APS

Palo Verde Nuclear
Generating Station

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Reference: 1. Letter No. 102-04641-CDM/RAB, Dated December 21, 2001, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations"

2. Letter, Dated June 14, 2002 from J. N. Donohew, USNRC, to G. R. Overbeck, "Palo Verde Nuclear Generating Station, Unit 2 – Request For Additional Information Regarding Power Uprate License Amendment Request (TAC No. MB3696)"

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS) Unit 2, Docket No. STN 50-529 Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request

In Reference 1, Arizona Public Service Company (APS) submitted a license amendment request to support steam generator replacement and uprated power operations for PVNGS Unit 2. In Reference 2, the NRC provided requests for additional information from the Mechanical and Civil Engineering Branch, the Reactor Systems Branch, the Materials and Chemical Engineering Branch, the Plant Systems Branch and the Probabilistic Safety Assessment Branch.

Attachment 2 to this letter provides written responses to the questions from the Mechanical and Civil Engineering Branch. Responses to questions from the remaining branches will be submitted separately.

Responses to questions 1, 1a, 1b, 2, 4, 5 and 7 in Attachment 2 refer to Enclosure 1, which is considered proprietary information by the Westinghouse Electric Corporation.

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Westinghouse requests that Enclosure 1 be withheld from public disclosure in accordance with 10CFR 2.790. Enclosure 2 contains an affidavit from the Westinghouse Electric Corporation stating the reasons that Enclosure 1 should be considered as a proprietary document.

No commitments are being made to the NRC in this letter.

Should you have any questions, please contact Thomas N. Weber at 623-393-5764.

Sincerely,

Paul F Crawley
for CDM

CDM/TNW/RAB/kg

Attachments:

1. Notarized Affidavit
2. Plant Systems Branch Questions and APS Responses

Enclosures:

1. Excerpts from the Impact of RSG and PUR on the Structural Integrity of the RCS Components and Supports
2. Affidavit from Westinghouse Electric Corporation Submitted Pursuant 10 CFR 2.790 to Consider Enclosure 1 as a Proprietary Document

cc:	E. W. Merschoff	(NRC Region IV)
	J. N. Donohew	(NRC Project Manager)
	D. G. Naujock	(NRC Project Manager)
	N. L. Salgado	(PVNGS)
	A. V. Godwin	(ARRA)

Attachment 1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Paul F. Crawley, represent that I am Director, Nuclear Fuel Management, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

Paul F. Crawley

Paul F. Crawley

Sworn To Before Me This 10 Day Of September, 2002.

Nora E. Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

Notary Commission Stamp

I, Ian C. Rickard, depose and say that I am the Licensing Project Manager of Westinghouse Electric Company LLC (WEC), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below. I have personal knowledge of the criteria and procedures utilized by WEC in designating information as a trade secret, privileged, or as confidential commercial or financial information.

This affidavit is submitted in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding proprietary information and in conjunction with the application of Arizona Public Service Company for withholding this information. The information for which proprietary treatment is sought is contained in the following document that has been appropriately designated as proprietary:

- Enclosure 1 to WEC letter LTR-OA-02-99, "Response to Questions 1, 1a, 1b, 2, 4, 5, and 7 of the Request for Additional Information Regarding the PVNGS-2 Power Uprate (Proprietary and Non-Proprietary Versions)," July 29, 2002

Pursuant to 10 CFR 2.790(b)(4) of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information included in the document identified above should be withheld from public disclosure.

1. The information sought to be withheld from public disclosure is owned and has been held in confidence by WEC. It consists of data regarding component loads, stresses, stress intensities, cumulative usage factors, and related design information and analytical models provided in responses to NRC Requests for Additional Information (RAIs) concerning the Palo Verde Nuclear Generating Station Unit 2 (PVNGS-2) power uprate program.
2. The information consists of analyses or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to WEC.
3. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements that provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of WEC because:
    a. A similar product or service is provided by major competitors of WEC.
    b. WEC has invested substantial funds and engineering resources in the development of this information. A competitor would incur similar expense in generating equivalent information.

c. The information consists of data regarding component loads, stresses, stress intensities, cumulative usage factors, and related design information and analytical models provided in responses to NRC RAIs concerning the PVNGS-2 power uprate program. The availability of such information to competitors would enable them to design their product or service to better compete with WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.

d. Significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included in pricing WEC's products and services. The ability of WEC's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

e. Use of the information by competitors in the international marketplace would increase their ability to market comparable products or services by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on WEC's potential for obtaining or maintaining foreign licenses.

Ian C. Rickard
Licensing Project Manager
Westinghouse Electric Company LLC

Sworn to before me this 29th day of July, 2002

Notary Public

My commission expires: 09/30/2004

Attachment 2

NRC Mechanical and Civil Engineering Branch Questions and APS Responses

Mechanical & Civil Engineering Branch

NRC Question 1:

The Nuclear Steam Supply System (NSSS) at Palo Verde Nuclear Generating Station (PVNGS), Unit 2 was approved by the NRC staff via NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System Combustion Engineering Standard Safety Analysis Report (CESSAR) System 80." The CESSAR describes the design of the reactor coolant system (RCS), its components, and their supports. The CESSAR describes the methodologies used to develop limiting loads and their locations, and also contains interface requirements between the CE-supplied System 80 NSSS and the rest of the plant. The PURLR, Attachment 6 to the application implies that the analyses which support steam generator replacement (SGR) and power uprate (PUR) may be significantly changing the CESSAR methodologies and assumptions for Unit 2.

NRC Question 1.a:

With respect to RCS stresses, including piping, components, supports, and tributary piping, provide a clear description of the methodologies used for determining the limiting stresses and cumulative usage factors (CUFs) for the SGR/PUR conditions. Describe any changes to the methodologies that were approved as part of CESSAR, and justify the acceptability of any methodology changes for showing compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (i.e., ASME Code) requirements. Describe any significant changes to the design transients that are used for the structural design of the NSSS. Also, discuss any changes to the interface requirements that resulted from the SGR/PUR.

APS Response:

The ASME Boiler and Pressure Vessel Code (B&PV) Code provided the basis for all RCS component stress evaluations. The original code year from the Analyses of Record (AORs) were maintained in all cases with the exception of the Replacement Steam Generators (RSGs). The Code of Record for Original Steam Generators (OSGs) is ASME, Sections II, III, V, and XI, 1971 up to and including Winter 1973 Addenda. The RSGs were designed and fabricated to the newer, up to date requirements of the 1989 Edition (no Addenda) of the ASME Code, Sections II, III, V, and XI.

The engineering evaluations performed for the RSGs and operation at PUR conditions consisted of establishing new revised loads and comparing them with the ASME code allowables for each load case and component.

The only methodology changes from CESSAR are:

1) Implementation of reduced dynamic effects of non-main coolant piping breaks due to application of the NRC approved Leak-Before-Break (LBB) methodology to the Main Coolant Loop (MCL) piping, and
2) the use of the ANSYS computer code to replace the STRUDL (used for RCS and Component Seismic analyses), CEDAGS (used for RCS Loss of Coolant

Accident (LOCA)), and SAPIV (used for a new three-dimensional CEDM model) computer codes.

The new ANSYS computer models were benchmarked with the STRUDL and CEDAGS original models and found to be acceptable for the intended application.

Before performing the structural integrity analyses of the Reactor Vessel (RV), Pressurizer, Reactor Coolant Pumps (RCPs), RCS piping and their supports, new Normal Operating Pressure (NOP), Seismic and LOCA (based on Branch Line Pipe Break (BLPB)/LBB methodology) loads on the RCS were determined for the RSG and operation at PUR conditions. New Dead Weight (DW) and thermal expansion loads (i.e., the load component comprising the NOP loads) were determined by applying conservative factors to the existing DW and thermal loads. Finite Element Analyses (FEA) using detailed RCS and structural building models were use to determine the new seismic and LOCA loads. Each model utilized appropriate boundary conditions. In the case of the LOCA analyses, the component and piping support gaps were also modeled, which required using non-linear FEAs. The seismic and LOCA analyses were based on the original licensing basis time history analysis methods, and the newly generated BLPB type forcing functions for simulated breaks involving the Main Steam Lines (MSLs), Feedwater (FW) lines, Shutdown Cooling (SDC) lines, Safety Injection (SI) lines, and the pressurizer surge line. These models utilized the ANSYS computer code.

All new component loads were tabulated and compared with the original component loads. In most cases, the original design loads remained bounding. For the few instances where new loads exceeded those of the AORs, new design loads (including nominal margins of 20%) were generated and incorporated into the component specifications. The new loads were either reconciled or re-analyzed, compared to the ASME code allowables to verify acceptability, and were documented in calculations and in the summary form in the Design Report Addenda for each affected component. For components where the original design loads did not change or remained bounding, both the component specification and Design Report were amended to indicate that the effects of RSG/PUR were examined and that there was no impact on the AORs.

The as-calculated loads and motions, for RSG and PUR operation, throughout the RCS formed the basis for determining conservative design loads for the various RCS components and their supports. It was concluded that normal operating conditions (pressure and temperature) did not change significantly from the original design used in the AOR. In addition, the effects of RSG/PUR on Anticipated Operational Occurrences (AOOs) were examined. It was determined that the original AORs for AOOs remained bounding for operation at PUR conditions. Therefore, the existing fatigue evaluations and resulting CUFs were not affected, and the existing AORs remained valid.

Asymmetric loadings on the RV head that result from the Control Element Drive Mechanisms (CEDMs) and RV head service structure were addressed using LBB methodology. This methodology was also used to address the In-Core Instrumentation (ICI) guide tubes and the ICI nozzles that are located at the bottom of the RV. BLPB type LOCA loads were applied to the MCL piping, RCPs, and their associated supports. The faulted loads used in the AORs remain bounding.

Interfacing loads with the containment building changed slightly. The RV supports were unaffected. The vertical loads on the Steam Generator (SG) sliding base increased due to a higher RSG weight. The effect of the higher loads on the stresses in the SG vertical support structures (pads, sliding base, bolting, key ways etc.) was verified to be acceptable. The horizontal loads from the RSG into the snubber/linkage system were determined both for Seismic Operating Basis Earthquake (OBE) and combined Seismic Safe Shutdown Earthquake (SSE)/LOCA conditions. The new snubber loads were below the original design loads. The new loads into the lateral key ways were increased somewhat over the originally calculated as-built loads, but were still bounded by the original design loads.

Enclosure 1, Excerpts from the Impact of RSG and PUR on the Structural Integrity of the RCS Components and Supports for Palo Verde Nuclear Generating Station, provides additional information regarding RCS components and supports. This enclosure is Westinghouse Electric Co. proprietary information and should be withheld from public disclosure pursuant to 10 CFR 2.790.

NRC Question 1.b:

The application indicates that leak before break (LBB) is being utilized for the design of more components than discussed in the CESSAR or in the current licensing basis for Unit 2. Describe and justify the new applications of LBB and/or changes in the postulated break locations. Discuss whether these applications of LBB are based on a generic staff safety evaluation, or whether they are changes to the licensing basis that require specific NRC staff review and approval. Also, provide additional information on the continued applicability of LBB for the SGR/PUR conditions (i.e., evaluate the SGR/PUR condition against the criteria evaluated in Supplement 3 to NUREG-0852, including the margin between the leakage-size crack and the critical-size crack, and the material properties of the replacement steam generators (RSGs), replacement cold leg elbows, and associated field welds).

APS Response:

In accordance with 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 4, that was in effect during original plant design, the mechanical design basis for the original RCS configuration included postulated breaks in all high energy piping greater than one inch (1") in diameter. The NRC granted a partial exemption from the requirements of 10 CFR Part 50, Appendix A, GDC 4 to APS to allow the application of LBB methodology to MCL piping to eliminate pipe whip restraints and jet shields that were previously needed to mitigate the effects of Main Coolant Loop Breaks (MCLBs). The exemption was granted in:

- Letter from G. W. Knighton, NRC, to E. E. Van Brunt, APS, Docket No. 50-529/530, November 29, 1985, Request for a Schedular Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as a Structural Design Basis for Palo Verde Nuclear Generating Station Units 2 and 3.

The technical bases that demonstrated that guillotine type failures of the RCS main loop piping need not be considered in the design basis for CE designed plants was provided to the NRC in the following letters:

- Letter from A. E. Scherer, CE, to Darrell G. Eisenhut, NRC, Docket No. STN 50-470, June 14, 1983, with enclosure, "Basis for Design of Plant Without Pipe Whip Restraints for RCS Main Loop Piping" and
- Letter from A. E. Scherer, CE, to Darrell G. Eisenhut, NRC, Docket No. STN 50-470F, December 23, 1983, with enclosure, "Leak Before Break Evaluation of the Main Loop Piping of a CE Reactor Coolant System," Revision 1, November 1983.

The NRC staff has previously reviewed and approved these analyses in:

- Letter from Cecil 0. Thomas, NRC, to A. E. Scherer, CE, Docket No. STN 50-470, October 11, 1984, with enclosure, "Safety Evaluation Report on the Elimination of Large Primary Loop Ruptures as a Design Basis."

In addition, the NRC has review and approved the application of LBB methodology to the PVNGS units in Section 3.6.2 of NUREG-0857, Safety Evaluation Report Related to the operation of Palo Verde Nuclear Generating Stations, Units 1, 2, and 3, through Supplement 12.

APS applied LBB methodology on a limited basis to allow removal of selected component and piping snubbers. APS increased the scope of the application of LBB methodology during the engineering and design phase of RSG/PUR to include all RCS tributary and MCL nozzles. This increased scope is within the scope of the approved LBB methodology. In addition,,the MCL piping, including the RSGs and replacement cold leg elbows, was evaluated at PUR conditions and compared to the above references to ensure that application of LBB methodology remained valid for the increased scope of application.

For RSG/PUR, LBB methodology was applied to all primary side branch line piping that interfaces with the RCS. The bounding BLPBs with respect to the RCS response, are breaks in the largest tributary pipes of:

- main steam line,
- feedwater line,
- pressurizer surge line,
- safety injection line, and
- shutdown cooling line.

The terminal end breaks in all five piping systems listed above plus intermediate breaks in the main steam and SI lines remained as controlling pipe breaks. The limiting BLPBs that are considered for RSG/PUR are listed in Table 1.b-1.

Table 1.b-1: Pipe Break Description	
1	FW terminal end (1A quadrant)
2	MSL terminal end (1A quadrant)
3	MSL intermediate (1A quadrant), X thrust
4	MSL intermediate (1A quadrant), Z thrust
5	Surge line (HL 1) terminal end
6	SDC (HL 1) terminal end
7	SI 1A terminal end
8	SI 1B terminal end
9	SI 1B intermediate break

The following paragraphs provide information regarding RCS piping cracks and replacement materials:

Leakage Detection and Crack Stability

The areas of concern are the RV inlet and outlet nozzles since these are regions of relatively high stress. The SG inlet and outlet nozzles are of lesser concern.

For the RSG/PUR configuration, the bending moments for the normal operating loads at both the RV outlet and inlet nozzles are smaller than the original bending moments used to determine the leakage crack length (see the Table 1.b-2). Consequently, the leakage crack under normal operating mechanical loads for the RSG configuration and operation at PUR conditions is longer than the leakage crack for the OSG configuration. The leakage crack is defined as a crack that will leak 10 gpm at normal operating primary system pressure.

Table 1.b-2: NOP Bending Moment Comparison (ft-kips)		
Location	Original Values Used for Leakage Crack Determination	RSG and PUR Configuration
RV Outlet Nozzle	6,406	6,049
RV Inlet Nozzle	2,175	790

A 10 gpm leak at the RV outlet nozzle requires a crack that is approximately 8.5% of the pipe circumference. At the RV inlet nozzle under the same conditions, a 10 gpm leak requires a crack that is approximately 11% of the pipe circumference. This means that with the addition of any mechanical load, the crack length required to open a detectable crack will be shorter than when subjected to system pressure alone. Application of LBB methodology demonstrated that a critical crack of 50% of the pipe circumference remains stable when subjected to both normal operating and SSE loads. This confirms that the leakage crack length for the RSG configuration is below the stability criterion used in the LBB methodology.

The normal operating and SSE loads used in the determination of crack stability for the OSG configuration and the normal operating and SSE loads for the RSG configuration are listed in Table 1.b-3. The combined normal operating and SSE loads for the RSG configuration are less than those used for the OSG configuration. Therefore, a leaking crack in the RV outlet nozzle or inlet nozzle occurring after RSG/PUR will be detectable well before the crack can grow to an unstable length.

Table 1.b-3: (NOP + SSE) Bending Moment Comparisons (ft-kips)		
Location	OSG Configuration	RSG and PUR Configuration
RV Outlet Nozzle	6,853	6,535
RV Inlet Nozzle	1,831	934

Replacement Piping, RSGs, and Weld Material Properties

The results of the evaluation of primary piping, RSGs, and weld material properties to be used for the LBB analysis are summarized as follows:

(1) The stress-strain curve used in the original LBB evaluation provides a reasonable representation of the nominal stress-strain properties for the MCL piping base, RSGs, and weld materials considered.

(2) The J-R Curve for SA-516 Grade 70 plate was a good lower bound estimate for plate material. However, some weld metals tend to have an even lower toughness property. The lower bound SA-516 weld metal curve is considered a more appropriate lower bound for the MCL piping, RSGs, and weld materials being considered in this evaluation.

(3) The original LBB analysis resulted in an acceptable margin when the measured toughness properties were degraded by a factor of four. Since the weld metal lower bound toughness properties are higher than one fourth of the toughness properties used in the original analysis, the original analysis remains conservative and valid for the lower bound weld metal J-R Curve.

In addition, the replacement RCS piping is at the RSG outlet nozzles, which are not at the critical stress locations used in the LBB analysis.

The short-term LOCA-related mass and energy releases are used as inputs to the subcompartment analyses that are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a high energy line rupture within that subcompartment. The evaluated subcompartments include the SG compartment, the reactor cavity region, and the pressurizer compartment. For the SG compartment and the reactor cavity region, LBB methodology was used to qualitatively demonstrate that any changes associated with operation at PUR conditions are offset by the LBB benefit of considering smaller RCS nozzle breaks. The current licensing bases for these compartments remains bounding.

The pressurizer subcompartment analysis assumes a double-ended guillotine break of the pressurizer surge line. The energy release rates were calculated for operation at PUR conditions and compared with the original plant design conditions. The original plant design energy release rates continue to bound the PUR plant energy release rates by approximately 10%. Since the AOR determined that the pressurizer subcompartment was adequate, and the energy release rates for the PUR condition remain bounded by the original energy release rates, the pressurizer subcompartment remains structurally acceptable.

The calculated peak internal containment pressure, which does not credit LBB, increases to 58 psig as a result of operation at PUR conditions and RSGs, but remains bounded by the original containment internal design pressure of 60 psig.

NRC Question 2:

For the RCS piping (including pressurizer (PZR) surge line and tributary piping), components (including reactor vessel (RV), reactor coolant pumps (RCPs), RSGs, and PZR), and supports, provide the calculated maximum stresses and CUFs at the critical locations. Include the ASME Code allowable limits and the ASME Code edition and addenda used in the evaluation of the SGR/PUR conditions. If different from the ASME Code of record, provide a justification.

APS Response:

The RCS piping (including pressurizer and RCS surge line nozzles and tributary line nozzles), components (including RV, RCPs, and pressurizer), and supports, were evaluated to address the effects of the RSGs and operation at PUR conditions. Enclosure 1 provides the calculated maximum stresses and CUFs for RCS piping, components, and supports at the critical locations.

The original ASME Code of Record for evaluation of the RCS tributary piping and pressurizer surge line is ASME Section III, 1974 Edition, including the Winter 1975 addenda (Summer 1979 addenda for Subsections NB-3650 through 3680) and is documented in UFSAR Table 3.9-5. The evaluation of RCS tributary piping and pressurizer surge line to support operation at PUR conditions with RSG were performed using the original ASME Code of Record.

Table 2-1 is a summary of the revised Class 2 stress analysis performed for MCL tributary piping and the pressurizer surge line. This table provides the maximum calculated stress for each ASME defined condition and the associated ASME allowable stress. Table 2-2 provides the Class 1 maximum and allowable stresses, and CUFs that were calculated in the original AOR. The original AOR bounds the stresses associated with RSG/PUR.

The RSGs have been designed and fabricated to the new, up to date requirements of the ASME Code 1989 Edition with no addenda. The OSG were designed and fabricated to the requirements of ASME Code, 1971 Edition including all addenda through Winter 1973. The differences between the code of design and fabrication and the Code of Record will be reconciled before installation of the RSGs as required by ASME sections NCA 1140 and NCA 3220.

The stresses at the critical component locations and CUFs for the RSGs are summarized in Table 2-3.

Table 2-1: Maximum Structural Stresses in Tributary Piping and Pressurizer Surge Line, Class 2

Piping Segment	Equation 8		Equation 9B		Equation 9D		Equation 10/11	
	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)
CVCS - Charging Line	10,313	15,900	17,284	19,080	27,835	38,160	25,752	27,475
CVCS - RCS Loop Drains	7,706	15,900	12,411	21,816	31,321	38,160	34,701	43,375
CVCS - Auxiliary Spray Lines	10,684	15,900	17,762	19,080	25,791	38,160	25,652	26,888
Pressurizer Surge Line	4,741	15,900	4,801	19,080	8,041	38,160	16,198	27,475
Pressurizer Spray Line	10,684	15,900	17,762	19,080	25,791	38,160	25,652	26,888
SDC Lines	8,775	15,900	18,171	19,080	31,464	38,160	40,877	43,375
SI Loops 1A and 1B	10,327	18,600	15,051	22,320	22,540	39,360	27,082	27,600
SI Loops 2A and 2B	10,864	18,600	19,394	19,680	33,116	39,360	27,935	28,150

Table 2-2: Class 1 Maximum Stresses and CUFs from original AOR Tributary Piping and Pressurizer Surge Line

System	Equation 9		Equation 10	Equation 12	Equation 13	Allowable	CUF
	Calculated	Allowable				3.0 x S_m	
CVCS Aux Spray Line Maximum stress	15,468	24,225	56,147	8,793	44,629	48,450	0.5137
CVCS Charging Line Maximum Stress	17,362	27,000	67,038	8,178	38,630	54,000	0.3204
RCS Coolant Loop Drains Maximum Stress	12,673	24,225	62,332	40,384	22,243	48,450	0.2302
CVCS – Letdown Line Maximum Stress	22,380	24,230	56,950	25,700	43,000	48,450	0.189
Pressurizer Spray Line Maximum Stress	20,716	24,225	61,817	26,150	47,255	48,450	0.6029
SDC - Loop 1 Maximum Stress	23,792	29,025	73,095	42,415	53,587	58,050	0.257

Table 2-2: Class 1 Maximum Stresses and CUFs from original AOR Tributary Piping and Pressurizer Surge Line							
System	Equation 9 Calculated	Equation 9 Allowable	Equation 10	Equation 12	Equation 13	Allowable 3.0 x S_m	CUF
SDC - Loop 2 Maximum Stress	13,843	29,025	68,304	49,413	44,646	56,100	0.1573
SI Maximum Stress Loop 1A	17,130	29,030	68,290	19,630	39,360	58,050	0.1819
SI Maximum Stress Loop 1B	13,840	24,900	49,310	0	0	58,050	0.1854
SI Maximum Stress Loop 2A	16,190	25,140	98,095	28,454	57,848	58,050	0.753
SI Maximum Stress Loop 2B	19,953	29,025	86,446	28,454	57,777	58,050	0.274
Pressurizer Surge Line Maximum Stress	13,677	24,000	58,319	41,700	39,461	48,564	0.495
Pressurizer Surge Line CUF Including Stratification							0.937

Table 2-3: Maximum Stresses and CUFs for RSG Components					
Component	Stress Category	Maximum Stress (psi) (A)	Allowable Stress (psi) (B)	Stress Ratio (A)/(B)	CUF
Tubesheet and Primary Head	P_m $P_l + P_b$ $P_l + P_b + Q$	26,500 38,300 79,800	$S_m = 26700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.99 0.96 0.997	0.996
Primary Inlet Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	24,300 29,900 47,600	$S_m = 26{,}700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.91 0.75 0.59	0.0416
Primary Outlet Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	17,000 23,600 42,400	$S_m = 26{,}700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.64 0.59 0.53	0.017
Primary Manway	P_m $P_l + P_b$ $P_l + P_b + Q$	5,900 22,900 41,100	$S_m = 26{,}700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.22 0.57 0.51	0.037
Stud	Replace every 6 (Six) years.				
Primary Divider Plate	P_m $P_l + P_b$ $P_l + P_b + Q$	8,600 30,500 64,500	$S_m = 26{,}700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.37 0.87 0.92	0.08
Support Skirt	P_m $P_l + P_b$ $P_l + P_b + Q$	19,100 23,600 79,400	$S_m = 26{,}700$ $1.5\ S_m = 40{,}050$ $3.0\ S_m = 80{,}100$	0.72 0.59 0.99	0.155

Table 2-3: Maximum Stresses and CUFs for RSG Components					
Component	Stress Category	Maximum Stress (psi) (A)	Allowable Stress (psi) (B)	Stress Ratio (A)/(B)	CUF
Tube to Tubesheet Weld	P_m	22,400	$S_m = 26,700$	0.96	0.668
Tubes and Tube Upper Supports	P_m $P_l + P_b$ $P_l + P_b + Q$	20,400 20,400 36,200	$S_m = 26,600$ $1.5\ S_m = 39,900$ $3.0\ S_m = 79,800$	0.77 0.51 0.45	0
Secondary Shell	P_m $P_l + P_b$ $P_l + P_b + Q$	25,350 31,600 39,000	$S_m = 26,700$ $1.5\ S_m = 40,050$ $3.0\ S_m = 80,100$	0.949 0.789 0.487	0.009
Economizer FW Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	14,600 24,000 79,800	$S_m = 18,250$ $1.35\ S_m = 24,640$ $3.0\ S_m = 80,100$	0.8 0.98 0.996	0.981
Downcomer Blowdown Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	13,700 30,400 54,400	$S_m = 21,500$ $1.5\ S_m = 32,300$ $3.0\ S_m = 64,500$	0.64 0.94 0.84	0.255
Downcomer FW Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	14,000 19,300 58,700	$S_m = 23,330$ $1.5\ S_m = 24,600$ $3\ S_m = 63,900$	0.6 0.78 0.92	0.996
Recirculation Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	11,800 28,400 47,500	$S_m = 18,200$ $1.5\ S_m = 40,000$ $3\ S_m = 80,100$	0.65 0.71 0.59	0.107
Steam Outlet Nozzle	P_m $P_l + P_b$ $P_l + P_b + Q$	22,500 27,000 38,400	$S_m = 26,700$ $1.5\ S_m = 40,000$ $3\ S_m = 80,100$	0.84 0.67 0.48	0.169
Secondary Shell Instruments	P_m $P_m + P_b + Q$	2,000 22,900	$S_m = 18,200$ $3\ S_m = 55,100$	0.11 0.41	Exempt
Primary Head Instruments	P_m $P_m + P_b + Q$	9,000 63,700	$S_m = 23,300$ $3\ S_m = 69,900$	0.39 0.91	Exempt
Tubesheet Blowdown	$P_l + P_b$ $P_m + P_b + Q$	26,000 69,900	$1.5\ S_m = 31,600$ $3\ S_m = 69,900$	0.82 1	Exempt
Secondary Manway	$P_l + P_b$ $P_l + P_b + Q$	32,600 58,600	$1.5\ S_m = 40,000$ $3\ S_m = 80,100$	0.81 0.73	0.128
Bolt					0.771
Secondary Handholes	P_m $P_l + P_b$ $P_l + P_b + Q$	26,300 32,500 68,900	$S_m = 26,700$ $1.5\ S_m = 40,000$ $3\ S_m = 80,100$	0.97 0.81 0.86	0.944
Stud	Replace studs every eighteen years				
Upper Support Lugs	P_m $P_l + P_b$ $P_l + P_b + Q$	12,300 27,900 46,800	$S_m = 23,300$ $1.5\ S_m = 40,000$ $3\ S_m = 80,100$	0.46 0.7 0.58	0.141

Table 2-3: Maximum Stresses and CUFs for RSG Components					
Component	Stress Category	Maximum Stress (psi) (A)	Allowable Stress (psi) (B)	Stress Ratio (A)/(B)	CUF
Dryers Assembly Design	$P_l + P_b$ Shear	3,500 5.9	28,700 20,000	0.12 0.3	Exempt
Dryers Assembly Design Level D	$P_l + P_b$ Shear	25,800 28,800	73,500 42,000	0.35 0.69	Exempt
Separators Design	P_m $P_l + P_b$	550 4,100	12,600 28,700	0.04 0.14	Exempt
Separators Design Level D	P_m $P_l + P_b$	740 42,900	40,600 73,500	0.12 0.58	Exempt
Shroud Assembly Design	P_m $P_m + P_b$	7,200 19,200	19,100 28,700	0.38 0.67	Exempt
Shroud Assembly Design Level D	P_m $P_m + P_b$	8,700 29,100	49,000 73,500	0.18 0.4	Exempt
Eggcrate Assembly Design	P_m $P_l + P_b$	8,700 0	18,100 21,100	0.48 0	Exempt
Eggcrate Assembly Design Level D	P_m $P_l + P_b$	29,600 24,400	40,600 58,500	0.73 0.42	Exempt
Downcomer FW Piping Assembly Design	$P_m + P_b$	8,700	27,100	0.32	Exempt
Downcomer FW Level A/B	$P_m + P_b$	19,000	47,500	0.4	0.125
Downcomer FW Level D	$P_m + P_b$	1,700	63,000	0	Exempt
FW Distribution Box	$P_l + P_b + Q$	52,500	57,500	0.91	0.961

<u>NRC Question 3:</u>

Section 5.3.3.1 of the PURLR indicates that the response spectrum for the containment basemat in the vertical direction for the operating basis earthquake is not bounded by the analysis of record. Provide an evaluation of the containment basemat stresses for this condition.

<u>APS Response:</u>

Section 5.3.3 of the PURLR describes the RV ICI Tubes. The wording of Section 5.3.3.1 should more appropriately read:

The ICI tubes are attached to the RV bottom head and terminate at the ICI seal table. The original analysis of the ICI tubes was based on a set of "Configuration Spectra" that was intended to envelop both the RV configuration spectra and the containment basemat spectra. RSG/PUR did not affect the containment basemat spectra. However, the RV configuration spectra were affected somewhat, which required reconciliation of the new RV spectra. Reconciliation demonstrated that the new RV spectra is comparable to the original design spectra.

Review of the original design OBE spectra verified that the existing ICI configuration spectra completely envelop both the PUR configuration RV spectra and the containment basemat spectra for the X direction (horizontal and parallel to the hot leg) and the Z direction (horizontal and perpendicular to the hot leg). However, for the vertical (Y) direction, both the PUR configuration RV spectra and the original containment basemat spectra exceed the existing ICI configuration spectra at certain frequencies. Since total enveloping of the containment basement and RV vertical spectra could not be demonstrated, it was necessary to recalculate the response of the ICI tubes to PUR RV conditions and to the original basement spectra, and to verify that this response is bounded by the original/existing ICI configuration spectra.

The AOR provided the response for selected locations for each mode and each excitation direction. The response to the PUR configuration spectra was calculated by multiplying each AOR modal response by the ratio of the new to the original modal accelerations. The response for each excitation direction was calculated by combining the modal responses in accordance with Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis. The total response of the ICI tubes was then calculated using the Square-Root-of-the-Sum-of-the-Squares (SRSS) of the response due to the X, Y, and Z directions. The load results demonstrated that the response of the ICI tubes to the OBE excitation for the PUR configuration is less than the response calculated for the existing configuration. Therefore, all ICI OBE results for the existing ICI spectrum configuration are valid and remain bounding for operation at PUR conditions.

The original AOR containment basemat spectra and the motions of containment were not changed as a result of the RSG/PUR.

<u>NRC Question 4:</u>

For the RV internals provide the maximum calculated stresses and CUFs for the SGR/PUR condition. Include the ASME Code allowable limits used in the evaluation, and the ASME Code edition and addenda. If different from the ASME Code of Record, provide a justification.

<u>APS Response:</u>

The ASME Code from the AORs was maintained in all cases with exception of the RSG. The Code of Record for OSGs is ASME Code, Section II, III, V, and XI, 1971, up to and including Winter 1973 Addenda. The RSGs were designed and fabricated to the requirements of the new, updated 1989 Edition of the ASME Code, Sections II, III, V, and XI, no Addenda.

Tables 4-1 and 4-2 provide the maximum stress versus allowable stress values for normal operation plus upset condition, and faulted design condition, respectively. Table 4-1 also includes the CUFs for the RSG/PUR condition. Supplemental information may be obtained from Enclosure 1.

Table 4-1: RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition					
Major Assembly	Component	Stress Category [5]	Maximum Stress (psi)	Allowable (psi) [1]	CUF (1.0 Allowable)
Core Support Barrel (CSB) Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	5,082 14,151 29,099	16,100 24,150 48,300	< 0.462
	Cylinders	P_m $P_m + P_b$ $P_m + P_b + Q$	5,278 9,214 21,694	16,100 24,150 48,300	< 0.462
	Lower Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	2,815 10,166 23,541	16,100 24,150 48,300	< 0.462
	Snubber to Cylinder Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	7,717 14,275 25,785	14,490 21,735 43,470	< 0.462
	CSB to Lower Support Structure (LSS) Flexure Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	2,814 3,947 18,107	14,490 21,735 43,470	< 0.462
LSS Components	Insert Pin	P_m $P_m + P_b$	2,520 11,377	43,300 64,950	< 0.078
	Main Beam to Short Beam Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	953 8,261 37,246	14,490 21,735 43,470	< 0.078
	Main Support Beam	P_m $P_m + P_b$ $P_m + P_b + Q$	10,668 14,418 28,317	14,490 21,735 43,470	< 0.078
	Cylinder	P_m $P_m + P_b$ $P_m + P_b + Q$	6,104 8,633 27,788	14,490 21,735 43,470	< 0.078
	Raised Bottom Plate	P_m $P_m + P_b$ $P_m + P_b + Q$	964 12,036 13,692	14,490 21,735 43,470	< 0.078

Table 4-1: RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition					
Major Assembly	Component	Stress Category [5]	Maximum Stress (psi)	Allowable (psi) [1]	CUF (1.0 Allowable)
Upper Guide Structure (UGS) Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	10,534 22,490 31,994	16,100 24,150 48,300	< 0.31
	Lower Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	1,651 12,918 22,833	16,100 24,150 48,300	< 0.31
	Control Element Assembly (CEA) Guide Tube	P_m $P_m + P_b$ $P_m + P_b + Q$	1,413 11,458 39,458	16,100 24,150 48,300	0
	Guide Tube to Upper Guide Structure Support Plate (UGSSP) Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	1,391 2,525 17,226	12,075 18,113 36,225	0
	UGS Support Plate	P_m $P_m + P_b$ $P_m + P_b + Q$	645 13,596 37,949	16,100 24,150 48,300	0.118
	Fuel Alignment Plate	P_m $P_m + P_b$ $P_m + P_b + Q$	705 13,891 24,270	16,100 24,150 48,300	0.721
	Tube to Fuel Alignment Plate (FAP) Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	421 4,166 4,166	12,075 18,133 36,225	0
Internal Structures	Core Shroud	P_m $P_m + P_b$ $P_m + P_b + Q$	9,825 22,838 39,920	16,100 24,150 43,470	N/A [3]
	CEA Shroud Tubes	$P_m + P_b$ $P_m + P_b + Q$	22,464 38,073	24,150 48,300	0.32
	CEA Shroud Tube to Web Weld	$P_m + P_b$ $P_m + P_b + Q$	7,890 23,499	8,452 48,300	0.093

Table 4-2: RVI Stress Summary for RSG and PUR - Faulted Design Condition				
Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
CSB Components	Upper Flange	P_m $P_m + P_b$	16,857 42,802	38,640 57,960
	Cylinders	P_m $P_m + P_b$	30,998 42,814	38,640 57,960
	Lower Flange	P_m $P_m + P_b$	14,826 34,622	38,640 57,960
	Snubbers @ Shell	P_m $P_m + P_b$	10,314 12,260	34,776 52,164
	CSB to LSS Flexure Weld	P_m $P_m + P_b$	14,826 31,160	34,776 52,164
LSS Components	Insert Pin	P_m $P_m + P_b$	6,152 27,779	91,000 136,500
	Main Beam to Short Beam Weld	P_m $P_m + P_b$	24,633 49,057	34,776 52,164
	Main Support Beam	P_m $P_m + P_b$	21,656 24,657	34,776 52,164
	Cylinder	P_m $P_m + P_b$	33,890 41,432	34,776 52,164
	Raised Bottom Plate	P_m $P_m + P_b$	2,673 34,113	34,776 52,164
UGS Components	Upper Flange	P_m $P_m + P_b$	35,415 49,758	38,640 57,960
	Lower Flange	P_m $P_m + P_b$	11,203 54,372	38,640 57,960
	CEA Guide Tube	P_m $P_m + P_b$	4,657 12,125	38,640 57,960
	Guide Tube to UGSSP Weld	P_m $P_m + P_b$	5,974 6,173	28,980 43,470
	UGS Support Plate	P_m $P_m + P_b$	2,199 44,894	38,640 57,960
UGS Components	Fuel Alignment Plate	P_m $P_m + P_b$	2,418 44,891	38,640 57,960
	Tube to FAP Weld	P_m $P_m + P_b$	451 5,076	28,980 43,470

Table 4-2: RVI Stress Summary for RSG and PUR - Faulted Design Condition				
Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
Internal Structures	Core Shroud	P_m $P_m + P_b$	31,677 57,747	38,640 57,960
	CEA Shroud Assembly	N/A (4)	.109 inch (4)	.628 inch (4)

Notes for Tables 4-1 and 4-2:

(1) Allowable stress criteria defined in ASME B&PV Code, Section III, Division 1, Subsection NG, 1974 Edition without addenda.

(2) Allowable stress criteria defined in ASME B&PV Code, Section III, Division 1, Appendix F, 1974 Edition without addenda.

(3) For RSG/PUR, it was determined that the AOR bound the structural evaluation of the Core Shroud. The AOR did not calculate fatigue usage.

(4) The CEA Shroud is deflection-limited, rather than stress-limited, in the faulted condition.

(5) Stress categories are as defined below:

P_m = Primary membrane stress

$P_m + P_b$ = Primary membrane plus bending stress

$P_m + P_b + Q$ = Primary membrane plus bending plus secondary stress

NRC Question 5:

For the control element drive mechanisms (CEDMs), the PURLR describes changes in the methodology for determining stresses and CUFs. Describe the benchmarking of the new methodology, and discuss the new methodology's acceptability for determining stresses and CUFs for the SGR/PUR condition. Provide the maximum calculated stresses and CUFs at the critical locations of the CEDMs for the SGR/PUR condition. Include the ASME Code allowable stresses and the ASME Code edition and addenda used in the evaluation of SRG/PUR. If different from the ASME Code of record, provide a justification.

APS Response:

The changes in the methodology are the use of a new three-dimensional CEDM model, use of the ANSYS code, rather than SAPIV code (for the response spectrum analyses), and inclusion of BLPB loads in the faulted loads.

The new CEDM model uses a sufficient number of nodes to accurately represent the dynamic characteristics of the CEDM nozzle components and to provide a detailed load response distribution throughout the CEDM structure. The model was benchmarked by comparison of the calculated natural frequencies and mode shapes to test data. The first, second, and third mode natural frequencies of the CEDM model with the longest nozzle length were calculated to be 2.33 Hz, 9.61 Hz, and 10.32 Hz. These calculated values compare favorably with experimentally measured values of 2.32 Hz, 9.2 Hz, and 11.6 Hz.

The CEDM model was used for DW, seismic, and BLPB analyses for the CEDM nozzles. The calculated response loads were used to evaluate the CEDM components operability for the upset condition (NOP plus OBE results) and the faulted condition (NOP plus SRSS combination of SSE and BLPB) results. CEDM operability was verified by comparing the predicted maximum CEDM deflections due to the combination of SSE and BLPB with those shown to be acceptable by static testing. A maximum calculated combined deflection was used in this comparison for conservatism.

The CUFs for all CEDM components calculated in the AOR remain valid for RSG/PUR. This is also true for the nozzle because the small change in the OBE stresses does not affect the fatigue evaluation.

Enclosure 1 provides a more detailed description of the CEDM analysis, as well as tables that list the results of the analysis, design loads used in the AOR, and a figure showing the CEDM model.

The CEDM analysis results can be summarized as follows:

1) The NOP mechanical loads are comparable to the AOR.
2) The calculated OBE loads are similar to the AOR loads.
3) The faulted loads are the sum of the NOP loads with the SRSS of the SSE and BLPB loads. The calculated faulted loads are lower than those in the AOR because the SSE horizontal response spectra are less severe.
4) The calculated loads for all CEDM components are less than the design loads used in the AOR except for the NOP plus OBE load on the nozzle and the axial load on the upper pressure housing.
5) The most limiting load for all CEDM components is the NOP plus OBE load on the nozzle. The calculated bending moment increased from the design value of 111.7 in-kips to 112.1 in-kips. However, the calculated nozzle stress remained less than the ASME Code allowable.
6) The increase in the axial load on the upper pressure housing is not significant for two reasons. The axial load causes a minimal increase in stress, and the AOR included a conservative DW load that covers the difference between the calculated and design loads. Therefore, the existing AOR remains bounding for operation at PUR conditions.

As discussed above, the design loads were all greater than the calculated loads, except for the NOP plus OBE load on the nozzle. Therefore, except for the Level A and B stress in the nozzle, all stresses calculated in the AOR remain valid. The calculated Levels A and B stress intensity increased by 2.8% from the AOR. The calculated Level A and B general primary membrane stress intensity in the CEDM nozzle is 22.86 ksi, compared to an allowable of 23.3 ksi. The allowable stress is from the ASME Code, 1971 Edition with Addenda through the Winter of 1973, which is the ASME Code of Record.

NRC Question 6:

Discuss the potential for flow-induced vibration of the steam generator (SG) tubes due to various mechanisms, including, the fluid-elastic instability, in the RSG at the PUR condition. Describe the analysis methodology, damping value of the tubes, and the computer code used in the analysis. Also, provide the results of the predicted vibration levels during the normal operating condition and the worst case transient condition, including the calculated fluid-elastic instability ratios. Explain whether the above analysis results are applicable to the degraded SG condition and why.

APS Response:

The potential for Flow-Induced Vibration (FIV) is minimized due to SG design. The FIV analyses were performed on selected tubes based on tube span parameters, such as frequency and mode shape of vibration, and fluid flow parameters, including flow velocity and fluid density. The selected tubes were modeled with the ANSYS finite element code in order to determine their natural frequencies and mode shapes. The finite element models represent the entire length of the tube above the tubesheet, and include the fluid entrance regions in the vertical tube spans and the fluid exit regions in the tube bends. The tube supports are represented by displacement boundary conditions, which model the following types of tube supports at the appropriate locations along the length of the tube:

- the diagonal and vertical strip supports restrain the out-of-plane displacement of the tube,
- eggcrate supports restrain the in-plane and out-of-plane displacement normal to the tube axis, and
- vertical strip supports with horizontal bars above and below the tube restrain all the in-plane and out-of-plane displacements.

The FIV analysis considers both fluid elastic instability and random turbulent excitation mechanisms. The methodology employed in the evaluation of fluid elastic instability is based on the experimental results and analytical procedures described in:

- Heilker, W.J., Vincent, R.Q., "Vibration in Nuclear Heat Exchangers Due to Liquid and Two-Phase Flow," Engineering for Power, April 1981, Vol. 103, No. 2.
- Heilker, W.J., Beard, N.L., Park, J.Y., "Flow Induced Vibration Analysis in Support of Design of the Yonggwang Units 3&4 Steam Generators," Proceedings of the International Symposium of Pressure Vessel Technology and Nuclear Codes and Standards, April 19-21, 1989.
- Conners, H.J., Jr., "Fluidelastic Vibration of Heat Exchanger Tube Arrays," ASME Publication 77-DET-90, 1977.
- Connors, H.J., Jr., "Fluidelastic Vibration of Tube Arrays Excited by Nonuniform Cross Flow", Flow-Induced Vibration of Power Plant Components, ASME, PVP-41, 1980, p. 93.

The random turbulent excitation was evaluated using the methodology outlined in Article N-1340, Appendix N of the 1995 Edition of the ASME B&PV Code Section III and in Welding Research Council Bulletin No. 372. The damping values used in both of these evaluations vary with mode frequency and were obtained from tests of prototypically supported tubes. All FIV calculations include thermal-hydraulic results from the ATHOS code, using the appropriate tube support arrangement.

The evaluation results showed a maximum Stability Ratio of 0.38 in the fluid exit bend region of the bundle. This result is below the design goal of 0.75. The turbulent displacements are also within the limit of 10 mils Root Mean Squared (RMS). The results reflect the beneficial effects of the design changes in the tube support system of the RSG. Results of the predicted vibration levels are not provided since the evaluation results demonstrated favorable stability ratios for normal and worst case transient conditions.

NRC Question 7:

Describe any changes to the thermal stratification of the PZR surge line and any changes to the thermal fatigue of the pressurizer spray nozzle.

APS Response:

The impact of operation at PUR conditions on the thermal load and transients for the pressurizer surge line and the spray line/nozzle were reviewed. The AORs remain bounding.

NRC Question 8:

Describe the methodology used to evaluate the balance of plant (BOP) piping, components (including pumps, valves, and heat exchangers), and supports. Justify differences from the original design methodology. Also, provide the calculated maximum stresses for the critical BOP piping systems. Include the ASME Code edition and addenda and ASME Code allowable limits. If different from the ASME Code of record, provide a justification.

APS Response:

The Balance of Plant (BOP) piping, components, and supports were originally evaluated using ASME Code, 1974 Edition, including the Winter 1975 addenda, as documented in UFSAR Table 3.9-5. The affected BOP piping, components, and supports were evaluated using the same methodology. Stress limits, as stated in Table 3.9-6 of the UFSAR, were used in BOP piping and component evaluations and stress limits, as stated in UFSAR Table 3.9-11, were used in BOP support evaluations.

There were no changes to the original ASME Code of Record for the BOP piping. The calculated maximum stresses in the affected BOP piping are provided in Table 8-1 for each ASME defined condition evaluated (i.e., normal, upset, emergency, and faulted).

The BOP heat exchangers, pumps, and valves that are most affected by RSG/PUR are those subcomponents of the main steam system. These include main steam safety valves (MSSVs), Main Steam Isolation Valves (MSIVs), turbine stop valves, turbine throttle valves, low pressure FW heaters, high pressure FW heaters, condensate pumps, FW pumps, and heater drain pumps. These components were evaluated at the predicted PUR mass flow rates, temperatures, and pressures. The original component design requirements were found to bound operation at PUR conditions.

Table 8-1: Maximum Stresses In BOP Piping Systems								
	Equation 8		Equation 9B		Equation 9D		Equation 10/11	
Piping Segment	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)	Calculated (psi)	Allowable (psi)
SG Blowdown	9,694	15,000	17,298	18,000	33,366	36,000	22,037	22,500
Downcomer FW & Recirculation	9,778	15,000	16,130	18,000	22,338	36,000	9,511	22,500
Main Steam	5,991	17,500	12,612	21,000	26,624	42,000	9,053	26,250

NRC Question 9:

The PUR results in an increase in the main steam flow and the feedwater flow. Discuss the potential for flow-induced vibration in the main steam and feedwater piping and the BOP heaters and heat exchangers following PUR. Also, clarify whether vibration monitoring, consistent with OM-3, will be included in the startup testing program for PUR.

APS Response:

All secondary side component steam and water operating velocities are predicted to remain below the original component design limits when operating at PUR conditions. Post-PUR startup testing will be conducted in accordance with the requirements of ASME OM-S/G-2000, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants", including Part 3, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems," which supercedes ANSI/ASME/OM-3.

NRC Question 10:

The SGR/PUR increases the post-accident containment temperature and pressure. Discuss the effects of the SGR/PUR on the overpressurization of isolated piping segments (reference: Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions").

APS Response:

The calculated post-LOCA peak containment temperature and the time temperature profile have increased slightly. The slight increase in temperature has a minimal effect on equipment in containment, and the existing analyses performed to address Generic Letter 96-06 remain bounding for operation at PUR conditions. The systems and components within containment subject to post accident environmental heating and pressurization remain capable of withstanding the predicted peak pressures and temperatures.

NRC Question 11:

Confirm whether the SGR/PUR will increase the accident sub-compartment temperature and pressure that affect the design basis for steel and concrete in the containment. If the structural steel and concrete will be affected, provide the design-basis margin and margins after considering increased accident loading due to the SGR/PUR.

APS Response:

Subcompartments within containment, principally the reactor cavity, the steam generator compartments, and the pressurizer compartment, are designed to withstand the transient differential pressures and jet impingement forces of a postulated pipe break. UFSAR Table 6.2-1 contains containment design parameters for postulated accidents considered for subcompartment peak pressure/temperature.

For the pressurizer subcompartment, the AOR assumes a double-ended guillotine break of the pressurizer surge line. The energy release rates were calculated for operation at PUR conditions and compared with the original plant design basis conditions. The original plant design energy release rates continue to bound the PUR plant energy release rates.

Containment Subcompartment Pressure Effects

For the SG and reactor cavity subcompartments, the original design basis for concrete in the containment bounds the predicted subcompartment accident pressures determined by the RSG/PUR analysis. The analyses performed in support of PUR/RSG, which includes the results of the application of the previously approved LBB methodology to the RCS piping, reduced the maximum calculated subcompartment accident pressure loading due to elimination of the dynamic effects of RCS MCL breaks. The original design for these subcompartments envelopes the predicted subcompartment accident pressure loading from other breaks that are postulated to occur. A separate calculation to evaluate the structural effects of the new peak accident pressure on the containment internals was not performed because it is bounded by the original design basis.

Containment Subcompartment Temperature Effects

A qualitative analysis of the accident temperature effects due to RSG/PUR on the containment internal structures was performed and compared to the original BSAP computer analysis of the internal structures. The results of this comparison determined that the original BSAP analysis remains bounding for RSG/PUR. Therefore, a detailed assessment of the design margins was not performed.

This qualitative analysis compared the RSG/PUR concrete thermal profiles that were generated using the COPATTA computer code to those values assumed in the original BSAP analysis for containment internal structures. The RSG/PUR concrete thermal profiles for the containment shell and containment basemat were used to determine the thermally induced loading and interaction forces on the containment internal structures. The resultant loading of the subcompartment walls due to RSG/PUR was compared to the original BSAP analysis by reviewing the RSG/PUR results against the critical wall elements in the BSAP analysis that have high thermal load elements. The results of the comparison showed that the original BSAP analysis for the subcompartment walls remains bounding.

Containment Shell Structure

UFSAR Table 6.2.1-3 identifies the principal design parameters of the containment shell structure. (i.e., internal design pressure of 60 psig and design temperature of 300 °F). These original design parameters were compared to the predicted RSG/PUR containment accident pressure/temperature profiles, which are shown in the PURLR Figures 6.2-2 and 6.2-5, respectively. The comparison determined that, although the design basis margin for containment accident pressure is decreased by RSG/PUR, the peak calculated containment accident pressures and temperatures remain bounded by the existing containment design parameters documented in UFSAR Table 6.2.1-3 (i.e., 60 psig and 300 °F). Therefore, a separate calculation to evaluate the structural effects of the new peak accident temperature on the containment concrete was not performed.

Containment Structural Steel

UFSAR Table 6.2.1-3 identifies the principal design parameters of the "Maximum DBA" temperature for Containment as 300 °F. The Main Steam Line Break (MSLB) temperature values from UFSAR Table 6.2.1-9 were compared to the SGR/PUR values from PURLR Table 6.2-6 and Figure 6.2-9 and the temperature profiles remain bounded by the existing containment design parameters.

<u>NRC Question 12:</u>

The acceptability of several secondary system items (i.e., steam traps) relies on an improvement in the steam quality to offset the increase in steam flow associated with PUR. Steam quality is expected to go from 0.25 percent to 0.1 percent as a result of the SGR. Clarify whether the startup testing program for PUR includes a test of the steam quality. Also, clarify whether the 0.25 percent steam quality assumed for current conditions is based on measurements or design numbers, and whether any secondary system items (i.e., steam traps) are close to their operational limits at current conditions.

<u>APS Response:</u>

The 0.1% moisture carryover design value stated in Section 8.8.6 of the PURLR is a RSG performance requirement. This parameter will be measured during the post-PUR startup test program.

The 0.25% moisture carryover criteria is a design value associated with the OSGs. The Unit 2 measured moisture carryover values for the OSGs are 0.6195% and 0.4840% for SG1 and SG2, respectively. These carryover values were last measured in June of 1996.

The amount of moisture carryover while operating at PUR conditions (using the anticipated 0.1% design value and the increased PUR steam mass flow rate) is less than one quarter of the actual amount of moisture carryover at current plant operating condition.

The secondary systems capacities are not challenged at current operating levels and, based on the above discussion, should not be challenged after PUR.

NRC Question 13:

Section 9.1 of the PURLR states that a modification will be made to the main steam isolation valve bypass valve. Describe the modification.

APS Response:

The Main Steam Isolation Valve Bypass Valves (MSIVBVs) are 4 inch Anchor Darling gate valves (Model 93-15199) with Parker Hannifin air operated piston actuators (Model 14KJB2AS-13-5.250). The valves have a safety function to close on a Main Steam Isolation Signal (MSIS), and they are normally closed during full power operation of the plant.

The MSIVBVs must be modified such that they are capable of closing under the maximum, worst-case differential pressure loads. The following modifications will be made to each valve:

1. The pre-load of the actuator stanchion springs will be increased to provide increased closing thrust.
2. The air supply pressure regulator will be removed. This will allow full instrument air system pressure to be applied to the actuator to open the valve.
3. The size of the actuator air exhaust solenoid valve and exhaust piping will be increased to improve the MSIVBV's closing stroke time.