APS

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear
Generating Station

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102-06210-JHH/GAM
June 29, 2010

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Sirs:

**Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528, 50-529 and 50-530
Response to June 2, 2010, Request for Additional Information Regarding Metal Fatigue for the Review of the PVNGS License Renewal Application, and License Renewal Application Amendment No. 18**

By letter dated June 2, 2010, the NRC issued a request for additional information (RAI) related to the PVNGS license renewal application (LRA). Enclosure 1 contains APS's response to the June 2, 2010, RAI. Enclosure 2 contains PVNGS LRA Amendment No. 18 to include new commitment nos. 57 and 58 in Table A4-1 for environmental factor (F_{en}) calculations to reflect the responses to RAI 4.3-6.

In addition, LRA Amendment No. 18 in Enclosure 2 contains (1) updates to LRA Section A3.2, Metal Fatigue Analysis, to reflect the changes submitted in LRA Amendment Nos. 14 and 16 in APS letter nos. 102-06175, dated April 28, 2010, and 102-06198, dated May 27, 2010, respectively, and (2) minor editorial corrections to LRA Table 4.3-3 and Section 4.3.5. Enclosure 3 contains LRA Amendment No. 18 underline and strike-out markup pages.

Should you need further information regarding this submittal, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

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NLR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 6/29/10
(date)

Sincerely,

JHH/RAS/GAM

Enclosures:

1. Response to June 2, 2010, Request for Additional Information Regarding Metal Fatigue for the Review of the PVNGS License Renewal Application
2. Palo Verde Nuclear Generating Station License Renewal Application Amendment No. 18 – Clean Pages
3. Palo Verde Nuclear Generating Station License Renewal Application Amendment No. 18 – Markup Pages

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	L. K. Gibson	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	L. M. Regner	NRC License Renewal Project Manager
	G. A. Pick	NRC Region IV (electronic)

ENCLOSURE 1

Response to June 2, 2010, Request for Additional Information Regarding Metal Fatigue for the Review of the PVNGS License Renewal Application

NRC RAI 4.3-1

Issue

In the public meeting between Arizona Public Service Company (APS) and the U.S. Nuclear Regulatory Commission (NRC) held on Thursday, May 6, 2010, APS indicated that it had updated the design basis transients for the metal fatigue time-limited aging analysis (TLAA) to be consistent with those listed in the updated final safety analysis report (UFSAR) for the facility. Further, APS stated that the updated transient projection basis is based on the applicant's updated transient recount activities for the TLAA. The applicant clarified that the 25 percent assumed transient occurrence basis used in the original TLAA was only applied to five or six transients for which recount data could not be found.

Request

Clarify which of the transients in Tables 4.3-2 and 4.3-3 of the LRA (as modified by Amendment 14) the 25 percent assumed transient occurrence basis remains applicable to and justify why the application of this assumption is considered to yield a conservative 60-year cycle occurrence basis for these transients.

APS Response to RAI 4.3-1

APS elected to retain the 25% assumed transient accumulation for fourteen transients, as shown in Table RAI 4.3-1 below.

In nine cases (Item Nos. 13, 26, 27, 57, 59, 60, 80, 82, and 83 in LRA Tables 4.3-2 and 3) the review of logs, Licensee Event Reports (LERs), NRC Monthly Operating Reports, and test records revealed either no occurrences of the events between 1985 and the end of 1995, or confirmed that the 25% assumption was not exceeded. The assumption of 25% was retained in these nine cases because the events are rare, there is ample margin to the design limit, and there was a desire to incorporate conservatism into the recount process.

In three cases (Item Nos. 8, 9, and 18 in LRA Tables 4.3-2 and 3), the counted accumulation of events between 1995 and 2005 were less than 5% of the limiting values (less than 20% of the values assumed for 1985 to 1995). The disparity in event totals validated the conservatism of the original assumption. In the case of shutdown cooling initiation, the total number of RCS cooldowns was known and provided additional assurance the assumption was conservative.

The remaining two events (Item Nos. 20 and 21 in LRA Tables 4.3-2 and 3) are the low pressure safety injection (LPSI) pump test run and the high pressure safety injection (HPSI) pump test run. The 25% assumption is an adequate representation of the period 1985 through 1995 for the following reasons:

- The tests occur at scheduled intervals with occasional tests being performed for post maintenance testing, so the rate of occurrence is fairly constant lending itself to reasonably accurate prediction of accumulation.

- The 25% assumption for a ten year period (1985 – 1995) is approximately equal to the actual count for the ten year period 1995 – 2005 for each unit. The actual LPSI pump runs counted from 1995 through 2005 in Unit 3 exceeded the 25% assumption by two occurrences (127 actual between 1995 and 2005 vs. 125 assumed for 1985 through 1995). However, Unit 3 only operated eight years between 1985 and 1995, so it is conservative to assume 125 occurrences (25%) for the eight years of operation (or an assumed average of 15.6 occurrences per year for 1985-1995 vs. the actual average of 12.7 occurrences per year between 1995 and 2005).

Please note that transient totals were projected to the end of the period of extended operation for information only. Some projections indicate that certain allowable cycles and fatigue limits may be approached during the period of extended operation. Therefore, specific and targeted action limits will be necessary to ensure actual fatigue limits are not exceeded. Corrective actions will be triggered by the action limits that will be established in the enhanced Metal Fatigue of Reactor Coolant Pressure Boundary program (B3.1).

Table RAI 4.3-1, Transients Retaining the 25% Assumed Transient Accumulation

1 LRA Table 4.3-3 Item No.	2 Transient Title	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
Transients With 25% Accumulation Assumption Justified by Operational History After 1995							
8	Startup of one reactor coolant pump at hot standby conditions	1000	273	281	275	936	This total includes actual data from 1995-2005 and the 25% assumption (250 events) for 1985-1995. The assumption of 25% was retained because inconsistencies in logs precluded an actual count. The 25% assumption appears very conservative based on the fact that the accumulated events counted between 1995 and 2005 are approximately 1/10th the number assumed for 1985-1995.
9	Coastdown of one reactor coolant pump at hot standby conditions	1000	269	275	268	916	This total includes actual data from 1995-2005 and the 25% assumption (250 events) for 1985-1995. The assumption of 25% was retained because inconsistencies in logs precluded an actual count. The 25% assumption appears very conservative based on the fact that the accumulated events counted between 1995 and 2005 are approximately 1/10th the number assumed for 1985-1995.
18	Startup of SDC system from standby to shutdown cooling (RCS >200F) to shutdown cooling (RCS <200F) to standby	500	136	148	145	493	This total includes actual data from 1995-2005 and the 25% assumption (125 events) for 1985-1995. Log entries did not always indicate which train of shutdown cooling was used or when a train swap occurred. Since the actual number of plant cooldowns was approximately half the total of assumed and counted SDC initiations the 25% assumption was retained for conservatism. The assumption of 25% appears very conservative based on the fact that the accumulated events counted between 1995 and 2005 are approximately 10% to 20% of the number assumed for 1985-1995, and the fact that the total number of plant cooldowns between 1985 and 2005 is approximately 60 in each unit.

Table RAI 4.3-1, Transients Retaining the 25% Assumed Transient Accumulation

1 LRA Table 4.3-3 Item No.	2 Transient Title	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
20	Standby to LPSI pump test to standby	500	239	228	252	839	This is the total from 73ST-9RC02 and is based on the sum of the 25% assumption (125 events) and the actual events counted after 1995. The current practice is to perform the test at 90% of the quarterly periodicity and as a post maintenance test. The actual count between 1995 and 2005 is roughly equal to the 25% assumed for the period 1985 - 1995. Unit 3 experienced 127 events between 1995 and 2005 vs. 125 assumed for the previous 10 years, but U3 did not begin operation until 1987. The 25% assumption is therefore reasonable.
21	Standby to HPSI pump test to standby	500	246	222	243	819	This is the total from 73ST-9RC02 and is based on the sum of the 25% assumption (125 events) and the actual events counted after 1995. The current practice is to perform the test at 90% of the quarterly periodicity and as a post maintenance test. The actual count between 1995 and 2005 is roughly equal to the 25% assumed for the period 1985 - 1995. The 25% assumption is therefore reasonable.
Transients With 25% Accumulation Assumption Justified by Review of Operating History 1985 - 1995							
13	Shift from normal to maximum purification flow at 100% power	1000	250	250	250	833	This total includes actual data from 1995-2005 and the 25% assumption (250 events) for 1985-1995. No events were noted in the U1, 2 and 3 log reviews. However, for conservatism the 25% assumption was retained. The assumption of 25% appears very conservative based on no events being discovered in the U1, 2 and 3 log reviews between 1985-1995 and due to the fact that no events were counted after 1995 for all three units.

Table RAI 4.3-1, Transients Retaining the 25% Assumed Transient Accumulation

1 LRA Table 4.3-3 Item No.	2 Transient Title	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
26	Low-low volume control tank/charging pump suction diversion to RWT	80	20	20	20	67	This total includes actual data from 1995-2005 and the 25% assumption (20 events) for 1985-1995. No events were noted in the U1, 2 and 3 log reviews. However, for conservatism the 25% assumption was retained. The assumption of 25% appears very conservative based on no events being discovered in the U1, 2 and 3 log reviews between 1985-1995 and due to the fact that no events were counted after 1995 for all three units.
27	Pressurizer level control, failure to full open	100	25	25	25	83	This total includes actual data from 1995-2005 and the 25% assumption (25 events) for 1985-1995. No events were noted in the U1, 2 and 3 log reviews. However, for conservatism the 25% assumption was retained. The assumption of 25% appears very conservative based on no events being discovered in the U1, 2 and 3 log reviews between 1985-1995 and due to the fact that no events were counted after 1995 for all three units.
57	Pressurization by spurious actuation of all pressurizer heaters at 100% power	10	2	2	2	7	This total includes actual data from 1995-2005 and the 25% assumption (2 events) for 1985-1995. No events were noted in the U1, 2 and 3 log reviews, or in the LER review. However, for conservatism the 25% assumption was retained. The assumption of 25% appears very conservative based on no events being discovered in the U1, 2 and 3 log reviews between 1985-1995 and due to the fact that no events were counted after 1995 for all three units.

Table RAI 4.3-1, Transients Retaining the 25% Assumed Transient Accumulation

1 LRA Table 4.3-3 Item No.	2 Transient Title	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
59	Loss of offsite and onsite ac power, with retention of onsite emergency ac and dc power at 100% power	5	1	2	2	7	The totals include one assumed event for 1985 - 1995. No events were noted in the U1, 2 and 3 log reviews, or in the LER review. However, for conservatism the 25% assumption was retained. The assumption is conservative based on no evidence of this event occurring during the period 1985-1995 and the fact that only two units accumulated 1 event after 1995.
60	Depressurization of the SIS, CSS, SCS by full opening of a safety or relief valve without reseating	5	1	1	1	3	The totals include one assumed event for 1985 - 1995. No events were noted in the U1, 2 and 3 log reviews, or in the LER review. However, for conservatism the 25% assumption was retained. The assumption is conservative based on no evidence of this event occurring during the period 1985-1995 and the fact that no events were counted after 1995 in any unit.
80	Secondary system leak test	200	50	50	50	167	This is the OSG total from 73ST-9RC02 and it is the assumption of 25% of the allowed events. The SGs have been replaced and the test has been performed once on one of the U2 replacement SGs. The secondary leak test is performed in cold shutdown with the SG primary plenum open for visual inspection to identify leaking SG tubes, and it is a major evolution requiring an extensive system lineup. The lead reviewer recounting transients was involved in steam generator activities from 1993 through 2004, and he only recalls two or three times the test was performed. After 1993 SG tube inspection activities were expanded to 100% eddy current inspections each refueling, and insitu tube pressurization tests made secondary pressure tests a rare event. The retention of the 25% assumption (50 tests) is therefore, very conservative.

Table RAI 4.3-1, Transients Retaining the 25% Assumed Transient Accumulation

1 LRA Table 4.3-3 Item No.	2 Transient Title	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
82	Standby to preoperational hydrostatic test to standby [Safety Injection]	10	2	2	2	7	This is a preoperational test evolution. Retention of the 25% assumption (2) was deemed conservative based on it being twice the normal number of events.
83	Standby to inservice hydrostatic test to standby (Safety Injection)	10	2	2	2	7	This total includes actual data from 1995-2005 and the 25% assumption (2 events) for 1985-1995. No events were noted in the U1, 2 and 3 log reviews. However, for conservatism the 25% assumption was retained. The assumption of 25% appears very conservative based on no events being discovered in the U1, 2 and 3 log reviews between 1985-1995 and due to the fact that no events were counted after 1995 for all three units.

NRC RAI 4.3-2

Issue

Table 4.3-3 in Amendment 14 of the LRA provides an adequate technical basis that PVNGS operates as a base load plant and that Transient No. 3, "5 percent per minute power ramp increase, from 15 percent to 100 percent power," and Transient No. 4, "5 percent per minute power ramp decrease, from 15 percent to 100 percent power," do not need to be counted relative to the 15,000 cycle limits for these transients. However, it appears that technical specification (TS) 5.5.5 and UFSAR Section 3.9.1.1 may still require these transients to be counted, specifically because these transients are currently listed as transients in Section I and II of UFSAR Table 3.9-1.

Section 4.3.2.1 of the LRA states that for the Unit 1 instrument nozzles, the calculated cumulative usage factor (CUF) of 0.68 is based on this 15,000 load following cycle limit. However, there is a factor of five difference in the CUF that is reported for these components for Unit 1 and those that are reported for the instrument nozzles at Units 2 and 3.

Request

1. Clarify, with justification, whether these transients are required to be counted per TS 5.5.5 and UFSAR Section 3.9.1.1. If these transients are required to be counted per TS 5.5.5 and UFSAR Section 3.9.1.1, clarify the actions that will be taken to resolve the inconsistency if it is determined there is a valid technical basis for not counting these transients.

2. Clarify whether either Transient No. 3 or Transient No. 4 has occurred at the PVNGS site to date. If either transient has occurred, clarify how this is consistent with the plant being operated as a base load plant and justify not counting these transients.

3. Clarify why there is a factor of five difference between the CUFs reported for the instrument nozzles at Unit 1 from those that are reported for the corresponding nozzles at Units 2 and 3.

APS Response to RAI 4.3-2

Response (1)

The PVNGS Technical Specification 5.5.5, *Component Cyclic or Transient Limit*, states:

> "This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits."

The Palo Verde program specified in Technical Specification 5.5.5, Component Cyclic or Transient Limit, provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits. The controls to track cyclic and transient occurrences are implemented by either counting the occurrences or by accounting for the occurrences such that components are maintained within the design limits. A Licensing Document Change Request is being developed to add this clarification to UFSAR Section 3.9.1.1.

In the case of power change transients in UFSAR Table 3.9.1-1 (Item Nos. 3 and 4 in LRA Tables 4.3-2 and 3), UFSAR Table 3.9.1-1 defines these "Power Changes" as follows:

> "15,000 power change cycles over the range of 15% to 100% of full load at a rate of 5% of full load per minute either increasing or decreasing."

The intent of Transient Nos. 3 and 4 was primarily to address the daily changes in grid demand that have been historically observed at other plants. PVNGS was designed to accommodate these types of cyclic load swings as well as the infrequent variations in power required by equipment maintenance, Technical Specifications action statements, or other operational considerations. However, for a variety of reasons PVNGS has never been operated as a load following plant, and has followed a base load strategy since initial operation in each of the three units.

Using a 90% capacity factor and 60 years of operation one can calculate that 15,000 power change cycles would require one cycle every 31.6 hours. Unless a plant operates with a load following strategy, this number is not credible since power changes for maintenance, Technical Specifications action statements and operational considerations are infrequent. The PVNGS operating strategy does not include load following. Therefore, these transients are accounted for such that components are maintained within the design limits.

<u>Response (2)</u>

There is no design feature that would prevent PVNGS from making power changes to load-follow at the request of the load dispatcher. However, a review of control room logs for the period of 1985 through 1995 to reconstruct transient history did not identify any load following power changes as defined in Response (1) above. A review was conducted of dispatch procedures, the PVNGS owner-participant agreement, and a recent operating agent filing of annual resource planning which determined that:

- In the event of a grid condition requiring power reduction the PVNGS units have priority to operate, so that fossil generation absorbs changes in consumer demand.

- As Operating Agent, APS determines the power level of each PVNGS unit in compliance with the Station license. The other six owners monitor plant performance through their attendance of the Administrative Committee Engineering and Operations Committee meetings.

- Generation planning models used by the operating agent, APS, indicate the intent to operate PVNGS as a base load plant barring unforeseen events.

The aggregate results of these reviews supports the conclusion that the PVNGS units have not experienced Transient Nos. 3 or 4 due to load following and that the intention for the foreseeable future is to continue to operate the PVNGS units as base loaded units.

The PVNGS staff maintains that power variations are rare and not a part of the operating strategy for the three PVNGS units. However, a record review was performed to determine the number of power changes that occurred in each unit during the 24-month period of 2006 through 2007 to validate this assertion. The results are as follows:

Unit	Power Increase	Power Decrease *
1 (10 Power Increases and 8 power decreases)	Post Refuel Startup (2) Post Trip Startup (1) Post Maintenance (6) Post LCO** (1)	Refueling Shutdown (1) Maintenance (6) LCO** (1)
2 (11 Power Increases and 7 power decreases)	Post Refuel Startup (1) Post Trip Startup (3) Post Cutback Increase (1) Post Maintenance (5) Post LCO** (1)	Refueling Shutdown (1) Maintenance (5) LCO** (1)
3 (9 Power Increases and 7 power decreases)	Post Refuel Startup (1) Post Trip Startup (3) Post Maintenance (5) Post LCO** (0)	Refueling Shutdown (2) Maintenance (5) LCO** (0)

* Power decreases associated with counted transients are not double counted (e.g., a reactor trip is not also counted as a power decrease).
** "LCO" is when a Technical Specification Limiting Condition for Operation requires a power decrease.

The data presented above is typical for PVNGS and results in approximately 5 cyclic power transients per plant per year. The rate required to reach 15,000 cycles in 60 years is 250 cyclic power transients per plant per year. The Technical Specification 5.5.5 transient tracking program accounts for these transients such that components are maintained within design limits because the rate of actual occurrences and projected occurrences is significantly below the limiting values.

Response (3)

The analyses for all three units include the same load following cycles (15,000 increase and 15,000 decrease). The differences are not due to differences in geometry, materials, loading, or transients, but are due to modeling and analysis methods and assumptions. These differences include:

- The Unit 1 analysis used a more-conservative treatment of vortex shedding.
- Some model differences resulting in a slightly-different limiting location.
- Arithmetic instead of vector load addition at the limiting Unit 1 location.

The vortex shedding difference produced a larger number of assumed vortex shedding load cycles for Unit 1, which was a significant factor in the difference. The stress ranges in some cases were slightly lower in the analyses for Units 2 and 3 as compared to Unit 1, and a small reduction in stress range yields a significant reduction in CUF.

NRC RAI 4.3-3

Issue

Section 4.3.5 of the LRA states that the calculated stresses in limiting locations were less than allowable in the revised design analyses for the reactor coolant hot leg sample lines piping and the steam generator (SG) downcomer and feedwater recirculation lines piping. However, LRA Section 4.3.5 does not provide sufficient information for the staff to confirm these assertions.

Request

Provide the code allowable stress limits and the stress ranges obtained in the revised design analyses for the reactor coolant hot leg sample line piping and the SG downcomer and feedwater recirculation line piping. Also, provide the American Society of Mechanical Engineer Code edition and specific subsection used for the revised design analyses for these piping components.

APS Response to RAI 4.3-3

The requested information is provided below in tabular form.

Reactor Coolant Hot Leg Sample Lines

Max. Calculated Stress per Eq. (11) (psi)	Original Max. Allowed Stress $S_h + S_A$ (psi)	Cycles	f	Revised Max Allowed Stress $S_h + 0.9\ S_A$ (psi)
39,872	43,375	8,273	0.9	40,628

Steam Generator Downcomer and Feedwater Recirculation Lines

Max. Calculated Stress Range per Eq. (10) (psi)	Existing Code Allowed Stress S_A (for f=1) (psi)	Cycles	f	Revised Allowed Stress S_A'=0.8 x S_A (psi)	Max. Calculated PBA* Stress (psi)	Existing Max. Allowed PBA* Stress (psi)	Revised Allowed PBA* Stress (psi)
14,364	23,263	15,336	0.8	18,610	20,286	32,578	28,856

*PBA means "Pipe Break Analysis"

The revised design analyses for these piping components was performed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1974 up to and including Winter 1975 Addenda. Stress range reduction factor f was obtained from Table NC-3611.2(e)-1 and the comparison of the calculated stress range vs. allowable stress limit was performed per the requirements of paragraph NC-3652.3.

NRC RAI 4.3-4

Issue

Section 4.3.4 of the LRA states that for reactor pressure vessel (RPV) shell and lower head, RPV inlet and outlet nozzles, and safety injection nozzle (forging knuckle), the maximum applicable environmental factors (F_{en}) for low alloy steel was used and was determined following NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels." However, LRA Section 4.3.4 does not provide sufficient information to confirm this statement.

Request

Demonstrate that the F_{en} factor used for assessment of the reactor coolant environment impact on the RPV shell and lower head, RPV inlet and outlet nozzles, and safety injection nozzle (forging knuckle) are the maximum applicable for a given material. Provide a basis and justification for any assumptions that were made for the parameters in the assessment, such as strain rate, dissolved oxygen, temperature and sulfur content.

APS Response to RAI 4.3-4

The "maximum applicable" F_{en}'s for the low alloy steel (LAS) locations cited above were all computed using the formulas from NUREG/CR-6583. In each case, a constant bounding F_{en} value was computed, using the following assumptions:

- Low concentration of dissolved oxygen (DO < 0.05 ppm) for times when water temperature was above 150°C (302°F)
- Most conservative value of T* (= 200 for LAS)
- Most conservative value for ε* (= ln(0.001))
- Most conservative value of S* (= 0.015 for LAS)

Where:

T* = Transformed temperature (C)
ε* = Transformed total strain rate
S*= Transformed sulphur content (wt%)

The values used in the analyses are summarized in the following table.

Location	Material	F_{en}	basis
RPV shell & lower head	Low Alloy	2.455	NUREG/CR-6583
RPV inlet/outlet nozzles	Low Alloy	2.455	NUREG/CR-6583
SI nozzle (at knuckle)	Low Alloy	2.455	NUREG/CR-6583

The dissolved oxygen (DO) value was selected based on industry experience and confirmed by plant Chemistry staff. The plant staff noted only a few instances when DO exceeded 0.05 ppm for a relatively short period of time.. These occurred following the startup of a third reactor coolant pump in hot standby after refueling, and these infrequent exceptions do not impact the validity of the assumed DO level.

NRC RAI 4.3-5

Issue

Note 7 and 9 of Table 4.3-11 of the LRA provides the reanalysis computed F_{en} values for load set pairs with a significant fatigue contribution for the charging system nozzle (safe end) and the safety injection nozzle (safe end), respectively. Section 4.3.4 of the LRA does not contain sufficient information on the assumptions that have been used for the environmental F_{en} factor calculations.

Request

1. Describe in detail the methodology that has been used for the environmental F_{en} factor calculation of the charging system nozzle and the safety injection nozzle.

2. Provide a basis for any assumptions that were made for the parameters, such as strain rate, dissolved oxygen, and temperature, in the assessment of a computed F_{en} value for the load set pairs with a significant fatigue contribution.

3. Confirm the value of the maximum F_{en} factor used for all remaining load set pairs.

APS Response to RAI 4.3-5

Response (1)

The F_{en} analyses for these locations are documented in detail in plant calculations. In the analyses, F_{en} values were determined for each load-set pair using the formulas from NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," as appropriate for stainless steel components.

Detailed F_{en} values were computed for load-set pairs that contributed more than 0.001 to the CUF for the given location. Load-set pairs that contributed less usage were conservatively assigned a F_{en} of 15.35 (for stainless steel). The total EAF usage was computed as the sum of (the CUF per load-set pair) × (the F_{en} for that pair), as illustrated in the tables below.

For each load-set pair, the detailed F_{en} was computed using the *Integrated Strain Rate* method, as described in MRP-47, Revision 1: "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." In this approach,

$F_{en,pair} = \Sigma(F_{en,i} * \Delta\varepsilon_i) / \Sigma(\Delta\varepsilon_i)$,

where:

$F_{en,i}$ = F_{en} computed at time point *i*, based on $\varepsilon_i = 100\ \Delta\varepsilon_i/\Delta t$ and parameters T* and O* determined for just that time point (see below)

$\Delta\varepsilon_i$ = change in strain at time point *i*, = $(\sigma_i - \sigma_{i-1})/E$

σ_i = stress intensity response of transient at time point *i*

Δt = change in time at point *i*, = $t_i - t_{i-1}$

E = Young's modulus (in psi) from the governing fatigue curve

The summation is over the portions of increasing tensile stress during each of the paired transients. For each time slice, T* is computed using the maximum temperature during the time slice, and O* is determined using the most conservative DO during the time slice (minimum DO is conservative for stainless steel, maximum DO is conservative for carbon steel and low-alloy steel).

Charging Nozzle:

	Load Set A		Load Set B	n	Sn, psi	Ke	Salt, psi	N	Usage	Fen	EAF
33	1_XLOL	34	2_XLOL	800	23981	1	106855	1441	0.5553429	5.144	2.8566
27	2_LOL	28	1_LtdnChgReco	300	23725	1	103739	1593	0.1883698	9.448	1.7797
28	1_LtdnChgReco	32	1_LOC	280	17780	1	73928	5590	0.0500914	13.02	0.6522
11	11_Heatup	26	1_LOL	300	12725	1	35941	198770	0.0015093	7.724	0.0117
11	11_Heatup	48	2_HydroCooldo	20	5574	1	28363	958801	0.0000209	15.35	0.0003
11	11_Heatup	14	2_Cooldown	580	6646	1	27306	1110800	0.0005221	15.35	0.0080
14	2_Cooldown	45	11_HydroHeatu	20	8249	1	26251	1263000	0.0000158	15.35	0.0002
9	9_Heatup	14	2_Cooldown	300	5589	1	26062	1293200	0.000232	15.35	0.0036
9	9_Heatup	22	1_Norm2MaxPur	600	5061	1	25339	1417400	0.0004233	15.35	0.0065
22	1_Norm2MaxPur	24	1_Norm2MaxPur	400	5937	1	25305	1423700	0.000281	15.35	0.0043
24	1_Norm2MaxPur	50	4_HydroCooldo	20	6684	1	24192	1648600	0.0000121	15.35	0.0002
16	4_Cooldown	24	1_Norm2MaxPur	580	7622	1	23140	1905800	0.0003043	15.35	0.0047
16	4_Cooldown	43	9_HydroHeatup	20	8224	1	22161	2258600	0.0000089	15.35	0.0001
5	5_Heatup	16	4_Cooldown	300	4892	1	21007	2838300	0.0001057	15.35	0.0016
5	5_Heatup	47	1_HydroCooldo	20	1970	1	20432	3195900	0.0000063	15.35	9.671E-05
5	5_Heatup	18	1_Combined	580	4204	1	19668	3761000	0.0001542	15.35	0.0024
18	1_Combined	29	2_LtdnChgReco	580	6977	1	18745	4618600	0.0001256	15.35	0.0019
18	1_Combined	39	5_HydroHeatup	20	6889	1	18646	4723900	0.0000042	15.35	6.447E-05
7	7_Heatup	18	1_Combined	900	4075	1	18187	5362600	0.0001678	15.35	0.0026
18	1_Combined	41	7_HydroHeatup	20	6538	1	17171	7583900	0.0000026	15.35	3.991E-05
3	3_Heatup	13	1_Cooldown	900	4292	1	15756	14409000	0.0000625	15.35	0.0010
18	1_Combined	37	3_HydroHeatup	20	6735	1	14291	51538000	0.0000004	15.35	6.14E-06
12	12_Heatup	18	1_Combined	900	5573	1	13579	infinite	0	15.35	0
								Total Usage =	0.797763		
										EAF =	5.338
										Weighted Average Fen =	6.691

Safety Injection Nozzle (Safe End):

	Load Set A		Load Set B	Cycles	Sn	Ke	Salt	Nallowed	Usage	Fen	EAF
4	Plant Cooldown	6	Safety Injection	200	10572	1	149854	481.71	0.41519	2.66	1.1063
3	Plant Cooldown	4	Plant Cooldown	300	12433	1	104954	1531	0.19595	2.55	0.4991
3	Plant Cooldown	7	Safety Injection	160	23758	1	94085	2242.8	0.07134	6.28	0.4478
3	Plant Cooldown	14	Safety Injection Check Valve Test	300	20453	1	78447	4439.01	0.00901	4.27	0.0385
13	Safety Injection Check Valive Test	14	Safety Injection Check Valve Test	20	15808	1	40078	110158	0.00109	9.37	0.0102
1	Plant Heatup	13	Safety Injection Check Valve Test	40	13523	1	27990	1024700	0.00004	15.35	0.0006
5	Plant Cooldown	9	Reactor Trip, Loss of Load, Loss of Flow	20	17850	1	21083	2794600	0.00017	15.35	0.0026
5	Plant Cooldown	8	Reactor Trip, Loss of Load, Loss of Flow	300	17834	1	21071	2801500	0.00001	15.35	0.0001
1	Plant Heatup	8	Reactor Trip, Loss of Load, Loss of Flow	460	17595	1	20771	2978600	0.00015	15.35	0.0024
11	Plant Loading	16	Leak Test 2250 psia, Down	200	14309	1	17194	7521900	0.00003	15.35	0.0004
11	Plant Loading	17	Hydrostatic Test	20	14556	1	17097	7782100	0.00000	15.35	0.0000
11	Plant Loading	18	Hydrostatic Test	580	14556	1	17097	7782100	0.00000	15.35	0.0000
11	Plant Loading	15	Leak Test 2250 psia, Up	20	13915	1	16672	9056600	0.00002	15.35	0.0003
								Total Usage =	0.6930		
										EAF =	2.108
										Weighted Average Fen =	3.042

Response (2)

A DO concentration < 0.05 ppm was assumed for stainless steel. This is the more conservative option and does not require justification. The DO value was selected based on industry experience and confirmed by plant chemistry staff. The plant staff noted only a few instances when DO exceeded 0.05 ppm for a relatively short period of time. These occurred following the startup of a third reactor coolant pump in hot standby after refueling, and these infrequent exceptions do not impact the validity of the assumed DO level. Both strain rate and water temperature were calculated from the design transient specifications and corresponding stress analyses. No assumptions were made for these parameters.

Response (3)

A F_{en} value of 15.35 was applied to all transient pairs that contributed less than 0.001 usage to the CUF for either of these two locations. This is the maximum possible F_{en} value for stainless steel, based on the formulas in NUREG/CR-5704.

NRC RAI 4.3-6

Background

LRA Section 4.3.4 states that a bounding F_{en} factor of 1.49 was used for the Alloy 600 component, pressurizer heater penetrations. NUREG/CR-6335, "Fatigue Strain–Life Behavior of Carbon and Low–Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," provides the statistical characterizations used to derive this F_{en} factor of 1.49 for Alloy 600, and states the fatigue S-N database (fatigue per load cycle curves) for Alloy 600 is extremely limited and does not cover an adequate range of material and loading variables that might influence fatigue life. It further states that the data was obtained from relatively few heats of material and are inadequate to establish the effect of strain rate on fatigue life in air or of temperature in a water environment. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," incorporates more recent fatigue data using a larger database for determining the F_{en} factor of nickel alloys.

Issue

The F_{en} factor of 1.49 for nickel alloys may be non-conservative. The F_{en} for nickel alloys based on NUREG/CR-6909 varies based on temperature, strain rate and dissolved oxygen. Based on actual plant operating conditions the F_{en} factor can vary from a value of 1.0 to 4.52 based on this methodology. Therefore, the CUF value for the pressurizer heater penetrations may be as high as 2.86 using the CUF presented in the LRA and the maximum F_{en} derived from NUREG/CR-6909 which would exceed the design limit of 1.0 when considering environmental effects of reactor coolant during the period of extended operation.

Request

1. Since the F_{en} for nickel alloys can vary from 1.0 to 4.52 based on NUREG/CR-6909 and the CUF value may exceed the design limit of 1.0 for the pressurizer heater penetrations when considering environmental effects, justify using a value of 1.49 for the F_{en} factor for this nickel alloy component.

2. Describe the current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component only, consistent with the methodology in NUREG/CR-6909. If there are no current or future planned actions to update the CUF calculation with F_{en} factor for the Alloy 600 component consistent with the methodology in NUREG/CR-6909, provide a justification for not performing the update.

APS Response to RAI 4.3-6

Response (1)

No later than two years prior to the period of extended operation, APS will confirm the conservatism of the F_{en} value of 1.49 using the methods specified in NUREG/CR-6909, and will use the F_{en} calculated using the NUREG/CR-6909 methods if it is more conservative than the 1.49 value. Item No. 57 has been added to the commitment list in LRA Table A4-1, as shown in LRA Amendment No. 18 in Enclosure 2, to reflect this response.

Response (2)

No later than two years prior to the period of extended operation APS will perform a reanalysis of the pressurizer heater penetrations to consider EAF effects using the formulas and methodology given in NUREG/CR-6909. Item No. 58 has been added to the commitment list in LRA Table A4-1, as shown in LRA Amendment No. 18 in Enclosure 2, to reflect this response.

ENCLOSURE 2

Palo Verde Nuclear Generating Station
License Renewal Application Amendment No. 18 – Clean Pages

LRA Section	Page Nos.	RAI No.
Table 4.3-3	4.3-27	na
4.3.5	4.3-101	na
Table A4-1 item 57	A-59	4.3-6
Table A4-1 item 58	A-59	4.3-6
A3.2	A-27 through A-37	na

Table 4.3-3, PVNGS Units 1, 2, and 3 Fatigue Cycle Count and Projections

1 Row No.	2 Transient Title (Shaded items are not counted)	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
22	Adding 40F feedwater at 875 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions	500	0*	0	0*	0	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. U2 count is actual data.
23	Adding 100F feedwater at 875 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions	500	0*	0	0*	0	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. U2 count is actual data.
24	Pressure transients of 85 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps	4000	0*	336**	0*	2238**	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. ** The U2 total is based on assuming all RCP starts and stops reported between '95 - '05 in transients 8 & 9 apply to the RSG. The sum (56) was multiplied by 6 assuming all 4 RCPs experienced a start and stop for each Mode 3 start and stop plus 2 pump start/stop cycles for sweeps. This was multiplied by 6.66 to account for the 336 being accumulated in a 10 year period versus 20 years.

4.3.5 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME III Class 2 and 3 Piping

Summary Description

None of the ANSI B31.1 or the ASME III Subsections NC and ND for Class 2 and 3 piping invokes fatigue analyses. However, piping in the scope of license renewal that is designed to these codes requires the application of a stress range reduction factor (SRRF) to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. The allowable secondary stress range is 1.0 S_A for 7000 equivalent full-range temperature cycles or less, and is reduced in steps to 0.5 S_A for greater than 100,000 cycles. Partial cycles are counted proportional to their temperature range.

These piping analyses are TLAAs because they are part of the current licensing basis, are used to support safety determinations, and depend on an assumed number of thermal cycles that can be linked to plant life.

Analysis

PVNGS Piping

The EPRI license renewal document, *"Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," [Ref 16]* includes temperature screening criteria to identify components that might be subject to significant thermal fatigue effects. Normal and upset operating temperatures less than 220 °F in carbon steel components, or 270 °F in stainless steel, will not produce significant thermal stresses, and will not therefore produce significant fatigue effects. A systematic survey of all plant piping systems found that with the exception of reactor coolant sampling lines and the steam generator downcomer and feedwater recirculation lines described in this section, the piping and components within the scope of license renewal:

- Do not meet the operating temperature screening criteria of the EPRI *Mechanical Tools,* and therefore do not experience significant thermal cycle stresses; or
- Clearly do not operate in a cycling mode that would expose the piping to more than three thermal cycles per week, i.e. to more than 7,000 cycles in 60 years; or
- The assumed thermal cycle count for the analyses depends closely on reactor operating cycles, and can therefore conservatively be approximated by the thermal cycles used in the ASME III Class 1 vessel and piping fatigue analyses.

For this last case, see the reactor coolant system thermal cycles listed inTable 4.3-2. Of these, those likely to produce full-range thermal cycles in balance-of-plant Class 2, 3, and B31.1 piping, in a 40-year plant lifetime, are the 500 heatup-cooldown cycles plus 240 reactor trips. Other events may contribute a few full-range cycles or a number of part-range cycles, but the total count of expected full-range thermal cycles is under 1000 for a 40-year plant life. This is true for in-scope balance-of-plant support systems, as well as the CVCS and ECCS piping more directly connected to the reactor system. For a 60-year life the

Table A4-1 License Renewal Commitments

Item No.	Commitment	LRA Section	Implementation Schedule
57	No later than two years prior to the period of extended operation, APS will confirm the conservatism of the F_{en} value of 1.49 using the methods specified in NUREG/CR-6909, and will use the F_{en} calculated using the NUREG/CR-6909 methods if it is more conservative than the 1.49 value. (RCTSAI 3488220)	Response to RAI RAI 4.3-6 (letter no. 102-06210, dated June 29, 2010)	No later than two years prior to the period of extended operation,[1]
58	No later than two years prior to the period of extended operation APS will perform a reanalysis of the pressurizer heater penetrations to consider EAF effects using the formulas and methodology given in NUREG/CR-6909. (RCTSAI 3488223)	Response to RAI RAI 4.3-6 (letter no. 102-06210, dated June 29, 2010)	No later than two years prior to the period of extended operation,[1]

A3.2 METAL FATIGUE ANALYSIS

This section describes:

- ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components
- ASME Section III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals
- Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)
- Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in B31.1 and ASME Section III Class 2 and 3 Piping

ASME III requires no fatigue analysis for Class 2 components. However, design of the following PVNGS Class 2 components is supported by Class 1 fatigue analyses:

- Secondary sides of the replacement steam generators
- Regenerative and letdown heat exchangers
- HPSI and LPSI pumps
- Main steam safety valves

<u>Basis of Fatigue Analysis</u>

ASME Section III Class 1 design specifications define a design basis set of static and transient load conditions. The design number of each transient specified was selected to be larger than expected to occur during the 40-year licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs. Although original design specifications commonly state that the transients are for a 40-year design life, the fatigue analyses themselves are based on the specified number of occurrences of each transient rather than on this lifetime.

Metal Fatigue of Reactor Coolant Pressure Boundary Program

The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 ensures that actual plant experience remains bounded by the assumptions used in the design calculations, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the ASME Section III limit of 1.0 for the fatigue cumulative usage factor is reached.

The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) was implemented in response to PVNGS Technical Specification 5.5.5 which requires the establishment of a "Component Cyclic or Transient Limit" program to track the occurrences specified in PVNGS UFSAR section 3.9.1.1."

A3.2.1 ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components

Fatigue analyses exist for ASME III Division 1 Class 1 piping, vessels, heat exchangers, pumps, and valves; and if applicable, their supports.

Class 1 fatigue analyses also support design of the following Class 2 components:

- Secondary sides of the replacement steam generators
- Regenerative and letdown heat exchangers
- HPSI and LPSI pumps
- Main steam safety valves

The Class 1 analyses have been updated to incorporate redefinitions of loads and design basis events, operating changes, and power uprate with steam generator replacement.

The PVNGS reactor vessel internals were analyzed to ASME Section III Subsection NG. See Subsection A3.2.2.

A3.2.1.1 Reactor Pressure Vessel, Nozzles, Head, and Studs

The PVNGS reactor pressure vessels were designed, built, and analyzed by Combustion Engineering to ASME Section III, Subsection NB (Class 1), 1971 Edition with addenda through Winter 1973. The reactor vessel primary coolant inlet and outlet nozzles and lower-head-to-shell juncture are evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

The analyses performed to incorporate the effects of power uprate (PUR) and replacement steam generators (RSG) into the current design bases demonstrated that the effects on fatigue analyses were limited to the inlet and outlet nozzles. The modification increased the CUF of the inlet nozzles and the outlet nozzles.

The 1991 CE Owner's Group review of Combustion Engineering Infobulletin 88-09, "*Nonconservative Calculation of Cumulative Fatigue Usage,*" identified a possible increase in the reactor vessel stud cumulative usage factor. The Owner's Group review found that the usage factor of reactor vessel studs at PVNGS could increase to greater than 1.0, if the more-conservative pressure curves were used. To accommodate the more-conservative pressure curves, the number of heatup-cooldown transients was reduced and the number of bolt-up transients was reduced.

The replacement reactor vessel closure heads will have been installed after more than 20 years of operation of each unit. The replacement reactor vessel closure heads were designed to ASME III, 1998 Edition up to and including the 2000 Addenda, for a 40-year operating period, and the design specification for the replacement heads includes design transients and seismic loads consistent with those for the original vessel and head. The fatigue analysis for the replacement heads and associated components therefore extends beyond the end of the period of extended operation.

The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits and will ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. In the period of extended operation the PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will monitor the environmentally assisted fatigue usage at NUREG/CR-6260 locations not monitored by cycle counting. Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached. The effects of fatigue in the reactor pressure vessel pressure boundary and its supports will thereby be managed for the period of extended operation.

A3.2.1.2 Control Element Drive Mechanism (CEDM) and Reactor Vessel Level Monitoring System (RVLMS) Pressure Housings

The CEDM and RVLMS pressure housings will have been replaced with the replacement reactor vessel closure heads after more than 20 years of operation of each unit. The replacement CEDM pressure housings and RVLMS pressure housings are designed to ASME III, Subsection NB (Class 1), 1998 Edition up to and including the 2000 Addenda, for a 40-year operating period, and the design specification for the replacement CEDM and RVLMS pressure housings included design transients and seismic loads consistent with those for the original vessel, head, and CEDM pressure housings. The CEDM pressure housing design includes a corrosion analysis for the design life.

Since the design life of the replacement CEDM and RVLMS pressure housings extend beyond the end of the period of extended operation, the respective analyses have been projected beyond the end of the period of extended operation.

A3.2.1.3 Reactor Coolant Pump Pressure Boundary Components

The CE System 80 reactor coolant pumps are designed to ASME III, 1974 Edition (no addenda) for Class 1 Vessels. The load definitions were updated for replacement steam generators (RSG) with power uprate and the code analyses were evaluated to determine the applicability of the analyses of record fatigue analyses with the new loads.

Fatigue usage factors in the reactor coolant pumps do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational and upset transient events, principally on heatup and cooldown transients. The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) tracks events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and ensure that fatigue will be adequately managed for the period of extended operation.

A3.2.1.4 Pressurizer and Pressurizer Nozzles

The PVNGS pressurizers are designed to ASME III, Subsection NB (Class 1), 1971 Edition with addenda through Winter 1973. The analyses have been updated from time to time to incorporate redefinitions of loads and design basis events, operating changes, power uprate, and modifications including; effects of NRC Bulletin 88-11 thermal stratification in the surge line, effects of Combustion Engineering Infobulletin 88-09 "*Nonconservative Calculation of Cumulative Fatigue Usage*", crack growth and fracture mechanics stability of postulated defects in heater sleeve attachment welds, thermal effects of replaced heater sleeves and their welds, and effects of nozzle weld overlays of the surge, spray, and relief nozzles and their safe ends and welds.

The pressurizer heater penetrations were screened for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260 and found to maintain an EAF <1.0 for the period of extended operation. See Section A3.2.3.

The PVNGS pressurizers have operated since startup with a continuous spray flow to prevent boron concentration stratification, and to mitigate spray line and spray nozzle fatigue.

The Linear Elastic Fracture Mechanics fatigue crack growth analysis of indications in a Unit 2 pressurizer support skirt forging weld will remain valid as long as the number of cyclic events assumed by the analysis is not exceeded. The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will be used to track events that are analyzed in non-fatigue cycle-based analyses such as this crack growth analysis, and will thereby ensure that appropriate corrective actions are completed before the design basis number of events is exceeded.

All other fatigue analyses supporting the pressurizer design either exhibit an acceptable fatigue usage factor and remain valid for the period of extended operation, or depend on an effect found to be acceptable for a limiting number of transient events. The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will ensure that the fatigue usage factors based on those transient events will remain within the code limit of 1.0 for the period of extended operation, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the cumulative usage factor exceeds the code limit of 1.0.

A3.2.1.5 Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses

The replacement steam generators (RSGs) are designed to ASME III, Subsection NB (Class 1) and NC (Class 2), 1989 Edition with no addendum. The design reports included design for a concurrent power uprate. Although the secondary side is Class 2, all pressure retaining parts of the steam generator satisfy the Class 1 criteria, including a Division 1, Section III fatigue analysis.

Although the steam generator tubes have a Class 1 fatigue analysis, the calculated usage factor is zero, and 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. Although the steam generator tube fatigue analysis is not considered a TLAA the Steam Generator Tube Integrity program (A1.8) will be used to manage steam generator tubes.

The fatigue analyses of the Unit 1 and 3 replacement steam generators are for a period sufficient to cover their installed life, and remain valid for the period of extended operation. However, PVNGS has chosen to apply aging management to all the Unit 1, 2 and 3 steam generators to achieve uniformity in aging management practices. The enhanced Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events to ensure that appropriate reevaluation or other corrective action will be initiated if an action limit is reached. Action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor exceeds the code limit of 1.0.

A3.2.1.6 ASME Section III Class 1 Valves

PVNGS Class 1 valves are designed to ASME Section III, Subsection NB, 1974 Edition with multiple addenda, the 1977 Edition with Winter 1977 addendum, and the 1989 Edition no addendum. ASME Section III requires a fatigue analysis only for Class 1 valves with inlets greater than four inches nominal. At PVNGS, specifications for some Class 1 valves with inlets four inches or less also require a fatigue analysis.

For the valve models with an NB-3545.3 normal duty operating cycle evaluation, the allowed NB-3545.3 N_A normal duty operations far exceed those expected to occur.

The calculated worst-case usage factors for the 16" Shutdown Cooling Suction Containment Isolation Valves, the 14" Safety Injection Tank Injection Discharge Isolation Gate Valves, the 14" Safety Injection Tank Injection Discharge Check Valves, the 12" HPSI/LPSI check valves, the ¾" Safety Injection Line Thermal Relief Valves, the pressurizer safety valves, the pressurizer relief valves, and the 2" isolation valves for the auxiliary spray indicate that the designs have large margins, and therefore that the pressure boundaries would withstand fatigue effects for at least 1.5 times the original design lifetimes. The calculated worst-case usage factors for the Unit 1, Class 1 Shutdown Cooling Suction Isolation Valve, and Charging Line Isolation Valves exceed 0.7. 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. The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events to ensure that appropriate reevaluation or other corrective action will be initiated if an action limit is reached. Action limits will be established to permit completion of corrective actions before the design basis number of events is exceeded. Effects of fatigue in Class 1 valve pressure boundaries will thereby be managed for the period of extended operation.

A3.2.1.7 ASME Section III Class 1 Piping and Piping Nozzles

Class 1 reactor coolant main-loop piping supplied by Combustion Engineering is designed to ASME Section III, Subsection NB, 1974 edition with addenda through Summer 1974. The main loop piping fatigue analysis was performed to the 1974 edition with addenda through Summer 1974. The fatigue analyses of piping outside the main loop used the 1974 edition with addenda through Winter 1975 or the 1977 edition with 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, steam generator replacement, and minor modifications.

See Section A3.2.1.8 for fatigue in the pressurizer surge lines.

In the primary coolant system, the most limiting calculated design basis usage factor occurs in the charging nozzle and approaches the limit of 1.0. The high usage factors are primarily due to transient thermal stresses from normal operating and upset injection events.

However, with the exception of the charging line nozzles, and possibly the pressurizer surge line discussed in Section A3.2.1.8 (if thermal stratification has not been completely mitigated); 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 Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) 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.

The charging nozzle safe ends, the safety injection nozzle forging knuckle and safe ends, and the shutdown cooling line long-radius elbow are evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

With the exception of the CVCS charging lines and nozzles and the pressurizer surge lines and nozzles; fatigue usage factors in Class 1 piping and nozzles 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 Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 counts significant transient events and thermal cycles, and tracks usage factors in a subset of Class 1 components to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.1.8 Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

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

The surge line hot leg elbow was evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

The surge lines are designed to ASME III, Subsection NB, 1977 edition with addenda through Summer 1979. The surge line design was reevaluated in 1991 through the Combustion Engineering Owners Group (CEOG) in response to the NRC Bulletin 88-11 thermal stratification concerns. The maximum calculated design basis (nominal 40-year) CUF at any location in the surge lines, including thermal stratification effects, is less than 1.0. However, when the environmental effects of reactor coolant on fatigue are considered the EAF exceeds 1.0 when the maximum Fen is applied. Therefore during the period of extended operation the surge line will be subject to stress-based fatigue monitoring under the Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1, which will ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.1.9 Class 1 Fatigue Analyses of Class 2 Regenerative and Letdown Heat Exchangers

The regenerative heat exchangers were designed and constructed to Class 2 rules on both shell and tube sides. The applicable code version date is 1974 with addenda through the Winter of 1975. The letdown heat exchangers were designed and constructed to Class 2 rules on the tube side, Class 3 on the shell side. However, although these are Class 2 and 3 heat exchangers, the specifications require a Class 1, NB-3222 fatigue analyses.

The regenerative heat exchanger fatigue analysis was performed with transients specified in the CE general specification for System 80 plants. The number of cycles for each transient event required by these specifications is consistent with or is greater than the number of cycles for each transient event that will be used as cycle counting action limits in the Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1).

The fatigue analysis for standard System 80 letdown heat exchanger was performed using the original System 80 transients. The letdown heat exchanger for PVNGS was built to Revision 4 of the CE general letdown heat exchanger specification for System 80 plants, which combined multiple transients from the previous revision of the specification. The new transients were found to bound those used in the standard System 80 letdown heat exchanger fatigue analysis. The numbers of events required by these specifications are consistent with or are greater than the number of transients that will be used as cycle counting action limits in the Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1)

A3.2.1.10 Class 1 Fatigue Analyses of Class 2 High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) Pumps for Design Thermal Cycles

The HPSI and LPSI pumps were designed to ASME III Class 2, for which the code requires no fatigue analysis. However UFSAR 3.9.3.5.3.3 describes design for a stated number of thermal transient cycles, and the Structural Integrity & Operability Analysis design reports for both the HPSI and LPSI pumps cite the Class 1 methods of ASME III Subparagraph NB-3222.4 when addressing these thermal transients.

Both the HPSI and LPSI pumps are designed for initiation of safety injection, which is classified as an upset condition. The LPSI pumps are also designed for shutdown cooling, which is a normal operating condition. The structural integrity and operability analyses for these pumps analyzed these transients and demonstrate sufficient margin for any possible increase in operating cycles above the original estimate

Although there is sufficient margin in the design of these pumps for the projected operating cycles these components are subject to aging management. The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events to ensure that appropriate corrective action will be initiated if an action limit is reached. Action limits will be established to permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor exceeds the code limit of 1.0. Cycle counting will assure that the effects of aging in the HPSI and LPSI pumps are managed for the period of extended operation

A3.2.1.11 Class 1 Analysis of Class 2 Main Steam Safety Valves

The main steam safety valves are ASME III Class 2. However UFSAR 5.2.2.4.3.2 describes a stated number of design transients, and the design includes a Class 1 fatigue analysis to Subarticle NB-3550, "Cyclic Loads for Valves".

The existing analysis demonstrates that the design is suitable for at least nine of the original 40-year design lifetimes and therefore remains valid for the period of extended operation.

A3.2.1.12 High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor

A leak-before-break analysis (LBB) eliminated large breaks in the main reactor coolant loops. Outside the main loop breaks are selected in accordance with Regulatory Guide 1.46 and Standard Review Plan Branch Technical Position MEB 3-1.

The citation of MEB 3-1 means that "intermediate breaks", between terminal ends in piping with ASME Section III Class 1 fatigue analyses are identified at any location where cumulative usage factor is equal to or greater than 0.1, with the stated exception of the reactor coolant system primary loops, to which the LBB analysis applies.

Break locations that depend on usage factor will remain valid as long as the calculated usage factors are not exceeded. The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will track events to ensure that the originally-calculated maximum usage factors are not exceeded, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits for the HELB design basis permit completion of corrective actions before the calculated design basis usage factors in Class 1 lines (outside the reactor coolant system loops) is exceeded.

A3.2.2 Fatigue and Cycle-Based TLAAs of ASME III Subsection NG Reactor Pressure Vessel Internals

The reactor vessel internals were designed and fabricated to Subsection NG rules of ASME III, 1974 Edition. The design reports indicate use of some later addenda for some parts.

The ASME Subsection NG design reports and addenda include calculated usage factors for the components. The report addenda for power uprate and steam generator replacement concluded that all code and specification requirements were satisfied.

The Subsection NG fatigue usage factors do not depend on flow-induced vibration or other high-cycle effects that are time-dependent at steady-state conditions, but depend more strongly on effects of operational, upset, and emergency transient events. Therefore, the increase in operating life to 60 years will not have a significant effect on these fatigue usage factors so long as the number of design basis transient cycles remains within the number assumed by the original analysis. The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will track events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded.

A3.2.3 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

Concerns with possible effects of elevated temperature, reactor coolant chemistry environments, and different strain rates prompted NRC-sponsored research to assess these effects, culminating in the guidance of NUREG/CR-6260, "*Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*". Although GSI 190 has been closed for plants with 40-year initial licenses, NUREG-1800 states that "The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review", noting the staff recommendation "...that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal".

NUREG/CR-6260 identifies seven sample locations for newer Combustion Engineering plants such as PVNGS:

- Reactor vessel shell and lower head
- Reactor vessel inlet nozzles
- Reactor vessel outlet nozzles
- Surge line
- Charging system nozzle
- Safety injection system nozzle
- Shutdown cooling line.

The thermal sleeves were removed from both the Loop 1 and Loop 2 safety injection nozzles, potentially increasing the CUF for the entire interior surface of the nozzle, including the knuckle location and safe end, because they were no longer protected by the thermal sleeves. Therefore two values were calculated for the safety injection nozzles, at the knuckle location and at the safe end. The safe ends were found to be limiting in the charging and safety injection nozzles.

The pressurizer heater penetrations may be subject to the effects of thermal stratification and insurge-outsurge transients, and have been subject to significant repair, modification, and reanalysis. APS has therefore elected to evaluate them with the locations listed in NUREG/CR-6260 for effects of environmentally-assisted fatigue. However, the screening evaluation determined that the EAF for the pressurizer heater penetrations is less than 1.0 when analyzed for the original number of design transients, and it was determined that the pressurizer heater penetrations need not be added to the list of NUREG/CR-6260 locations for EAF monitoring.

APS therefore evaluated a total of nine locations for effects of the reactor coolant system environment on fatigue life and selected seven for monitoring.

PVNGS performed plant-specific calculations for the NUREG/CR-6260 sample locations. The analyses used F_{en} relationships as appropriate for the material at each of the locations. F_{en} values for carbon and low-alloy steels are taken from NUREG/CR-6583. F_{en} values for stainless steels are from NUREG/CR-5704. F_{en} values for the charging nozzle safe ends and safety injection nozzle safe ends were developed using EPRI MRP-47 integrated strain rate methods and the NUREG/CR-5704 values. EAF values for the charging nozzle safe end, the pressurizer surge line elbow, and the shutdown cooling line elbow were developed using reasonable projections of transients based on analyst review of plant-specific transient data. The analyses found that the EAF usage factor in the surge line elbow, when projected to the end of a 60-year design life, may exceed 1.0. The charging inlet nozzle safe end, safety injection nozzle safe end, and shutdown cooling long radius elbow may also exceed an EAF of 1.0 if the 60 year projected cycles are exceeded.

NUREG/CR-6260 advises that conservative assumptions remain which could be removed to reduce the CUF values below the 1.0 allowable. The best method to lower the CUF for the few worst locations is fatigue monitoring, using realistic numbers of cycles, realistic severity of transients, and more refined analyses. However, in some cases, a combination of fatigue monitoring and revised analyses may be needed.

All of the NUREG/CR-6260 locations except the first, the vessel lower head to shell juncture, will be monitored for EAF in the Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 during the period of extended operation. The reactor vessel shell and lower head (juncture) will be monitored by cycle counting. The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events and usage factors to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.4 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME Section III Class 2 and 3 Piping

PVNGS ASME III Class 2 and 3 piping is designed to the 1974 edition, Summer 1975 addenda; plus later editions and addenda for certain requirements. None of ANSI B31.1 or ASME Section III Subsections NC and ND invokes fatigue analyses. However, if the number of full-range thermal cycles is expected to exceed 7,000, these codes require the application of a stress range reduction factor (SRRF) to the allowable stress range for expansion stresses (secondary stresses). The allowable secondary stress range is 1.0 S_A for 7000 equivalent full-temperature thermal cycles or less and is reduced in steps to 0.5 S_A for greater than 100,000 cycles. Partial cycles are counted proportional to their temperature range. Therefore, so long as the estimated number of cycles remains less than 7000 for a 60-year life, the stress range reduction factor remains at 1 and the stress range reduction factor used in the piping analysis will not be affected by extending the operation period to 60 years.

The survey of all plant piping systems found that the reactor coolant hot leg sample lines may be subject to more than 7000 significant thermal cycles in 60 years, requiring a reduction in SRRF to 0.9; and that the steam generator downcomer and feedwater recirculation lines may be subject to more than 15,000, requiring a reduction in SRRF to 0.8. The applicable PVNGS design analyses were revised, and found that the secondary stress ranges are within the limits imposed by these reduced SRRFs. The pipe break analysis included in the revised analysis of the steam generator downcomer and feedwater recirculation lines required no change to break locations or break types. These analyses have therefore been extended to the end of the period of extended operation.

The number of equivalent full-range thermal cycles for all other B31.1 and ASME III Class 2 and 3 lines within the scope of license renewal is expected to be only about 1500 or less in 60 years, which is only a fraction of the 7000-cycle threshold for which a stress range reduction factor is required in the applicable piping codes. The piping analyses for these remaining lines therefore require no change to the SRRF of 1.0 and remain valid for the period of extended operation.

A3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS

Aging evaluations that qualify electrical and I&C components required to meet the requirements of 10 CFR 50.49 are evaluated to demonstrate qualification for the 40 year plant life are TLAAs. The existing PVNGS Environmental Qualification program will adequately manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be

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Table 4.3-3, PVNGS Units 1, 2, and 3 Fatigue Cycle Count and Projections

1 Row No.	2 Transient Title (Shaded items are not counted)	3 Limiting Value	4 Unit 1 Accumulation as of January 2006	5 Unit 2 Accumulation as of January 2006	6 Unit 3 Accumulation as of January 2006	7 Highest Unit 60 yr Projection (Highest Unit Total X 3.33)	8 Notes
22	Adding 40F feedwater at 875 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions	500	0*	0	0*	0	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. U2 count is actual data.
23	Adding 100F feedwater at 875 gpm to the steam generator through the downcomer feedwater nozzle during loading conditions	500	0*	0	0*	0	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. U2 count is actual data.
24	Pressure transients of 85 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps	4000	0*	336 ~~224~~**	0*	2238**	* Note that per UFSAR 5.4.2.1 this is a SG transient. SGs were replaced in the fall outages of 2003, 2005 and 2007 for U2, U1 and U3 respectively resetting this event to zero. Therefore, the U1 and U3 totals are reported as zero. ** The U2 total is based on assuming all RCP starts and stops reported between '95 - '05 in transients 8 & 9 apply to the RSG. The sum (56) was multiplied by 6 assuming all 4 RCPs experienced a start and stop for each Mode 3 start and stop plus 2 pump start/stop cycles for sweeps. This was multiplied by 6.66 to account for the 336 being accumulated in a 10 year period versus 20 years.

4.3.5 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME III Class 2 and 3 Piping

Summary Description

None of the ANSI B31.1 or the ASME III Subsections NC and ND for Class 2 and 3 piping invokes fatigue analyses. However, piping in the scope of license renewal that is designed to these codes requires the application of a stress range reduction factor (SRRF) to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. The allowable secondary stress range is 1.0 S_A for 7000 equivalent full-range temperature cycles or less, and is reduced in steps to 0.5 S_A for greater than 100,000 cycles. Partial cycles are counted proportional to their temperature range.

These piping analyses are TLAAs because they are part of the current licensing basis, are used to support safety determinations, and depend on an assumed number of thermal cycles that can be linked to plant life.

Analysis

PVNGS Piping

The EPRI license renewal document, *"Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," [Ref. 16]* ~~**Error! Reference source not found.**~~ ~~[Ref. **Error! Reference source not found.**16]~~ includes temperature screening criteria to identify components that might be subject to significant thermal fatigue effects. Normal and upset operating temperatures less than 220 °F in carbon steel components, or 270 °F in stainless steel, will not produce significant thermal stresses, and will not therefore produce significant fatigue effects. A systematic survey of all plant piping systems found that with the exception of reactor coolant sampling lines and the steam generator downcomer and feedwater recirculation lines described in this section, the piping and components within the scope of license renewal:

- Do not meet the operating temperature screening criteria of the EPRI *Mechanical Tools,* and therefore do not experience significant thermal cycle stresses; or
- Clearly do not operate in a cycling mode that would expose the piping to more than three thermal cycles per week, i.e. to more than 7,000 cycles in 60 years; or
- The assumed thermal cycle count for the analyses depends closely on reactor operating cycles, and can therefore conservatively be approximated by the thermal cycles used in the ASME III Class 1 vessel and piping fatigue analyses.

For this last case, see the reactor coolant system thermal cycles listed inTable 4.3-2. Of these, those likely to produce full-range thermal cycles in balance-of-plant Class 2, 3, and B31.1 piping, in a 40-year plant lifetime, are the 500 heatup-cooldown cycles plus 240 reactor trips. Other events may contribute a few full-range cycles or a number of part-range cycles, but the total count of expected full-range thermal cycles is under 1000 for a 40-year plant life. This is true for in-scope balance-of-plant support systems, as well as the CVCS

Table A4-1 License Renewal Commitments

Item No.	Commitment	LRA Section	Implementation Schedule
57	No later than two years prior to the period of extended operation, APS will confirm the conservatism of the F_{en} value of 1.49 using the methods specified in NUREG/CR-6909, and will use the F_{en} calculated using the NUREG/CR-6909 methods if it is more conservative than the 1.49 value. (RCTSAI 3488220)	Response to RAI RAI 4.3-6 (letter no. 102-06210, dated June 29, 2010)	No later than two years prior to the period of extended operation.[1]
58	No later than two years prior to the period of extended operation APS will perform a reanalysis of the pressurizer heater penetrations to consider EAF effects using the formulas and methodology given in NUREG/CR-6909. (RCTSAI 3488223)	Response to RAI RAI 4.3-6 (letter no. 102-06210, dated June 29, 2010)	No later than two years prior to the period of extended operation.[1]

A3.2 METAL FATIGUE ANALYSIS

This section describes:

- ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components
- ASME Section III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals
- Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)
- Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in B31.1 and ASME Section III Class 2 and 3 Piping

ASME III requires no fatigue analysis for Class 2 components. However, design of the following PVNGS Class 2 components is supported by Class 1 fatigue analyses:

- Secondary sides of the replacement steam generators
- Regenerative and letdown heat exchangers
- HPSI and LPSI pumps
- Main steam safety valves

<u>Basis of Fatigue Analysis</u>

ASME Section III Class 1 design specifications define a design basis set of static and transient load conditions. The design number of each transient specified was selected to be larger than expected to occur during the 40-year licensed life of the plant, based on operating experience, and on projections of future operation based on innovations in the system designs. Although original design specifications commonly state that the transients are for a 40-year design life, the fatigue analyses themselves are based on the specified number of occurrences of each transient rather than on this lifetime.

<u>Metal</u> Fatigue ~~Management~~<u>of Reactor Coolant Pressure Boundary</u> Program

The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 ensures that actual plant experience remains bounded by the assumptions used in the design calculations, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the ASME Section III limit of 1.0 for the fatigue cumulative usage factor is reached.

~~The PVNGS fatigue management program was implemented in response to industry experience that indicated that the design basis set of transients used for Class 1 analyses of the reactor coolant pressure boundary did not include some significant transients, and therefore might not be limiting for components affected by them.~~

The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) was implemented in response to PVNGS Technical Specification 5.5.5 which requires the establishment of a "Component Cyclic or Transient Limit" program to track the occurrences specified in PVNGS UFSAR section 3.9.1.1."

A3.2.1 ASME Section III Class 1 Fatigue Analysis of Vessels, Piping, and Components

Fatigue analyses exist for ASME III Division 1 Class 1 piping, vessels, heat exchangers, pumps, and valves; and if applicable, their supports.

Class 1 fatigue analyses also support design of the following Class 2 components:

- Secondary sides of the replacement steam generators
- Regenerative and letdown heat exchangers
- HPSI and LPSI pumps
- Main steam safety valves

The Class 1 analyses have been updated to incorporate redefinitions of loads and design basis events, operating changes, and power uprate with steam generator replacement.

The PVNGS reactor vessel internals were analyzed to ASME Section III Subsection NG. See Subsection A3.2.2.

A3.2.1.1 Reactor Pressure Vessel, Nozzles, Head, and Studs

The PVNGS reactor pressure vessels were designed, built, and analyzed by Combustion Engineering to ASME Section III, Subsection NB (Class 1), 1971 Edition with addenda through Winter 1973. The reactor vessel primary coolant inlet and outlet nozzles and lower-head-to-shell juncture are evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

The analyses performed to incorporate the effects of power uprate (PUR) and replacement steam generators (RSG) into the current design bases demonstrated that the effects on fatigue analyses were limited to the inlet and outlet nozzles. The modification increased the CUF of the inlet nozzles and the outlet nozzles.

The 1991 CE Owner's Group review of Combustion Engineering Infobulletin 88-09, "*Nonconservative Calculation of Cumulative Fatigue Usage*" identified a possible increase in the reactor vessel stud cumulative usage factor. The Owner's Group review found that the usage factor of reactor vessel studs at PVNGS could increase to greater than 1.0, if the more-conservative pressure curves were used. To accommodate the more-conservative pressure curves, the number of heatup-cooldown transients was reduced and the number of bolt-up transients was reduced.

~~The segment of the Unit 2 head vent line with wall thickness reduced by the removal of indications will be replaced, and its fatigue analysis will be revised. The repair and the revised fatigue analysis will demonstrate an adequate fatigue life, projected to the end of the period of extended operation.~~

~~The PVNGS fatigue management program will track events to~~
The replacement reactor vessel closure heads will have been installed after more than 20 years of operation of each unit. The replacement reactor vessel closure heads were designed to ASME III, 1998 Edition up to and including the 2000 Addenda, for a 40-year operating period, and the design specification for the replacement heads includes design transients and seismic loads consistent with those for the original vessel and head. The fatigue analysis for the replacement heads and associated components therefore extends beyond the end of the period of extended operation.

The PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits and will ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. ~~Action~~ In the period of extended operation the PVNGS Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will monitor the environmentally assisted fatigue usage at NUREG/CR-6260 locations not monitored by cycle counting. Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached. ~~The reactor vessel studs will be tracked by the cycle-based fatigue method.~~ The effects of fatigue in the reactor pressure vessel pressure boundary and its supports will thereby be managed for the period of extended operation.

A3.2.1.2 Control Element Drive Mechanism (CEDM) ~~Nozzle Pressure Housings~~ and Reactor Vessel Level Monitoring System (RVLMS) Pressure Housings

The ~~PVNGS~~ CEDM and RVLMS ~~nozzle~~pressure housings will have been replaced with the replacement reactor vessel closure heads, after more than 20 years of operation of each unit. The replacement CEDM pressure housings and RVLMS pressure housings are designed to ASME III, Subsection NB (Class 1), ~~1974~~1998 Edition ~~with addenda through Winter 1974. The reactor vessel design reports include the structural analysis of~~ up to and including the ~~CEDM nozzle pressure housings. The analysis was re-examined~~2000 Addenda, for ~~the power uprate~~a 40-year operating period, and ~~steam generator~~the design specification for the replacement ~~modifications.~~

~~The maximum calculated usage factor in the~~ CEDM and RVLMS pressure housings ~~indicates that the~~ included design ~~has significant margin to~~ transients and seismic loads consistent with those for the original vessel, head, and CEDM pressure housings. The CEDM pressure housing design includes a corrosion analysis for the ~~limit of 1.0 and therefore remains valid for the~~ design life.

Since the design life of the replacement CEDM and RVLMS pressure housings extend beyond the end of the period of extended operation, the respective analyses have been projected beyond the end of the period of extended operation.

A3.2.1.3 Reactor Coolant Pump Pressure Boundary Components

The CE System 80 reactor coolant pumps are designed to ASME III, 1974 Edition (no addenda) for Class 1 Vessels. The load definitions were updated for replacement steam generators (RSG) with power uprate and the code analyses were evaluated to determine the applicability of the analyses of record fatigue analyses with the new loads.

Fatigue usage factors in the reactor coolant pumps do not depend on effects that are time-dependent at steady-state conditions, but depend only on effects of operational and upset transient events, principally on heatup and cooldown transients. The PVNGS ~~fatigue management~~Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) tracks events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and ensure that fatigue will be adequately managed for the period of extended operation.

A3.2.1.4 Pressurizer and Pressurizer Nozzles

The PVNGS pressurizers are designed to ASME III, Subsection NB (Class 1), 1971 Edition with addenda through Winter 1973. The analyses have been updated from time to time to incorporate redefinitions of loads and design basis events, operating changes, power uprate, and modifications including; effects of NRC Bulletin 88-11 thermal stratification in the surge line, effects of Combustion Engineering Infobulletin 88-09 "*Nonconservative Calculation of Cumulative Fatigue Usage*", crack growth and fracture mechanics stability of postulated defects in heater sleeve attachment welds, thermal effects of replaced heater sleeves and their welds, and effects of nozzle weld overlays of the surge, spray, and relief nozzles and their safe ends and welds.

The pressurizer heater penetrations ~~are evaluated~~were screened for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with

NUREG/CR-6260~~.~~ and found to maintain an EAF <1.0 for the period of extended operation. See Section A3.2.3.

The PVNGS pressurizers have operated since startup with a continuous spray flow to prevent boron concentration stratification, and to mitigate spray line and spray nozzle fatigue.

The ~~Liquid~~Linear Elastic Fracture Mechanics fatigue crack growth analysis of indications in a Unit 2 pressurizer support skirt forging weld will remain valid as long as the number of cyclic events assumed by the analysis is not exceeded. The PVNGS ~~fatigue management~~Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will be used to track ~~these~~ events ~~to~~that are analyzed in non-fatigue cycle-based analyses such as this crack growth analysis, and will thereby ensure that appropriate corrective actions are completed before the design basis number of events is exceeded.

All other fatigue analyses supporting the pressurizer design either exhibit an acceptable fatigue usage factor and remain valid for the period of extended operation, or depend on an effect found to be acceptable for a limiting number of transient events. The PVNGS ~~fatigue management~~Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will ensure that the fatigue usage factors based on those transient events will remain within the code limit of 1.0 for the period of extended operation, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the cumulative usage factor exceeds the code limit of 1.0.

A3.2.1.5 Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses

The replacement steam generators (RSGs) are designed to ASME III, Subsection NB (Class 1) and NC (Class 2), 1989 Edition with no addendum. The design reports included design for a concurrent power uprate. Although the secondary side is Class 2, all pressure retaining parts of the steam generator satisfy the Class 1 criteria, including a Division 1, Section III fatigue analysis.

Although the steam generator tubes have a Class 1 fatigue analysis, the calculated usage factor is zero, and 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. ~~The code~~ Although the steam generator tube fatigue analysis ~~of the tubes~~ is ~~therefore~~ not considered a TLAA the Steam Generator Tube Integrity program (A1.8) will be used to manage steam generator tubes.

The fatigue analyses of the Unit 1 and 3 replacement steam generators are for a period sufficient to cover their installed life, and remain valid for the period of extended operation.

~~The fatigue analyses of~~ However, PVNGS has chosen to apply aging management to all the Unit~~-~~ 1, 2 ~~replacement~~and 3 steam generators ~~are for a period sufficient to cover all but~~

~~about two years of their expected 42-year installed life, including the period of extended operation. The~~to achieve uniformity in aging management practices. The enhanced Metal Fatigue of Reactor Coolant Pressure Boundary program ~~described in Section~~ (A2.1) will track events to ensure that appropriate reevaluation or other corrective action ~~is~~will be initiated if an action limit is reached. Action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the ~~ASME code~~ cumulative ~~fatigue~~ usage factor exceeds the code limit of 1.0 ~~is reached~~.

A3.2.1.6 ASME Section III Class 1 Valves

PVNGS Class 1 valves are designed to ASME Section III, Subsection NB, 1974 Edition with multiple addenda, the 1977 Edition with Winter 1977 addendum, and the 1989 Edition no addendum. ASME Section III requires a fatigue analysis only for Class 1 valves with inlets greater than four inches nominal. At PVNGS, specifications for some Class 1 valves with inlets four inches or less also require a fatigue analysis.

For the valve models with an NB-3545.3 normal duty operating cycle evaluation, the allowed NB-3545.3 N_A normal duty operations far exceed those expected to occur.

The calculated worst-case usage factors for the 16" Shutdown Cooling Suction Containment Isolation Valves, the 14" Safety Injection Tank Injection Discharge Isolation Gate Valves, the 14" Safety Injection Tank Injection Discharge Check Valves, the 12" HPSI/LPSI check valves, the ¾" Safety Injection Line Thermal Relief Valves, the pressurizer safety valves, the pressurizer relief valves, and the 2" isolation valves for the auxiliary spray indicate that the designs have large margins, and therefore that the pressure boundaries would withstand fatigue effects for at least 1.5 times the original design lifetimes. ~~The design of these valves for fatigue effects is therefore valid for the period of extended operation.~~

The calculated worst-case usage factors for the Unit 1, Class 1 Shutdown Cooling Suction Isolation Valve, and Charging Line Isolation Valves exceed 0.7. 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. ~~The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 tracks events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded. The charging line isolation valves are subject to similar but less-severe cyclic events than the charging nozzles, whose fatigue usage is tracked by the stress-based method. The shutdown cooling suction isolation valve is the limiting location on the shutdown cooling line which will be tracked by the cycle-based fatigue method.~~ The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events to ensure that appropriate reevaluation or other corrective action will be initiated if an action limit is reached. Action limits will be established to permit completion of corrective actions before the design basis number of events is exceeded. Effects of fatigue in Class 1 valve pressure boundaries will thereby be managed for the period of extended operation.

A3.2.1.7 ASME Section III Class 1 Piping and Piping Nozzles

Class 1 reactor coolant main-loop piping supplied by Combustion Engineering is designed to ASME Section III, Subsection NB, 1974 edition with addenda through Summer 1974. The main loop piping fatigue analysis was performed to the 1974 edition with addenda through Summer 1974. The fatigue analyses of piping outside the main loop used the 1974 edition with addenda through Winter 1975 or the 1977 edition with 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, steam generator replacement, and minor modifications.

See Section A3.2.1.8 for fatigue in the pressurizer surge lines.

~~The CVCS~~ In the primary coolant system, the most limiting calculated design basis usage factor occurs in the charging nozzle and approaches the limit of 1.0. The high usage factors are primarily due to transient thermal stresses from normal operating and upset injection events.

However, with the exception of the charging line nozzles, and possibly the pressurizer surge line ~~hot leg nozzle, and the surge line elbows are the limiting~~ discussed in Section A3.2.1.8 (if thermal stratification has not been completely mitigated); fatigue usage factors in these components ~~for fatigue in the Class 1 charging lines and surge line. These~~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 Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track these events, the design basis fatigue usage factor limit (1.0) will not be exceeded in these locations ~~are subject to stress-based fatigue monitoring under the PVNGS fatigue management program~~without an appropriate evaluation and any necessary mitigating actions.

The charging nozzle safe ends, the safety injection nozzle forging knuckle and safe ends, and the shutdown cooling line long-radius elbow are evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

With the exception of the CVCS charging lines and nozzles and the pressurizer surge lines and nozzles; fatigue usage factors in Class 1 piping and nozzles 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 Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 counts significant transient events and thermal cycles, and tracks usage factors in ~~the bounding set~~a subset of ~~sample locations~~Class 1 components to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.1.8 Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

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

The surge line hot leg elbow ~~is~~was evaluated for effects of the reactor coolant environment on fatigue behavior of these materials, consistent with NUREG/CR-6260. See Section A3.2.3.

The surge lines are designed to ASME III, Subsection NB, 1977 edition with addenda through Summer 1979. The surge line design was reevaluated in 1991 through the Combustion Engineering Owners Group (CEOG) in response to the NRC Bulletin 88-11 thermal stratification concerns. The maximum calculated design basis (nominal 40-year) CUF at any location in the surge lines, including thermal stratification effects, is less than 1.0. ~~The surge line is~~However, when the environmental effects of reactor coolant on fatigue are considered the EAF exceeds 1.0 when the maximum Fen is applied. Therefore during the period of extended operation the surge line will be subject to stress-based fatigue monitoring under the Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1, which will ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.1.9 Class 1 Fatigue Analyses of Class 2 Regenerative and Letdown Heat Exchangers

The regenerative heat exchangers were designed and constructed to Class 2 rules on both shell and tube sides. The applicable code version date is 1974 with addenda through the Winter of 1975. The letdown heat exchangers were designed and constructed to Class 2 rules on the tube side, Class 3 on the shell side. However, although these are Class 2 and 3 heat exchangers, the specifications require a Class 1, NB-3222 fatigue analyses.

The regenerative ~~and letdown~~ heat exchanger fatigue ~~analyses were~~ analysis was performed with transients specified in the ~~original~~ CE general specification for System 80 plants. The number~~s~~ of cycles for each transient event~~s~~ required by these specifications is~~are~~ consistent with or is~~are~~ greater than the number~~s~~ of cycles for each transient event~~s~~ that will be used as cycle counting action limits in the Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1)~~used in the PVNGS fatigue management program~~.

The fatigue analysis for standard System 80 letdown heat exchanger was performed using the original System 80 transients. The letdown heat exchanger for PVNGS was built to Revision 4 of the CE general letdown heat exchanger specification for System 80 plants, which combined multiple transients from the previous revision of the specification. The new transients were found to bound those used in the standard System 80 letdown heat exchanger fatigue analysis. The numbers of ~~transient~~ events ~~used in the PVNGS fatigue management program.~~required by these specifications are consistent with or are greater than the number of transients that will be used as cycle counting action limits in the Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1)

~~Fatigue in the regenerative and letdown heat exchangers was originally determined to be bounded by the fatigue of the charging nozzle. Fatigue usage in the charging nozzles is affected by the same transients that have significant effects on fatigue in these heat exchangers. The charging nozzles are monitored by stress-based fatigue monitoring under the Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1. The combination of cycle counting and stress-based fatigue monitoring of the charging nozzles will assure that the effects of aging in the regenerative and letdown heat exchangers are managed for the period of extended operation. The program will track events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached.~~

A3.2.1.10 Class 1 Fatigue Analyses of Class 2 ~~HPSI and LPSI~~High Pressure Safety Injection ~~Safeguard~~(HPSI) and Low Pressure Safety Injection (LPSI) Pumps for Design Thermal Cycles

The HPSI and LPSI ~~safety injection safeguard~~ pumps were designed to ASME III Class 2, for which the code requires no fatigue analysis. However UFSAR 3.9.3.5.3.3 describes design for a stated number of thermal transient cycles, and the Structural Integrity & Operability Analysis design reports for both the HPSI and LPSI pumps cite the Class 1 methods of ASME III Subparagraph NB-3222.4 when addressing these thermal transients.

Both the HPSI and LPSI pumps are designed for initiation of safety injection, which is classified as an upset condition. The LPSI pumps are also designed for shutdown cooling, which is a normal operating condition. The structural integrity and operability analyses for these pumps analyzed these transients and demonstrate sufficient margin for any possible increase in operating cycles above the original estimate~~. The design of the HPSI and LPSI pumps is therefore valid for the period of extended operation.~~

Although there is sufficient margin in the design of these pumps for the projected operating cycles these components are subject to aging management. The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events to ensure that appropriate corrective action will be initiated if an action limit is reached. Action limits will be established to permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor exceeds the code limit of 1.0. Cycle counting will assure that the effects of aging in the HPSI and LPSI pumps are managed for the period of extended operation

A3.2.1.11 Class 1 Analysis of Class 2 Main Steam Safety Valves

The main steam safety valves are ASME III Class 2. However UFSAR 5.2.2.4.3.2 describes a stated number of design transients, and the design includes a Class 1 fatigue analysis to Subarticle NB-3550, "Cyclic Loads for Valves".

The existing analysis demonstrates that the design is suitable for at least nine of the original 40-year design lifetimes and therefore remains valid for the period of extended operation.

A3.2.1.12 High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor

A leak-before-break analysis (LBB) eliminated large breaks in the main reactor coolant loops. Outside the main loop breaks are selected in accordance with Regulatory Guide 1.46 and Standard Review Plan Branch Technical Position MEB 3-1.

The citation of MEB 3-1 means that "intermediate breaks", between terminal ends in piping with ASME Section III Class 1 fatigue analyses are identified at any location where cumulative usage factor is equal to or greater than 0.1, with the stated exception of the reactor coolant system primary loops, to which the LBB analysis applies.

Break locations that depend on usage factor will remain valid as long as the calculated usage factors are not exceeded. The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will track events to ensure that the originally-calculated maximum usage factors are not exceeded, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits for the HELB design basis permit completion of corrective actions before the calculated design basis usage factors in Class 1 lines (outside the reactor coolant system loops) is exceeded.

A3.2.2 Fatigue and Cycle-Based TLAAs of ASME III Subsection NG Reactor Pressure Vessel Internals

The reactor vessel internals were designed and fabricated to Subsection NG rules of ASME III, 1974 Edition. The design reports indicate use of some later addenda for some parts.

The ASME Subsection NG design reports and addenda include calculated usage factors for the components. The report addenda for power uprate and steam generator replacement concluded that all code and specification requirements were satisfied.

The Subsection NG fatigue usage factors do not depend on flow-induced vibration or other high-cycle effects that are time-dependent at steady-state conditions, but depend more strongly on effects of operational, upset, and emergency transient events. Therefore, the increase in operating life to 60 years will not have a significant effect on these fatigue usage factors so long as the number of design basis transient cycles remains within the number assumed by the original analysis. The Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1 will track events to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded.

A3.2.3 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

Concerns with possible effects of elevated temperature, reactor coolant chemistry environments, and different strain rates prompted NRC-sponsored research to assess these effects, culminating in the guidance of NUREG/CR-6260, "*Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*". Although GSI 190 has been closed for plants with 40-year initial licenses, NUREG-1800 states that "The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review", noting the staff recommendation "...that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal".

NUREG/CR-6260 identifies seven sample locations for newer Combustion Engineering plants such as PVNGS:

- Reactor vessel shell and lower head
- Reactor vessel inlet nozzles
- Reactor vessel outlet nozzles
- Surge line
- Charging system nozzle
- Safety injection system nozzle
- Shutdown cooling line.

The thermal sleeves were removed from both the Loop 1 and Loop 2 safety injection nozzles, potentially increasing the CUF for the entire interior surface of the nozzle, including the knuckle location and safe end, because they were no longer protected by the thermal sleeves. Therefore two values were calculated for the safety injection nozzles, at the knuckle location and at the safe end. The safe ends were found to be limiting in the charging and safety injection nozzles.

The pressurizer heater penetrations may be subject to the effects of thermal stratification and insurge-outsurge transients, and have been subject to significant repair, modification, and reanalysis. ~~Accumulation of fatigue usage in them is therefore of concern for the period of extended operation~~. APS has therefore elected to ~~include~~evaluate them ~~in~~with the locations ~~monitored~~listed in NUREG/CR-6260 for effects of environmentally-assisted fatigue. However, the screening evaluation determined that the EAF for the pressurizer heater penetrations is less than 1.0 when analyzed for the original number of design transients, and it was determined that the pressurizer heater penetrations need not be added to the list of NUREG/CR-6260 locations for EAF monitoring.

APS therefore evaluated a total of nine locations for effects of the reactor coolant system environment on fatigue life and selected seven for monitoring.

PVNGS performed plant-specific calculations for the NUREG/CR-6260 sample locations. The analyses used F_{en} relationships as appropriate for the material at each of the locations. F_{en} values for carbon and low-alloy steels are taken from NUREG/CR-6583. F_{en} values for stainless steels are from NUREG/CR-5704. F_{en} values for the charging nozzle safe ends and safety injection nozzle safe ends were developed using EPRI MRP-47 integrated strain rate methods and the NUREG/CR-5704 values. ~~F_{en}~~ EAF values for the charging nozzle safe end, the pressurizer surge line elbow, and the shutdown cooling line elbow were developed using reasonable projections of transients based on analyst review of plant-specific transient data. The analyses found that the EAF usage factor~~s~~ in ~~two of the NUREG/CR-6260 locations~~the surge line elbow, when projected to the end of a 60-year design life, may exceed 1.0. The charging inlet nozzle safe end, safety injection nozzle safe end and shutdown cooling long radius elbow may also exceed an EAF of 1.0 if the 60 year projected cycles are exceeded.

NUREG/CR-6260 advises that conservative assumptions remain which could be removed to reduce the CUF values below the 1.0 allowable. The best method to lower the CUF for the few worst locations is fatigue monitoring, using realistic numbers of cycles, realistic severity of transients, and more refined analyses. However, in some cases, a combination of fatigue monitoring and revised analyses may be needed.

All of the NUREG/CR-6260 locations except the first, the vessel lower head to shell juncture, ~~are included~~will be monitored for EAF in the Metal Fatigue of Reactor Coolant Pressure Boundary program described in Section A2.1~~.~~ during the period of extended operation. The ~~first location is not monitored because the low projected usage factor, when multiplied~~reactor vessel shell and lower head (juncture) will be monitored by ~~the applicable F_{en}, remains negligible. For the remaining locations the~~cycle counting. The Metal Fatigue of Reactor Coolant Pressure Boundary program (A2.1) will track events and usage factors to ensure that appropriate reevaluation or other corrective action is initiated if an action limit is reached. Action limits permit completion of corrective actions before the design basis number of events is exceeded, and before the ASME code cumulative fatigue usage limit of 1.0 is reached.

A3.2.4 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME Section III Class 2 and 3 Piping

PVNGS ASME III Class 2 and 3 piping is designed to the 1974 edition, Summer 1975 addenda; plus later editions and addenda for certain requirements. None of ANSI B31.1 or ASME Section III Subsections NC and ND invokes fatigue analyses. However, if the number of full-range thermal cycles is expected to exceed 7,000, these codes require the application of a stress range reduction factor (SRRF) to the allowable stress range for expansion stresses (secondary stresses). The allowable secondary stress range is 1.0 S_A for 7000 equivalent full-temperature thermal cycles or less and is reduced in steps to 0.5 S_A for greater than 100,000 cycles. Partial cycles are counted proportional to their temperature range. Therefore, so long as the estimated number of cycles remains less than 7000 for a 60-year life, the stress range reduction factor remains at 1 and the stress range reduction factor used in the piping analysis will not be affected by extending the operation period to 60 years.

The survey of all plant piping systems found that the reactor coolant hot leg sample lines may be subject to more than 7000 significant thermal cycles in 60 years, requiring a reduction in SRRF to 0.9; and that the steam generator downcomer and feedwater recirculation lines may be subject to more than 15,000, requiring a reduction in SRRF to 0.8. The applicable PVNGS design analyses were revised, and found that the secondary stress ranges are within the limits imposed by these reduced SRRFs. The pipe break analysis included in the revised analysis of the steam generator downcomer and feedwater recirculation lines required no change to break locations or break types. These analyses have therefore been extended to the end of the period of extended operation.

The number of equivalent full-range thermal cycles for all other B31.1 and ASME III Class 2 and 3 lines within the scope of license renewal is expected to be only about 1500 or less in 60 years, which is only a fraction of the 7000-cycle threshold for which a stress range reduction factor is required in the applicable piping codes. The piping analyses for these remaining lines therefore require no change to the SRRF of 1.0 and remain valid for the period of extended operation.

A3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS

Aging evaluations that qualify electrical and I&C components required to meet the requirements of 10 CFR 50.49 are evaluated to demonstrate qualification for the 40 year plant life are TLAAs. The existing PVNGS Environmental Qualification program will adequately manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished or replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

Continuing the existing 10 CFR 50.49 EQ program ensures that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. The Environmental Qualification of Electrical Components program is described in Section A2.2.