsection of this chapter which addresses each component.

The reactor vessel internals are not designed to ASME Section III Class 1 but are analyzed to ASME Section III Subsection NG. See Section 4.3.3.

Table 4.3-3 WCGS Class 1 Components and Piping

Component	WCGS Number(1)	Subsection
Reactor Pressure Vessel, Head, and Studs	BB-RBB01	4.3.2.1
Control Rod Drive Motor (CRDM) Housings	[BB-RBB01]	4.3.2.2
CRDM Head Adapter Plugs	[BB-RBB01]	4.3.2.2
CRDM Seismic Support Platform, Spacer Plates, and Tie Rods	-	No fatigue analysis
Reactor Coolant Pump Casings, Integral Supports, Main Flanges and Thermal Barrier Heat Exchangers(2)	BB-PBB01A, B, C, D	4.3.2.3
Pressurizer(2)	BB-TBB03	4.3.2.4
Steam Generators (Primary or Tube Side and Shell Side)(3)	BB-EBB01A, B, C, D	4.3.2.5
Valves(2)	See list in 4.3.2.6	4.3.2.6
Piping	-	4.3.2.7
Main Reactor Coolant Loop Piping Nozzles and Thermowells	-	4.3.2.7

Table 4.3-3 WCGS Class 1 Components and Piping

Component	WCGS number(1)	Subsection
Supports for Class 1 Piping and Valves	-	No fatigue analysis

Notes to Table
[1] Brackets indicate a subcomponent.
[2] Pressure-retaining bolting for the reactor coolant pumps, pressurizer, and valves is included in the component code analyses but is not described separately.
[3] The steam generator shell (steam) side is Class 2 but also received a Class 1 analysis.

4.3.2.1 Reactor Pressure Vessel, Nozzles, Head, and Studs

Summary Description

The WCGS reactor pressure vessel (RPV) is designed to ASME Section III, Subsection NB (Class 1), 1971 Edition with addenda through Winter 1972.

Pressure-retaining and support components of the reactor pressure vessel are subject to an ASME Boiler and Pressure Vessel Code, Division 1, Section III, fatigue analysis. This analysis has been updated to incorporate redefinitions of loads and design basis events, operating changes, power rerate, and minor modifications. The currently-applicable fatigue analyses of these components are TLAAs.

See Section 4.3.2.9, "Primary Coolant System Heatup Expansion Noise Events," for the evaluation of certain noise events affecting the fatigue analyses of the primary coolant system and RPVI.

Analysis

Effects of Power Rerate, T_{hot} Reduction, and up to 10 Percent Steam Generator Tube Plugging on the Vessel Fatigue Analysis

The WCGS power rerate modification included evaluation of a proposed reduction in normal operating hot leg temperature (T_{hot} reduction) and operation with up to 10 percent of steam generator tubes plugged. The code design report revision included review of operating conditions, including specified transient definitions for the original power rating, rerated power, T_{hot} reduction, and maximum and minimum tube plugging, to determine the most limiting parameters. Stresses and fatigue usage were calculated for these most limiting parameters. The calculated design stresses and fatigue usage factors in the revised design

Insurge-Outsurge Transients

Insurge-outsurge events during startup or shutdown can introduce cooler water through the pressurizer surge line into the pressurizer, against the wall previously heated by hot pressurizer water. This causes a significant thermal gradient in the pressurizer wall.

Surge effects in the pressurizer are mitigated by Technical Specification heatup and cooldown rate limits; and by the use of continuous spray during heatup and cooldown transients. Continuous spray maintains a small flow from the pressurizer to the hot leg during heatup and cooldown, which maintains a uniform fluid temperature below the pressurizer heaters and in the upper portion of the surge line to minimize thermal stratification. The heatup and cooldown rate operating limits and the use of continuous spray flow were instituted in 1993, and have been very effective in reducing fatigue usage accumulation in the lower pressurizer components since that time. Based on this experience, WCGS has concluded that a generic Westinghouse analysis of fatigue usage, including insurge/outsurge transient effects (Reference 4.9.18), is conservative for WCGS for 60 years of operation.

Effect of a Pinned Support on the Relief Line to BB-V-8010C

A pinned support in the discharge line from this pressurizer code relief valve had been recognized in 1987, and the effect of thermal cycles under this pinned condition on the pressurizer nozzle usage factor was calculated and included in the fatigue analysis and in the pressurizer design report. The pinned support also produced a plastic displacement in the relief valve line. An evaluation of the effect of the plastic displacement on the code fatigue analysis of the line found a small effect on usage factor in the line. The evaluation determined, by comparison that the additional effect on the nozzle was negligible (<0.001). Therefore no change to the pressurizer fatigue analysis was initiated.

Effect of a Pressurizer-Surge-Nozzle-to-Safe-End Weld, Safe End, and Safe-End-to-Surge-Line Weld Overlay

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 inspection and by flaw propagation 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.

Summary of Analyses

With the design basis set of transients, including the power rerate and T_{hot} modification and other effects described above, worst-case fatigue usage factors for the present design exceed 0.9 at three pressurizer locations.

Although the pressurizer surge and spray nozzles and pressurizer lower head are subject to significant operating thermal cycles from thermal stratification and insurge-outsurge transients not considered in the original code analysis, operating procedure changes have minimized these transients, and updated analyses confirm that fatigue usage factors in the affected pressurizer components will be within acceptable limits for the originally-specified design transient events, plus the number of these additional transient events expected for an operating life of 60 years.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The WCGS fatigue management program will track events to ensure either that the number of assumed events are sufficient and the usage factors are not exceeded, or that appropriate reevaluation or other corrective measures maintain the design and licensing basis.

The pressurizer surge nozzle, spray nozzle, and lower head may be subject to significant operating thermal stress cycles due to thermal stratification and insurge-outsurge cycles, and are therefore expected to be the limiting pressurizer components for fatigue. As a result the fatigue usage factors of these locations are specifically monitored.

Therefore the effects of fatigue in the pressurizer Class 1 pressure boundary and supports will be managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.5 Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses

Summary Description

The steam generators are designed to ASME Section III, Subsection NB (Class 1) and Subsection NC (Class 2), 1971 Edition with addenda through Summer 1973.

Pressure-retaining and support components of the primary coolant side of the steam generators are subject to an ASME Boiler and Pressure Vessel Code, Division 1, Section III fatigue analysis. Although the secondary side is Class 2, pressure retaining parts of the steam generator satisfy the Class 1 criteria, including the Class 2 secondary side boundaries. These analyses have been updated to incorporate redefinitions of loads and

design basis events, operating changes, power rerate, primary loop T_{hot} reduction, and minor modifications. The currently-applicable fatigue analyses of these components are TLAAs.

There are also some fatigue evaluations of feedwater lines and of the auxiliary feedwater tees. However, none of these evaluations have produced licensing basis commitments or safety determinations supported by fatigue analyses. Therefore the fatigue evaluations associated with feedwater lines and auxiliary feedwater tees are not TLAAs.

Analysis

Steam Generator Tube Code Fatigue Analysis Not a TLAA

The design of the steam generators includes a code fatigue analysis of the steam generator tubes. This analysis would be a TLAA if the safety determination depended upon it. However the code fatigue analysis has not proved sufficient to support the safety determination.

The various tube degradation mechanisms not anticipated in the original design have required stringent periodic inspection programs in order to ensure adequate steam generator tube integrity. The fatigue analysis is therefore no longer the basis of the safety determination that the tubes will maintain their pressure boundary function (Criterion 5).

Therefore, even in installations (such as WCGS) with excellent material and chemistry control, the safety determination for integrity of steam generator tubes now depends on managing aging effects by a periodic inspection program rather than on the fatigue analysis. Therefore the code fatigue analysis of the tubes is not a TLAA.

Steam Generator Fatigue Analysis Including Effects of Power Rerate, T_{hot} Reduction, and up to 15 Percent Steam Generator Tube Plugging

The WCGS power rerate modification included evaluation of a proposed reduction in normal operating hot leg temperature (T_{hot} reduction) and operation with up to 10 percent of steam generator tubes plugged. The current analysis encompasses these design limits, including up to 15 percent plugged tubes.

The current steam generator code design report reflects refinements in the analyses of some components and resulting reductions in calculated usage factors, including qualification of the primary manway studs and secondary closure bolting by test. It also includes effects of up to 15 percent steam generator tube plugging, revised design basis transients, revised seismic spectra, and feed line acoustic pressure pulse transients. Therefore the calculated design stresses and fatigue usage factors in the revised design report bound known operating conditions, at up to rerated power, with or without T_{hot} reduction, and for any level of tube plugging up to 15 percent.

With power rerate and the T_{hot} modification the worst-case usage factors calculated for the specified set of design basis transients exceed 0.9 in two steam generator locations.

However, excepting the tubes (for which the safety determination depends on managing aging effects by a periodic inspection program rather than on the fatigue analysis), fatigue usage factors in the steam generator components do not depend on flow-induced vibration or other effects that are time-dependent at steady-state conditions, but depend only on effects of operational and upset transient events. The WCGS fatigue management program tracks these operational and upset events to ensure that the design basis number of them is not exceeded without an appropriate evaluation and any necessary mitigating actions.

Primary Manway Studs, with Power Rerate and T_{hot} Reduction

The primary manway bolts were replaced by bored studs to permit hydraulic tensioning and measurement of preload. The replacement studs met code stress criteria, but high calculated usage factors would have required their 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.

Secondary Manway Bolts, Handhole (Inspection Port) Bolts, and Instrument Opening Bolts, with Power Rerate and T_{hot} Reduction

The secondary bolting is qualified for fatigue effects by analyses that apply the results of the primary manway stud tests.

The high calculated usage factor in bolting for these openings originally required their periodic replacement, as determined by the rate of transient cycle accumulation implied by the original 40-year design life. The increased rate of accumulation of fatigue usage factor with rerate and T_{hot} reduction reduced the secondary manway bolt replacement interval from 20 to 18 years. Since the bolting replacement interval was less than the design life, its basis was not, at that point, a TLAA.

However, the current code design report extends the primary manway bolting fatigue qualification tests to the secondary side bolting, and the basis of the safety determination for this bolting is qualification by test. The secondary-side steam generator pressure boundary bolts are no longer periodically replaced, therefore the application of the primary bolting fatigue test as the basis for secondary side bolting qualification for fatigue effects is now a TLAA. If the number of load cycles assumed by the evaluation, used in the fatigue test, is not exceeded, the qualification basis will remain valid.

Stub Barrels and Channel Heads Drilled and Tapped for DMIMS-DX Loose Parts Monitor Accelerometer Mountings

The code stress report includes effects of tapping 1/4" - 28 UNF2B holes in the stub barrel and channel head of each steam generator for mounting digital metal impact monitoring system (DMIMS) accelerometers. 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.

No TLAA in the Finite Element Analysis in Support of Feedwater Nozzle Thermal Stratification Transfer Functions for Stress-Based Fatigue Monitoring

NRC Information Notices 91-38 "Thermal Stratification in Feedwater System Piping" and 93-20 "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators" identified concerns with thermal stratification in feedwater piping and nozzles. The WCNOC resolution of this problem included thermal monitoring over several operating cycles to assess the severity of the concern. Analysis of these effects indicated more-rapid-than-design accumulation of fatigue usage factor, which then prompted plant operational changes and addition of a startup feedwater heating system.

A finite element analysis identified the limiting locations, and provided scaling for global-to-local transfer functions to permit stress-based fatigue monitoring at the limiting locations. However, thermal monitoring following the startup feedwater heating addition and operational changes did not indicate any significant alteration of the code analysis results. Therefore the code analysis does not reflect these effects. Since the finite element analysis used to develop the fatigue transfer functions has not been incorporated into the code stress analysis, the finite element analysis is not a TLAA.

Repair of Primary Chamber Drains

The 2005 refueling outage inspections found cracked welds at the connection of the C and D steam generator primary drains to the lower heads. The welds and the drain couplings in all four steam generators were excavated and replaced, and the steam generator analysis was amended. However the limiting usage factor at these drains is inside the heads (locations unaffected by the repair), and remains limiting at this location in the steam generator heads.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The fatigue analysis of the steam generator tubes does not support the safety determination and is therefore not a TLAA. Fatigue usage factors in other steam generator pressure boundary and Class 1 support components, and qualification of the primary manway studs by test, do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational, abnormal, and upset transient events.

Manway, Handhole, and Instrument Opening Bolting – Possible Requalification or Replacement

Appropriate corrective measures, which may include requalification or replacement, will ensure that the design basis of the bolting is maintained if fatigue monitoring indicates that the numbers of load cycles assumed by the qualification by test may be exceeded.

All Components

The WCGS fatigue management program will track events to ensure that appropriate reevaluation or other corrective action is taken if a design basis number of events is

exceeded, and will maintain a current record of cumulative usage factor for each monitored location.

Therefore, effects of fatigue in the steam generator pressure boundaries and their supports will be managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.6 ASME Section III Class 1 Valves

Summary Description

WCGS Class 1 valves (power-operated relief valves, pressurizer safety valves, control valves, motor- and air-operated valves, manual valves, and check valves) are designed to ASME Section III, Subsection NB, 1974 Edition and later addenda.

A review of WCGS Class 1 valve analyses and specifications found that fatigue analyses and possible TLAAs were performed only for the 6-inch pressurizer safety valves, for Class 1 check and gate valves over 4 inches nominal, and for one model of 1 1/2 inch angle globe valves.

However, the fatigue analysis for the 1 1/2 inch angle globe valves was not required by code or specification, is not discussed in any licensing basis document, and was therefore not the basis for a safety determination. Therefore the 1 1/2 inch angle globe valve fatigue analysis is not a TLAA. With that exception, no fatigue analyses were applied to any valves of four inches nominal inlet or less. Therefore no Class 1 fatigue analysis TLAAs support design of valves with inlets four inches or less. Conversely, fatigue analyses were applied to Class 1 valves with inlets greater than four inches, and all fatigue analyses of Class 1 valves greater than four inches are TLAAs.

Analysis

TLAA fatigue analyses or evaluations were performed for the following Class 1 valves:

Table 4.3-4 Class 1 Valves With TLAA Fatigue Analyses

Tag Number	Description	Normal Duty Ops N_A	CUF I_t
BB8010A, B, C	6" x 6" Pressurizer Safety Valves	$>10^6$	<0.4
BBPV8702A, B EJHV8701A, B	12" RHR Suction Gate Valves	$>10^6$	<1.0
BB8949A,B,C,D EJ8841A,B EP8818A,B,C,D	6" Swing Check Valves	$>10^6$	<0.4
BB8948A,B,C,D EP8956A,B,C,D	10" Swing Check Valves	820,000	<1.0

The allowed NB-3545.3 N_A normal duty operations far exceed those expected to occur. The calculated cumulative usage factors I_t for NB-3550 cyclic loads are less than the code limit of 1.0, and in all but the 12 inch RHR gate valves and 10 inch swing checks I_t is less than 0.4.

Disposition: Validation, 10 CFR 54.21(c)(1)(i); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Validation

The calculated worst-case usage factors for Class 1 pressurizer safety valves and for Class 1, 6 inch swing check valves indicate that the designs have large margins, and the pressure boundaries would withstand fatigue effects for at least two of the original design lifetimes. Therefore the design of these valves for fatigue effects is valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Aging Management

The calculated worst-case usage factors for the Class 1, 12 inch RHR suction gate valves and the Class 1, 10 inch check valves exceed 0.4. However, fatigue usage factors in these valves do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational, abnormal, and upset transient events. As discussed in Section 4.3.2.7, "ASME Section III Class 1 Piping and Piping Nozzles," the 40-year design basis number of events should be sufficient for 60 years of operation, of Class 1 piping systems containing valve. Therefore the calculated usage factors should not be exceeded. (The exceptions discussed in Section 4.3.2.7 are the surge line and surge line nozzle, which contain no valves).

The WCGS fatigue management program will ensure that calculated usage factors will not be exceeded, or that appropriate corrective action is taken if a design basis number of events is exceeded. Therefore, effects of fatigue in Class 1 valve pressure boundaries will be managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.7 ASME Section III Class 1 Piping and Piping Nozzles

Summary Description

Class 1 reactor coolant main-loop piping designed and supplied by Westinghouse is designed to ASME Section III, Subsection NB, 1974 edition with addenda through Winter 1975. The main loop piping fatigue analysis was performed to the 1974 edition with addenda through Winter 1975. The fatigue analyses of piping outside the main loop used code addenda through Summer 1979.

These analyses have been updated from time to time to incorporate redefinitions of loads and design basis events, operating changes, power rerate, and minor modifications. The currently-applicable fatigue analyses of these components are TLAAs.

For fatigue in the pressurizer surge line see Section 4.3.2.8, "Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification."

For the evaluation of certain noise events affecting the fatigue analyses of the primary coolant system and reactor pressure vessel see Section 4.3.2.9, "Primary Coolant System Heatup Expansion Noise Events." The evaluation of these noise events found no effect on the primary coolant piping fatigue analysis.

Analysis

In the primary coolant system and large-bore emergency core cooling (ECCS) lines the attachment welds to the reactor vessel inlet and outlet nozzles, and the primary coolant system ECCS injection nozzles (Loop 1 and 4 CVCS charging nozzles, BIT (HHSI) nozzles, and accumulator safety injection (ACCSI) nozzles) have the most limiting calculated design basis usage factors. In a number of these nozzles, the analysis of record was refined only to the level necessary to demonstrate a usage factor just under 1.0. The high usage factors in the ECCS injection nozzles are primarily due to transient thermal stresses from normal operating and upset injection events.

Calculated design basis usage factors in smaller Class 1 lines also exceed 0.9 at a number of locations, in many cases due to operating transients that specifically affect the location.

With the exception of the thermowells and pressurizer surge line nozzle discussed in this section and the pressurizer surge line discussed in Section 4.3.2.8, "Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification," fatigue usage factors in these components do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational, abnormal, and upset transient events. Since the WCGS fatigue management program will track these events, the design basis fatigue usage factor limit (1.0) will not be exceeded in these locations without an appropriate evaluation and any necessary mitigating actions.

Analysis of Supports

Only the pressurizer surge line was re-evaluated to a code edition and addendum (1986, no addenda) which in some cases would have required design of the supports for stress limits based on a finite number of lifetime load cycles. However, the original code of record (1974 W'75, fatigue analysis to 1977 S'79) was the same as that for other Class 1 lines and did not invoke this requirement, and as permitted by code rules, the later edition was not invoked for the support reanalysis. The supports were analyzed to the 1974 W'75 code of record. See Section 4.3.2.8, "Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification."

Effects of NRC Bulletin 88-11 Thermal Stratification on the Hot Leg Pressurizer Surge Line Nozzle

The current code analysis includes this effect. See also Section 4.3.2.8. The WCGS fatigue management program calculates fatigue usage factor in this nozzle.

Effects of Power Rerate, T_{hot} Reduction, and Allowance for Increased Steam Generator Plugging on the Piping Fatigue Analyses

Evaluations performed to incorporate the effects of power rerate, T_{hot} reduction (hot leg normal operating temperature reduction), and steam generator tube plugging selected parameters and transient descriptions that envelope worst case conditions for original power rating, rerated power, original T_{hot}, proposed (but not implemented) reduced T_{hot}, and steam generator tube plugging up to 10 percent. Therefore, the analysis results are conservative for any combination of these conditions.

Charging Lines and Nozzles

In 1990 Westinghouse identified concerns with CVCS injection path switching and containment isolation testing practices that might introduce a larger-than-design number of significant thermal transients in these nozzles. WCGS therefore revised operating procedures to ensure that significant injection nozzle thermal cycles are minimized.

RTD Nozzles

The Resistance Temperature Detector (RTD) piping has been removed and these nozzles will therefore accumulate no significant additional fatigue usage factor. Thermowells have been added at some of these nozzles, as described below.

Thermowells added at RTD Nozzles

The modification that removed RTD bypass piping added thermowells at the 12 primary loop hot leg RTD scoop nozzles (3 nozzles per hot leg) and at the 4 cold leg RTD nozzles, and capped the 4 return nozzles to the crossover legs. The thermowells were analyzed for fatigue due to flow-induced vibration. The maximum calculated usage factor for a 40-year life at 75 per cent availability is 0.025, in the cold leg thermowells

Fatigue due to these loads is proportional to operating time. The worst-case usage factor can therefore be projected to and validated for a 60-year life. The 90 percent capacity factor now expected requires no more than about 98 percent availability. Hence the worst usage factor in any of the RTD thermowell locations would be

$$CUF_{60} = CUF_{40} \times 0.98/0.75 \times 60/40$$

$$CUF_{60} = 0.025 \times 0.98/0.75 \times 60/40 = 0.049,$$

and therefore remains negligible.

Effects of NSAL-94-025 Reactor Coolant Pump Support Column Tilt on Main Loop Piping and Supports

Westinghouse identified a concern that reactor coolant pump support column tilt may have an adverse effect on main loop piping thermal stresses during heatup and cooldown transients. The Westinghouse evaluation found a large increase in the crossover and cold leg stresses at the reactor coolant pump, and a significant change in the load of the tilted column; but since original analysis stresses were low the effects on stresses and usage factors would not affect code compliance or the conclusions of the leak-before-break analysis. See also Section 4.3.2.11, "Fatigue Crack Growth Assessment in Support of a Fracture Mechanics Analysis for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures".

Disposition: Validation, 10 CFR 54.21(c)(1)(i); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Validation

The fatigue analysis of primary loop thermowells has been validated for the period of extended operation as described above, in accordance with 10 CFR 54.21(c)(1)(i).

Aging Management

With the exception of the thermowells and surge line nozzle discussed above and the pressurizer surge line discussed in Section 4.3.2.8, "Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification," usage factors in Class 1 piping pressure boundaries do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational, abnormal, and upset transient events.

The WCGS fatigue management program will ensure either that the original design basis number of events or usage factor is not exceeded, or that appropriate reevaluation or other corrective action is taken.

Therefore, effects of fatigue in the Class 1 piping pressure boundary will be managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.8 Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

NRC Bulletin 88-11 "Pressurizer Surge Line Thermal Stratification" requested that licensees "establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification" and required licensees to "inform the staff of the actions taken to resolve this issue."

A similar earlier Bulletin 88-08 "Thermal Stresses in Piping Connected to Reactor Coolant System" requested that licensees review the primary coolant pressure boundary and connected interfaces for possible effects of thermal cycles due to leaking interface valves. WCGS installed temperature monitoring to detect leakage past the auxiliary spray valve. Monitoring has not prompted a revision to the piping analysis.

See Section 4.3.2.4, "Pressurizer and Nozzles," for effects on the pressurizer surge nozzle. See Section 4.3.2.7, "ASME Section III Class 1 Piping and Piping Nozzles," for effects on the hot leg surge nozzle.

Summary Description

The original surge line fatigue analysis used code addenda through summer 1979. The surge line design was re-analyzed to the 1986 code in response to the NRC Bulletin 88-11 thermal stratification concerns. This analysis was later reevaluated for effects of snubber removals. The results of these analyses have been incorporated into the piping and main-loop nozzle code design reports.

Winter 1982 and later code addenda provide stress limits for high-cycle fatigue of Class 1 supports, under Subsubarticle NF-3330. However the re-evaluation of the surge line for NRC Bulletin 88-11 did not retroactively impose these requirements. Therefore no TLAA arises for design of the supports.

Analysis

Effects of Thermal Stratification on the Surge Line Piping Fatigue Analysis

The maximum calculated CUF at any location in the surge lines, under the current analysis of record, including thermal stratification effects, is less than 0.1.

Effects of Power Rerate and T_{hot} Reduction on the Surge Line Piping Fatigue Analysis

The evaluation of these modifications found that the resulting changes in temperature ranges have negligible effect on the surge line analysis.

Effect of a Pressurizer-Surge-Nozzle-to-Safe-End Weld, Safe End, and Safe-End-to-Surge-Line Weld Overlay

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 fatigue usage 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. The flaw propagation analyses are not TLAAs.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The surge line is subject to fatigue monitoring. The WCGS fatigue management program will ensure either that the usage factor remains valid for the period of extended operation or that appropriate corrective measures maintain the design and licensing basis. Therefore, effects of fatigue in the Class 1 surge line will be managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.9 Primary Coolant System Heatup Expansion Noise Events

Summary Description

Since 1990, abrupt audible events have been heard inside containment at WCGS toward the end of primary system heatups. They have been attributed to an abrupt release of differential expansion energy, originally believed to be at the crossover piping support saddle shims, later found to have probably also occurred between the reactor vessel support pads and shoes, under the vessel main loop nozzles. The evaluation of these effects on the vessel, piping, nozzle, and component fatigue analyses is a TLAA, as is the projection of shakedown effects in a reactor vessel support element.

Analysis

The driver for these events was modeled as the simultaneous sudden release of compressive loads on the reactor vessel nozzles. The resulting piping, support, and nozzle loads are within allowable limits, with the exception of a local region of the reactor vessel support cooling box, described below.

For purposes of evaluating fatigue effects the analysis assumed 330 such heatup noise events would occur in a 60-year design life.

Monitoring for the noise occurrence from Refuel 5 through Refuel 14 has detected 16 noise events, or about 1.6 per refueling cycle. Assuming a total of 30 cycles have already been expended, 300 cycles remain for the remaining life of the plant. Plant life from the present to the end of the extended operating period is an additional 40 years, or about an additional 27 refuelings at the present 18-month cycle. Up to 11 noise events could therefore occur per refueling cycle, at the magnitude assumed in the fatigue calculation, and still remain within the limits assumed by the fatigue calculation.

Reactor Pressure Vessel Structural Analysis

The increase in cumulative usage factor (CUF) in the affected inlet and outlet nozzle-to-shell junctures was calculated by combining the effects of 330 such peak stress range events with the appropriate ranges of related events from the original vessel fatigue analysis, under the Code stress range combination rules for fatigue. The resulting total CUFs are nominal, about 0.11 for the inlet nozzles, 0.18 for the outlet nozzles.

Reactor Coolant Loop Piping Analysis

Resulting stresses in piping are much less than the endurance limit and the resulting moment stress ranges at piping nozzle welds are less than the T-Z limit. Therefore, the events have no effect on fatigue usage nor on the conclusions of the piping analysis.

Reactor Coolant Loop Leak-Before-Break (LBB) Analysis

The recommended LBB margins are maintained. Therefore the conclusions of the LBB analysis remain valid.

Reactor Coolant Loop Primary Component Supports Evaluation

All of the primary equipment supports were qualified for normal and upset allowables for the sudden-release loads of the noise event.

Reactor Vessel Support Cooling Box Evaluation

The support cooling box is not a pressure boundary component. A local region of the cooling box bearing on the imbedded steel, may have exceeded yield, but shakedown is occurring or has occurred and the support is and will remain stable. This was confirmed by an elastic-plastic shakedown analysis, and is indicated by the fact that the event occurred at increasingly higher temperatures with successive heatups, and finally at the same temperature for the last three heatups prior to the May 2001 report date.

Steam Generator Primary Nozzle Evaluation

The effect of these noise events on the steam generator primary nozzle fatigue analysis was evaluated assuming 330 noise events might occur through the end of an extended 60-year licensed operating period. The sum of the added fatigue usage factor due to the noise events, plus the originally-calculated 40-year usage factor, is only 0.063 at the worst location in the primary nozzles. If the originally-calculated 40-year usage factor were multiplied by 1.5 to account for the increased life, the sum of these two values would be only 0.087.

Results of the Noise Event Monitoring Program

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 data to date indicates 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, and no increase in event severity.

Since the original WCGS LRA was filed, WCNOC has conducted a preliminary examination of Refuel 15 monitoring data. These results introduced some uncertainty in the statement that previously appeared under this subheading 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 Refuel 5. Indicated severity has not been uniform between occurrences. This variation is 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 Refuel 15 indicate somewhat higher responses than those observed during Refuels 13 and 14, and with these uncertainties, WCGS no longer concludes that there have been "apparent declines" in event severity. However, even with these uncertainties and the Refuel 15 data, the measured magnitudes and characteristics of these events collected over the period from Refuel 5 through Refuel 15 continue to indicate that effects are very limited and that the characteristics remain consistent. Therefore WCNOC concludes that results of previous evaluations remain valid and will continue to remain valid.

Effects of Power Rerate and T_{hot} Reduction on the Analysis of Effects of the Noise Events

The rerate report found that revised fatigue results "...accounted for both the noise program loadings and the modified design thermal transient conditions of the rerating program," and that other effects of rerate on the analysis of these events are negligible.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Reactor Pressure Vessel, Reactor Coolant Loop, Primary Loop Component Supports, and LBB Analyses

The evaluation found that the effect of these events on the reactor pressure vessel and reactor coolant loop and support fatigue analyses, and on the reactor coolant loop LBB analysis, is zero or insignificant for the period of extended operation. The effect of these events has therefore been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Reactor Vessel Supports

The effect of this event on vessel supports is within normal and upset allowables, with the exception of a local region of the reactor vessel support cooling box. This region is shaking down or has shaken down to a stable response to these events, and will therefore be suitable for the period of extended operation. The effect of these events has therefore been

projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Steam Generator Primary Nozzles

The effect of these events on the steam generator primary nozzles has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.2.10 High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor

Summary Description

Selection of pipe failure locations for evaluation of the consequences of a high energy line break on nearby essential systems, components, and structures, except for the reactor coolant loop, is in accordance with Regulatory Guide 1.46, and NRC Branch Technical Positions ASB 3-1 and MEB 3-1.

A revised stress analysis also permitted omission of the pressurizer surge line intermediate breaks.

A leak-before-break analysis (LBB) eliminated large breaks in the main reactor coolant loops. See Section 4.3.2.11, "Fatigue Crack Growth Assessment in Support of a Fracture Mechanics Analysis for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures."

Analysis

With the stated exception of the reactor coolant system primary loops, the citation of MEB 3-1 means that breaks in piping with ASME Section III Class 1 fatigue analyses are identified based on a limiting stress criterion (a break is assumed if the ASME Section III NB-3653 Equation (10), (12), or (13) stress range between any two load sets is greater than 2.4 S_M); and on a cumulative usage factor criterion (a break is assumed if the cumulative usage factor exceeds 0.1). Therefore, the location determinations that depend on usage factor are time-dependent and are TLAAs.

The surge line intermediate break locations were eliminated based on usage factor. The most recent piping analysis confirmed the elimination of these break locations. Therefore, the analysis that justified the elimination of these intermediate locations in the surge line is a TLAA.

The same would be true of other line sections with no intermediate locations with fatigue usage factors above 0.1, if this analysis result were used to eliminate intermediate breaks - that is, the determination that there are no intermediate breaks in these sections based on a low usage factor would, for the same reason, be a TLAA. However, no additional cases similar to the surge line occur in the WCGS licensing basis. Therefore, the scope of these

HELB-location TLAAs is limited to ASME Section III Class 1 piping analyses of other than the RCS primary coolant loops, including the surge line.

WCGS has containment penetration break exclusion regions ("no break zones"). However, these contain no ASME Section III Class 1 piping with fatigue analyses. Therefore, their qualification is based only on calculated stress and the break locations in these no break zones are independent of time and are not supported by a TLAA.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

Break locations which depend on usage factor, and their absence in the surge line, will remain valid as long as the calculated usage factors are not exceeded.

The WCGS fatigue aging management program will ensure that the calculated fatigue usage factors upon which the HELB break locations are based, and the HELB locations, will remain valid for the period of extended operation, or that appropriate corrective measures maintain the design and licensing basis.

4.3.2.11 Fatigue Crack Growth Assessment in Support of a Fracture Mechanics Analysis for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures

Summary Description

A leak-before-break analysis eliminated the large breaks in the main reactor coolant loops, which permitted omission of evaluations of their jet and pipe whip effects. This permitted omission of large jet barriers and whip restraints. The containment pressurization and equipment qualification analyses retained the large-break assumptions.

The dynamic effects from postulated pipe breaks have been eliminated from the structural design basis of the reactor coolant system primary loop piping, as allowed by revised General Design Criterion 4. The elimination of these breaks is the result of the application of leak-before-break (LBB) technology which has been approved for WCGS by the NRC.

The final licensing basis LBB submittal for WCGS is the proprietary WCAP-10691, "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Callaway and Wolf Creek Plants."

The NRC approval of this use of LBB at WCGS was granted with the Safety Evaluation Report for the original license as a 10 CFR 50.12 exemption from parts of General Design Criterion 4 (GDC-4). See Supplement 5 Section 3.6.1.1 of the NUREG-0881 SER for the original WCGS operating license.

The fracture mechanics analysis is not time-dependent and therefore is not a TLAA. However, the final LBB submittal is also supported by a fatigue crack growth assessment for a 40-year design life (WCAP-10691 Section 6.0), which is a TLAA.

There is no licensing basis evaluation of embrittlement of the cast reactor coolant piping or other cast austenitic stainless steel (CASS) at WCGS, apart from the LBB question.

Analysis

Fracture Mechanics Analysis (not a TLAA)

The fracture mechanics analysis depends in part on a material property, the crack initiation energy integral, J_{IN}. The primary coolant loops at WCGS are SA 351 Grade CF8A cast stainless steel, which at PWR operating temperatures is subject to time-dependent thermal embrittlement. Embrittlement reduces the J_{IN} integral.

Thermal embrittlement effects depend approximately logarithmically on time (more rapid initial change, achieving a saturation value after a long time). The available margins permitted use of a saturation value of J_{IN} for this analysis. Since a saturated J_{IN} is not time-dependent, this fracture mechanics analysis is not a TLAA.

Other supporting parts of the fracture mechanics analysis are supported in part by calculation of J_{IN} values for WCGS for a 40-year life. These supporting analyses are therefore time-dependent. However, the conclusion and safety determination of the LBB analysis does not depend on these supporting time-dependent analyses. Therefore the supported fracture mechanics analysis is not a TLAA. Additionally, since the supporting analyses do not determine the result of the safety determination, the supporting analyses are also not TLAAs.

Fatigue Crack Growth Assessment

The final LBB submittal for WCGS includes a fatigue crack growth assessment for a typical-plant case representative of WCGS, for a range of materials at a typical location. The analysis includes estimates of effects of the reactor coolant environment, and concludes that fatigue crack growth effects will be negligible. The evaluation of these typical 40-year LBB fatigue crack growth effects assumed load transients, stresses, and numbers of transient events which are representative of the WCGS reactor coolant system primary loop design.

Effects of Power Rerate and T_{hot} Reduction on the LBB Analysis

The power rerate and T_{hot} reduction modifications had no effects on the LBB analysis.

Effects of the Primary System Heatup Expansion Noise Events on the LBB Analysis

The evaluation of effects of the primary system noise events considered possible effects on the LBB analysis and found that recommended LBB margins are maintained, and that the conclusions of the LBB analysis remain valid. See Section 4.3.2.9, "Primary Coolant System Heatup Expansion Noise Events."

Effects of NSAL-94-025 Reactor Coolant Pump Support Column Tilt

Westinghouse identified a concern that reactor coolant pump support column tilt may have an adverse effect on main loop piping thermal stresses during heatup and cooldown transients. As described in Section 4.3.2.7, "ASME Section III Class 1 Piping and Piping Nozzles," the Westinghouse evaluation found a large increase in the crossover and cold leg stresses at the reactor coolant pump, and a significant change in the load of the tilted column; but since original analysis stresses were low the effects on stresses and usage factors would not affect code compliance or the conclusions of the LBB analysis.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The LBB analysis found that fatigue crack growth effects will be negligible. The basis for evaluation of fatigue crack growth effects in the LBB analysis will remain unchanged so long as the number of occurrences of each transient remains below the number assumed for the existing analysis of fatigue crack growth effects.

The WCGS fatigue aging management program will ensure that the number of occurrences of each transient cycle in the primary loop piping remains below the number specified by the design specifications during the period of extended operation, and therefore below the number assumed for the existing analysis of fatigue crack growth effects; or that appropriate corrective measures maintain the design and licensing basis. Therefore, the effects will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.3 ASME Section III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals

Summary Description

The WCGS reactor vessel internals were designed after the incorporation of Subsection NG into the 1974 Edition of Section III of the ASME Boiler and Pressure Vessel Code. The design meets the intent of paragraph NG-3311(c); that is, design and construction of *core support* structures meet Subsection NG in full, and other internals are designed and constructed to ensure that their effects on the core support structures remain within the core support structure code limits.

The four WCGS 10-inch accumulator and RHR cold leg safety injection nozzles (ACCSI nozzles) correspond to the NUREG/CR-6260 "inlet transition." Therefore, the WCGS evaluation includes these four nozzles and both charging nozzles. See Table 4.3-5.

Analysis of Sample Locations

WCGS performed plant-specific calculations for the seven sample locations applicable to WCGS identified in NUREG/CR-6260 for newer Westinghouse plants. WCGS evaluated effects of the reactor coolant environment on fatigue calculations using the appropriate F_{en} factors from NUREG/CR-6583 for carbon and low-alloy steels and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each of these seven locations. See the notes to Table 4.3-5 for the method used for each F_{EN} multiplier.

At the first location, the vessel lower head to shell juncture, the expected 60-year fatigue usage factor was determined by multiplying the design basis 40-year usage factor times 1.5. All others were projected from historical and current rates of accumulation of transient cycles and usage factors, using either the cycle-based method or the stress-based method of the fatigue monitoring program described in Section 4.3.1, "Fatigue Management Program." The inlet, outlet, safety injection (BIT), and accumulator-RHR nozzle predictions used the cycle-based method. The remaining hot leg and charging nozzle predictions used the stress-based method.

Table 4.3-5 Summary of Fatigue Usage Factors at NUREG/CR-6260 Sample Locations, Adapted to WCGS

Location	Material	CUF Expected at 60 Years	F_{EN}	Expected 60-year CUF with F_{EN}
Reactor Vessel Lower Head to Shell Juncture (Not Monitored)	SA-533 Gr. B Cl. 1	0.1005	2.45(1)	0.2462
Reactor Vessel Primary Coolant Inlet Nozzle	SA-508 Cl. 3	0.13467	2.45(1)	0.3299
Reactor Vessel Primary Coolant Outlet Nozzle	SA-508 Cl. 2	0.21597	2.45(1)	0.5291
Surge Line Highest-CUF Location, Hot Leg Nozzle	SA-182 F316N	0.05849	8.593(2)	0.50257
Charging Nozzles -Normal, Loop 1	SA-182 F316N	0.15863	5.486(2)	0.87028
Charging Nozzles -Alt., Loop 4	SA-182 F316N	0.09847	5.486(2)	0.54024

BC-TOP-1 Loading Condition V - Normal Penetration Thermal Gradient Plus Startup-Shutdown - 100 Cycles

For BC-TOP-1 Part II Loading Condition V (normal thermal gradient plus operating cycle), the analysis compares the allowed value of S_a from the S-N diagram for 100 cycles, 200,000 psi, to the calculated maximum alternating stress intensity for this load combination,

$$S'_a = S_a K_e = 52{,}800 \text{ psi}$$

—which is acceptable, since S'_a is less than S_a [Ref. 4.9.13, Part II Section 5.3.5.(c)].

Again, BC-TOP-1 did not calculate a usage factor for this case, but referring to the ASME Section III-1971 Figure I-9-1 S-N diagram, about 3,600 cycles are allowed for the 52,800 psi applied stress range S'_a, for an equivalent usage factor of

$$100/3{,}600 = 0.028$$

—for the 100 cycles assumed.

Estimated Number of BC-TOP-1 Loading Condition V Events in 60 Years

The operating history to date indicates that the original design basis 100 operating cycles assumed for main steam penetrations will be adequate for the 60-year extended operating period. Table 4.3-1, "Significant Transient Cycle Limits Tracked by the WCGS Fatigue Management Program," Item 1, shows only 27 startup cycles in the 19 years through 2004. 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-shutdown cycles at two per refueling. Since there are no inboard MSIVs in this PWR design, main steam penetration thermal cycles do not occur separately from reactor coolant system startup-shutdown cycles. Therefore, the same number of main steam penetration full-range thermal cycles (BC-TOP-1 Part II "Condition V" events) is expected in 60 years.

Combined Effect of BC-TOP-1 Loading Conditions IV and V with a Large Increase in the Number of Condition V Events

The faulted Loading Condition IV event would not affect the fatigue calculation for an ASME Section III Class 1 pressure boundary component. The fatigue evaluation for this load is also not time-dependent and is therefore not a TLAA; but the main steam penetrations are not ASME Section III Class 1 components and must also function following this faulted event. To assess the combined effect, the effect of the Condition V normal operating loads is therefore combined with the effect of the Condition IV faulted event.

Even if the main steam penetrations experience a very large number of BC-TOP-1 Part II Condition V events, an examination of the analysis basis demonstrates that the design is more than adequate. The Condition V events do not contribute significantly to usage factor, and a revised BC-TOP-1 analysis for any reasonably expected increase in the number of

these events demonstrates adequate margin to the stress limit determined by the elastic-plastic analysis.

As noted above, ASME Section III-1971 Figure I-9-1 indicates that about 3,600 cycles could be accommodated by the BC-TOP-1 simplified elastic-plastic analysis at $S_a' = 52{,}800$ psi, as calculated for the worst-case Condition V event.

Alternatively, adding the usage factor estimated above for 10 Loading Condition IV events to 25 times the usage factor estimated above for the original 100 normal Condition V operating cycles, for as many as 2500 Condition V events, would still result in a usage factor estimate less than 1.0, and therefore would not affect the conclusion of the analysis:

$$0.270 + 25.0 \times 0.028 = 0.970.$$

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The original number of thermal cycles assumed for the main steam line penetrations is adequate for the period of extended operation; and there is more than sufficient margin in the design for any possible increase in operating cycles above the original estimate. The design of the main steam penetrations is therefore valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).