APS

10 CFR 50.90

Palo Verde Nuclear
Generating Station

David Mauldin
Vice President
Nuclear Engineering
and Support

TEL (623) 393-5553
FAX (623) 393-6077

Mail Station 7605
P O Box 52034
Phoenix, AZ 85072-2034

102-04834–CDM/TNW/RAB
August 29, 2002

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555

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 Materials and Chemical Engineering Branch. Responses to questions from the remaining branches will be submitted separately.

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

A001

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

Sincerely

Angela Krainik for David Mauldin

CDM/TNW/RAB/kg

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

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, Angela K. Krainik, represent that I am Director, Emergency Services, 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.

Angela K. Krainik

Angela K. Krainik

Sworn To Before Me This 29 Day Of August, 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

Attachment 2

NRC Materials and Chemical Engineering Branch Questions and APS Responses

NRC Question 1:

In Section 5.5 of the PURLR, the licensee stated that RSGs are being designed and analyzed in accordance with the ASME Code for structural acceptability, U-bend fatigue, tube degradation, tube plugging, and repair requirements. Discuss the following:

NRC Question 1.a:

The structures and components in the replacement SGs that were being analyzed in the structural acceptability analysis. Discuss whether each of the structures and components has satisfied the relevant ASME Code allowable stresses and fatigue usage factors. Discuss specific ASME subsections and equations used in the structural analysis.

APS Response:

The components of the Replacement Steam Generators (RSGs) that were analyzed for Power Uprate (PUR) condition are summarized in Table 1.a-1. Each component of the RSG listed in Table 1.a-1 has been analyzed and demonstrated to meet the relevant ASME Code allowable stresses and fatigue usage factors. ASME subsections used in the analyses are NB-3222, NB-3652, NB-3653, NB-3654, and NB-3656. Specific equations used in the evaluation are Equation 9, Equation 10, Equation 11, Equation 12, and Equation 13.

The resulting stresses and the Cumulative Usage Factor (CUF) are summarized in Table 1.a-1.

Table 1.a-1: 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

Table 1.a-1: Maximum Stresses and CUFs for RSG Components					
Component	Stress Category	Maximum Stress (psi) (A)	Allowable Stress (psi) (B)	Stress Ratio (A)/(B)	CUF
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
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 Feedwater (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

Table 1.a-1: Maximum Stresses and CUFs for RSG Components					
Component	Stress Category	Maximum Stress (psi) (A)	Allowable Stress (psi) (B)	Stress Ratio (A)/(B)	CUF
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
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

NRC Question 1.b:

The U-bend fatigue analysis and whether the U-bend fatigue calculation satisfies NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tube."

APS Response:

NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes, describes the circumstances surrounding a July 15, 1987 Steam Generator Tube Rupture (SGTR) event at North Anna Unit 1. The cause of the SGTR was determined to be high cycle fatigue. The most significant contributors were; (1) a reduction in damping at the tube to tube support intersection caused by denting and (2) local high velocities caused by non-uniform Anti-Vibration Bar (AVB) penetrations into the tube bundle.

The RSGs contain design and materials features that preclude denting and a proven upper support design similar to the Original Steam Generators (OSGs) that is significantly different than the AVB design in the North Anna Unit 1 Steam Generators (SGs).

Fluid Stability

As reported in Bulletin 88-02, flow induced vibration has a significant effect on tube response in cases where the fluid elastic stability ratio equals or exceeds 1.0. For the RSGs, a structural and fatigue analysis was performed. For conservatism, a maximum stability ratio of 0.7 was imposed. Each tube was analyzed at critical velocity, effective velocity and stability ratio. For all tubes the limit of 0.7 for stability ratio is satisfied. The maximum stability ratio value (0.38) is achieved for the mode at 31.7 Hz of tube Row 133 Line 40. This mode is the first out-of-plane bending mode of the bend region. Therefore, the fluid elastic stability is demonstrated.

Design and Material Features

The support material for the RSGs is A 176 Type 409 stainless steel. This material is the same as the OSGs and is not susceptible to the denting conditions experienced by the carbon steel support plates in the North Anna SGs. This position has been verified by inspection and tube pulls in the OSGs. The upper supports in the RSGs are similar to the OSGs and include a combination of eggcrates, vertical strap supports, and batwing supports. The batwings are located at every tube line and are designed to prevent out-of-plane deflection and thus preclude the deflection amplitude required for fatigue. The basic configuration of the upper bundle support design has nearly 30 years of operating experience with no evidence of fatigue issues.

Based on the aforementioned information, it is concluded that for Palo Verde the issues identified in Bulletin 88-02 are not applicable to the RSGs.

NRC Question 1.c:

The analysis to satisfy the tube plugging limit of 40 percent tube wall thickness in the plant technical specifications.

APS Response:

The PVNGS design basis for allowable tube wall thickness is derived from the margins defined in Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, Revision 0, August 1976, and the design requirements and guidance provided in ASME Section III. For computing t_{min}, a derivation of the pressure stress equation from Subsection NB-3324.1 of ASME Section III is used. The margins applied in the minimum wall calculation are in accordance with Regulatory Guide 1.121 and are defined as:

A. Loadings associated with normal plant conditions, including startup, operation in power range, hot standby, and cooldown, as well as all anticipated transients (e.g., loss of electrical load, loss of offsite power) that are included in the design specifications for the plant, should not produce a primary membrane stress in excess of the yield stress of the tube material at operating temperature.
B. The margin between the maximum internal pressure to be contained by the tubes during normal plant conditions and the pressure that would be required to burst the tubes should remain consistent with the margin incorporated in the design rules of Section III of the ASME Code.
C. Loadings associated with a Loss of Coolant Accident (LOCA) or a Steam Line Break (SLB), either inside or outside the containment and concurrent with the Safe Shutdown Earthquake (SSE), should be accommodated with the margin determined by the stress limits specified in NB-3225 of Section III of the ASME Code and by the ultimate tube burst strength determined experimentally at the operating temperature.

The limiting wall thickness was calculated in the RSG certified design report to be 0.0121 inches. This represents 32% of the nominal wall thickness (0.042 inches) or a 68% allowable degradation. Using the same standard assumptions for growth rate (10%) and eddy current uncertainty (10%), the plugging limit of 40% is conservatively justified. This limit is also consistent with ASME Section XI Subsection IWB-3521.1 that puts the allowable flaw size at 40% for tubing having a radius (0.333 inch) to thickness (r/t) ratio less than 8.70. For the PVNGS RSGs the r/t ratio is 0.333/0.042 or 7.92.

Also, the original assumptions for growth and eddy current uncertainty are based on testimony by James Knight and the Atomic Safety and Licensing Board in the matter of the Prairie Island Operating License circa 1975 and are consistent with the assumptions made in the PVNGS Technical Specification Amendment Request Sleeving Process for Steam Generator Repair – letter 102-03325-WLS/SAB/JRP dated April 18, 1995 and approved in the Palo Verde Nuclear Generating Station Units 1, 2, and 3 - Issuance of Amendments re: Steam Generator Tube Sleeving (TAC NOS. M98920, M98921 AND M98922) dated August 5, 1999.

NRC Question 1.d:

The analysis and/or tests to demonstrate the structural integrity of the SG tubes.

APS Response:

PVNGS assesses the integrity of SG tubing via technical specification requirements for inspection and the evaluation, testing, and analytical processes for condition monitoring and operational assessment as specified per plant procedures. The PVNGS procedures adopt the techniques and guidance specified in NEI 97-06, Steam Generator Program Guidelines, EPRI Guidelines (Steam Generator Integrity Assessment Guidelines, In Situ Pressure Test Guidelines), and the EPRI Flaw Assessment Handbook. These processes are currently in place for the OSGs and no changes are required for the RSGs.

The structural performance of Alloy 690TT has been demonstrated by industry analysis, testing, and operational history. No new forms of degradation with respect to morphology or physical characteristics are anticipated. No additional plant specific testing has been performed for the RSGs.

NRC Question 1.e:

How the ASME Code is used to determine tube degradation or repair requirements and how tube degradation and repair requirements are being analyzed.

APS Response:

The RSGs are designed and analyzed in accordance with the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code Section III, 1989 edition, no Addenda. A certified design specification and design report has been completed detailing the analyses performed to document compliance with the ASME Code.

Tube degradation is identified, monitored, and assessed via a Technical Specification inspection program. The program details the frequency and sampling requirements for eddy current inspections of SG tubing. The SG inspection program meets the requirements of ASME Section XI, Rules for the Inservice Inspection of Nuclear Power Plant Components, 1992 edition (including Code Cases N-356, N401-1 and N402-1) and NRC Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes, Revision 0, June 1974. Additional information with respect to the conduct and assessment of SG inspections have been provided to the NRC in response to Generic Letters 95-03, Circumferential Cracking in Steam Generator Tubes and 97-05, Steam Generator Tube Inspection Techniques.

The assessment of degraded tubing with respect to repair and tube integrity has been addressed in the response to Question 1.d.

NRC Question 2:

Discuss tests and/or analyses performed to demonstrate the corrosion resistance of Alloy 690 SG tubing under the PUR conditions.

APS Response:

With respect to the material improvement associated with Alloy 690TT material, there is a significant database of industry literature with regard to its corrosion resistant performance. First, it should be recognized that Alloy 690TT is, in reality, a third generation SG tube material. The benefit of thermal treatment was first observed with Alloy 600. There are a number of Alloy 600TT plants operating in the United States with up to 20 years operating experience with no confirmed evidence of Stress Corrosion Cracking (SCC). The improvements in both primary and secondary side corrosion resistance are considerably better with Alloy 690TT as observed by testing in the laboratory. With the exception of caustic environments containing lead, Alloy 690TT has been shown to have superior corrosion resistance over Alloy 600MA. For environments bounding the PUR conditions, improvement factors of 2-10 times have been verified in the laboratory.

Additionally, Alloy 690TT has been proven in the field to have complete resistance to Primary Water Stress Corrosion Cracking (PWSCC). For example, thousands of Alloy 690TT tube plugs have been installed with up to 10 years experience, with no PWSCC cracks observed at a wide variety of operating temperatures. Additionally up to 11 years of SG operating experience with Alloy 690TT have been accumulated with no identified SCC to date.

Finally, it should be noted with respect to corrosion resistance, the design and operating conditions associated with the PUR (e.g., temperature and pressure) are bounded by the conditions specified for the NRC approved PVNGS Technical Specification amendment request for Alloy 690TT Sleeves (Reference Letter 102-04292-JML/SAB/GAM, May 26, 1999, subsequently approved by the NRC in a letter dated August 5, 1999).

NRC Question 3:

Discuss tests and/or analyses performed to demonstrate leakage integrity of the replacement SG tubes under the PUR conditions.

APS Response:

The changes resulting from PUR with respect to leakage integrity (e.g., temperature and pressure) are similar or bounded by the OSG design basis. The assessment of leakage integrity for accident conditions does not change as a result of the RSGs or PUR.

Additionally, PVNGS employs a conservative Primary-to-Secondary Leakage (PSL) monitoring program that exceeds the Technical Specification and EPRI guidance. Plant shutdown is initiated if PSL exceeds 50 gpd.

Based on tubing material properties, there is a negligible difference in leakage resulting from the change in tubing materials, with better performance (less predicted leakage) with 690TT. Based on this, no additional testing or analysis was deemed necessary with respect to leakage integrity for RSG/PUR.

NRC Question 4:

Describe briefly the RSG. For example, provide information on (1) the model SG, (2) the nominal diameter and wall thickness of the tubes, (3) the configuration and material of the tube support, (4) U-bend support configuration, and (5) designs that would mitigate the potential for tube degradation or internal component degradation.

APS Response:

(1) The RSGs are a unique model manufactured in Italy by Ansaldo – Camozzi Energy Special Components spa. The RSGs design is based on the NRC approved Combustion Engineering System 80+ design. The RSGs are of the recirculating vertical shell and tube heat exchanger type with integral axial flow economizer and moisture separating equipment.

(2) The nominal diameter of the tube is 0.75 inch and nominal thickness is 0.042 inch.

(3) Tube supports in RSG design are a grid structure of corrosion resistant stainless steel. Except for a slight variation to the upper support configuration to address central cavity tube wear phenomena the RSGs are very similar to design that is incorporated in the NRC approved System 80+ SG design. The System 80+ design was most recently applied in the Korean Plants. Tube supports are of three basic configurations:
   a. Horizontal grids (eggcrates) that provide support to the vertical runs of tubes
   b. Vertical grids that provide vertical and horizontal support to the horizontal run of tubes in upper bend region and
   c. Diagonal strips that provide out-of-plane support to 90 degree bends

   The support material is Type 409 stainless steel, selected for its resistance to erosion/corrosion and thinning. Type 409 thermal expansion properties are compatible with the Alloy 690TT tubes.

(4) The main difference from the OSGs is that the U-bend upper bundle support structure includes welded connections between the vertical grids and the diagonal supports. Relocation of the diagonal supports to bisect tube bends, narrower perforated bend region supports, and ventilated upper tube bundle support system are features of the RSGs. The changes in the mechanical design improve the thermal/hydraulic conditions in the upper bundle region preventing crevice dry out and reducing secondary side fouling, as well as addressing tube wear phenomena evidenced in the OSG's.

(5) The eggcrate supports for the SG tubes provide excellent support and have the lowest flow resistance in the industry.

The upper tube bundle support system has three functions:

a. to support the horizontal tube spans against high velocity two phase cross flow,
b. to permit expanded vertical tube pitch (from 1.0 in. to 1.75 in.) that promotes free flow through the bend region and prevents low flow dryout regions, and
c. to support the upper tube bundle via structural beams against hypothetical accident conditions, seismic loading, transportation loads and Dead Weight (DW).

These functions collectively mitigate the potential for tube or internal component degradation.

NRC Question 5:

Because the effects of flow accelerated corrosion (FAC) on the degradation of carbon steel components are plant-specific, the NRC staff requests the licensee to provide a predictive analysis methodology that must include the values of the parameters affecting FAC, such as velocity and temperature, and the corresponding changes in component wear rates before and after the PUR. Please include predicted FAC wear rate changes in balance of plant components and those components most susceptible to FAC.

APS Response:

The plant components which are susceptible to FAC are modeled in CHECWORKS, EPRI's predictive model for FAC. The changes to the model due to the PUR will be made before the installation of RSGs and PUR. The changes will include all parameters affecting FAC, and at that time the component wear rates before and after the PUR can be compared. The model changes are expected to be minor in nature and result in insignificant changes to the plant components susceptibility to FAC. Changes to the current FAC model due to the PUR are planned to be completed by July 31, 2003.

NRC Question 6:

The NRC staff requests that the licensee indicate the degree of compliance with NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." This letter requires that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems. The licensee should describe how this program was modified to account for the PUR, If there is a generic computer code (e.g., CHECWORKS) used in predicting wall thinning by FAC, specify it; however, if the code is plant-specific to Unit 2, provide a description of the code.

APS Response:

The FAC Program fully complies with NRC Generic Letter 89-08. The program and procedures are in place to monitor and maintain the structural integrity of high-energy carbon steel systems. The FAC Program utilizes CHECWORKS in predicting wall thinning by FAC. PVNGS has a CHECWORKS model for each unit allowing the plant-specific model for Unit 2 to incorporate the PUR changes to the FAC susceptible components. Changes to the current Unit 2 FAC model due to the PUR are planned to be completed by July 31, 2003.

NRC Question 7:

The PURLR does not discuss the power-uprate-related effects on RV integrity. Discuss the effect of the PUR on the following for Unit 2: pressurized thermal shock, fluence evaluation, heat-up and cooldown pressure temperature limit curves, low temperature overpressure protection, upper shelf energy, and surveillance capsule withdrawal schedule.

APS Response:

The factors influencing Reactor Vessel (RV) integrity are the initial properties of the materials and the neutron fluence incident on the materials. PUR does not affect the initial material properties, but the neutron fluence can change. The effect of neutron fluence changes on vessel integrity is assessed below using 10 CFR Part 50, Appendices G and H, and 10 CFR Part 50.61.

a) Pressurized Thermal Shock (PTS) - The screening criteria in 10 CFR Part 50.61 is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials. The highest RT_{PTS} value for Unit 2 at the end of the current license was determined to be 78 °F for a plate from the intermediate shell course of the RV. The projected RT_{PTS} value at the end of the current license for the Unit 2 beltline materials are summarized in Figure 7-2 (from the NRC Reactor Vessel Integrity Database (RVID). These values represent conditions for PUR (3990 MW_t) given that the projected fluence at end-of-license, 3.29×10^{19} n/cm^2, E>1 MeV, is bounded by the Analysis of Record (AOR) and the method for predicting RT_{PTS} is unchanged.

b) Vessel Fluence Evaluation - The AOR end-of-life fluence is 3.29E+19 n/cm^2 for the vessel inside surface. The AOR is based on a core power level of 4200 MW_t. In 1994, WCAP-13935 (Analysis of 137 Degree Capsule from the Arizona Public Service Company Palo Verde Unit No. 2 Reactor Vessel Radiation Surveillance Program) was issued. Based on that analysis, the 32 EFPY peak azimuthal fluence is 2.047E+19 n/cm^2 for the vessel inside surface. The WCAP-13935 analysis showed that the projected end-of-life (32 EFPY) fluence was approximately one-third lower than the value in the AOR (i.e., one-third more conservative than the assessment done for the PUR submittal). The large difference between the AOR and the WCAP-13935 analysis is based on the fact that the latter did account for actual plant operation, and much of the difference is a reflection of the low leakage fuel management program employed. The PUR submittal concerning vessel fluence was based on the AOR and showed that value to be bounding.

c) Heat-up and Cool-down Pressure Temperature Limit Curves and Low Temperature Overpressure Protection - 10 CFR Part 50 Appendix G addresses the limits on pressure and temperature that are placed on heat-up and cool-down during normal operation. There are no changes to the values used to establish the Appendix G normal operating limits. The limits represent conditions for PUR (3990 MW_t) given that the projected fluence at end-of-license, 3.29×10^{19} n/cm^2, E>1 MeV, is bounded by the AOR such that the predicted vessel material properties used to establish the heat-up and cool-down limits are unchanged. The low temperature overpressure protection limits for PUR conditions are unchanged for those same reasons.

d) Upper Shelf Energy (USE) – 10 CFR Part 50 Appendix G requires that the upper shelf energy throughout the life of the vessel be no less than 50 ft-lb. For Unit 2, the lowest USE value at the end of the current license was determined to be 74 ft-lb from the lower shell course of the RV. The projected USE value at the end of the current license for the Unit 2 beltline materials is summarized in Figure 7-1 (from the NRC RVID). These values represent conditions for PUR (3990 MW_t) given the projected fluence at end-of-life, 3.29×10^{19} n/cm^2, E>1 MeV.

e) Surveillance Capsule Withdrawal Schedule - 10 CFR Part 50 Appendix H defines the RV surveillance program that is to be used by the licensee to monitor the neutron radiation induced changes in fracture toughness of the vessel during the life of the plant. It includes requirements to establish a surveillance capsule withdrawal schedule. The schedule was established based on the original calculation of fluence that was shown to bound conditions for PUR (3990 MW_t). The detailed surveillance schedule is discussed in UFSAR Section 5.3.1.6.6 and Table 5.3-19. Therefore, the existing surveillance capsule withdrawal schedule remains applicable under conditions for PUR.

Figure 7-1: Upper Shelf Energy Data for Palo Verde 2, Upper Shelf Energy Information for Vessel Materials from NRC RVID Database					
Beltline Material I.D.	Material Type	1/4T USE	1/4T Fluence	Unirr. USE	Method IUSE
Int. Shell Axial Welds, 101-124 A/C	Mil B-4, SAW	78	1.681E+19	100	Direct
Int. Shell Plate, F-765-4	A 533B-1	90	1.681E+19	114	Direct
Int. Shell Plate, F-765-5	A 533B-1	95	1.681E+19	121	Direct
Int. Shell Plate, F-765-6	A 533B-1	99	1.681E+19	126	Direct
Int./Lwr Shell Circ. Weld 101-171	Mil B-4, SAW	75	1.681E+19	95	Direct
Lower Shell Plate, F-773-1	A 533B-1	82	1.681E+19	105	Direct
Lower Shell Plate, F-773-2	A 533B-1	100	1.681E+19	127	Direct
Lower Shell Plate, F-773-3	A 533B-1	101	1.681E+19	129	Direct
Lwr Shell Axial Welds 101-142 A/C	Mil B-4, SAW	74	1.681E+19	100	Direct

Figure 7-2: NRC PTS Summary Report for Palo Verde 2 Calculated RTPTS Values for Vessel Beltline Materials from RVID Database						
Beltline Material Identification	IRTNDT (°F)	CF (°F)	End-of-Life Fluence	RTNDT (°F)	Margin (°F)	RTPTS (°F)
Int. Shell Axial Welds 101-124 A/C	-60	33.6	3.29E+19	44.1	44.08	28
Int. Shell Plate, F-765-4	-20	20	3.29E+19	26.2	26.24	32
Int. Shell Plate, F-765-5	10	20	3.29E+19	26.2	26.24	62
Int. Shell Plate, F-765-6	10	26	3.29E+19	34.1	34	78
Int./Lwr Shell Circ. Weld 101-171	-30	26.55	3.29E+19	34.8	34.83	40
Lower Shell Plate, F-773-1	10	20	3.29E+19	26.2	26.24	62
Lower Shell Plate, F-773-2	0	26	3.29E+19	34.1	34	68
Lower Shell Plate, F-773-3	-60	31	3.29E+19	40.7	34	15
Lwr Shell Axial Welds 101-142 A/C	-80	44.2	3.29E+19	58	56	34