APS

Palo Verde Nuclear
Generating Station

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10 CFR 50.90

102-05116-CDM/TNW/RAB
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, D.C. 20555

References: 1. Letter dated September 29, 2003 from B. M. Pham, USNRC to G. R. Overbeck, "Palo Verde Nuclear Generating Station, Unit 2 (PVNGS-2) – Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations"

2. 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"

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528, 50-529 and 50-530
Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations in Units 1 and 3, and Associated Administrative Changes for Unit 2

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) submits herewith a request to amend Facility Operating Licenses (OL) NPF-41, NPF-51 and NPF-74 and Appendix A, Technical Specifications (TS) for PVNGS Units 1, 2 and 3. The proposed amendments would allow operation of PVNGS Units 1 and 3 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94% above the current licensed power level of 3876 MWt. The proposed amendments would also make administrative changes to the Unit 2 Technical Specifications so that the changed pages would apply to the three PVNGS units. Operation at the uprated power level with replacement steam generators has been approved for Unit 2 (Reference 1).

These proposed amendments would revise paragraph 2.C.(1) of the Units 1 and 3 Operating Licenses. The proposed amendments would also revise Units 1 and 3 Technical Specifications 1.1, "Definitions"; 3.3.1, "Reactor Protection System (RPS)

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Instrumentation – Operating"; 3.3.2, "Reactor Protection System (RPS) Instrumentation – Shutdown"; 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"; 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; 3.7.1, "Main Steam Safety Valves (MSSVS)"; and 5.5.16, "Containment Leakage Rate Testing Program."

The proposed amendments would establish OL and TS requirements for both pre-PUR conditions (3876 MWt RTP) and post-PUR conditions (3990 MWt RTP). After PUR is implemented in all three units, OL amendments will be requested to delete the 3876 MWt RTP requirements from the OLs and TSs.

The proposed amendments are requested to improve the economic performance of PVNGS Units 1 and 3. Replacing the steam generators and increasing the rated thermal power limit of PVNGS Units 1 and 3 from 3876 MWt to 3990 MWt would result in an increase in electrical output of approximately 55 megawatts electric (MWe) in each unit.

On December 21, 2001, APS submitted Reference 2 requesting a License Amendment to support replacement of steam generators and uprated power operations for Palo Verde Unit 2. On September 29, 2003, the NRC issued Reference 1 approving the request. Attachment 6 to Reference 2, the Power Uprate Licensing Report (PULR), provided a description of the analyses and evaluations performed to demonstrate that Unit 2 would continue to operate safely with replacement steam generators at the uprated power level. Since this amendment request for Units 1 and 3 is similar to the request approved for Unit 2, and the three Palo Verde units are virtually identical, APS is referencing the PULR for Unit 2 as the basis for the analyses and evaluations for Units 1 and 3. Each section of the Units 1 and 3 PULR has been compared to the same section of the Unit 2 report as modified/clarified by Requests for Additional Information. APS has identified the differences and also noted whether or not the NRC Staff's conclusions in the Unit 2 Safety Evaluation Report would be affected when applied to Units 1 and 3. This format was discussed with the NRC Staff in a meeting held on November 18, 2003.

Table 6.4-1 of Reference 2, Attachment 6, incorrectly states the value of Kr-88 as 1.30E+07. Table 6.5-1 of Reference 2, Attachment 6 correctly states the value as 1.30E+08. APS used the correct value for Kr-88 for the dose consequences reported in Reference 2. In Reference 1, Table 1 the NRC staff states the incorrect value of Kr-88 from Table 6.4-1. If the NRC staff used this value for an independent evaluation, the results would yield lower doses than the values reported by APS. The higher dose rate in Exclusion Area Boundary (EAB)/Low Population Zone (LPZ), calculated by APS, continue to meet regulatory requirements.

The proposed TS changes include an increase in calculated peak containment internal pressure (P_a) from 52 psig to 58 psig, identical to the change requested for Unit 2 in Reference 2, and approved in Reference 1. Prior to the submittal of Reference 2, APS successfully performed the Integrated Leak Rate Test (ILRT) for the Unit 2 containment at higher P_a during the normally scheduled 10 CFR 50, Appendix J test. The ILRT at the higher P_a has not yet been performed for the Units 1 and 3 containment structures.

Therefore, APS requests a license condition to specify that the performance of the Appendix J-required ILRT using the new P_a will be due when the next ILRT is performed in accordance with the Appendix J schedule after implementation of PUR in Units 1 and 3. The last Unit 1 ILRT was performed in 1999, and the last Unit 3 ILRT was performed in 2000. Local leak rate testing will be performed at the higher P_a, prior to implementing the PUR in Units 1 and 3 requested in this submittal. Since the last tests performed in Units 1 and 3, there have been no modifications made to the containment liners, and the new steam generators will be installed through the existing equipment hatch. Please refer to Section 9.2 of Enclosure 2, Attachment 4.

The Westinghouse Electric Company (Westinghouse) requests that Attachment 5 to Enclosure 2 be withheld from public disclosure in accordance with 10 CFR 2.390. Attachment 6 to Enclosure 2 is an affidavit from Westinghouse stating the reasons that Attachment 5 should be considered as a proprietary document.

APS requests approval of these proposed amendments by June 30, 2005. Once approved, the amendments will be implemented within 120 days. After implementation of the amendments, the 3876 MWt RTP (pre-PUR) limits will continue to apply to Unit 1 through operating cycle 12 and to Unit 3 through operating cycle 13. The 3990 MWt RTP (post-PUR) limits will apply to Unit 1 after operating cycle 12, scheduled for Fall 2005 and to Unit 3 after operating cycle 13, scheduled for Fall 2007.

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

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

Sincerely,

David Mauldin

CDM/TNW/RAB

Enclosures:

1. Notarized Affidavit
2. Arizona Public Service Company's Evaluation of the Proposed Changes

Attachments:

1. Proposed Operating License and Technical Specification Changes (mark-up)
2. Proposed Operating License and Technical Specification Pages (retyped)
3. Changes to TS Bases Pages (for information only)
4. Power Uprate Licensing Report
5. Westinghouse Electric Company Proprietary Information In Support of PVNGS-1 and 3 Power Uprate Submittal (includes non-proprietary version)
6. Affidavit from the Westinghouse Electric Company Submitted Pursuant to 10 CFR 2.390 to Consider Attachment 5 as a Proprietary Document

cc:	B. S. Mallett	NRC Region IV Regional Administrator
	M. B. Fields	NRC NRR Project Manager
	N. L. Salgado	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)

ENCLOSURE 1

NOTARIZED AFFIDAVIT

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, 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.

David Mauldin
David Mauldin

Sworn To Before Me This 9th Day Of July, 2004.

Susie Lynn Ergish
Notary Public

OFFICIAL SEAL
Susie Lynn Ergish
NOTARY PUBLIC - STATE of ARIZONA
MARICOPA COUNTY
MY COMM. EXPIRES July 14, 2007

Notary Commission Stamp

ENCLOSURE 2

ARIZONA PUBLIC SERVICE COMPANY'S EVALUATION OF THE PROPOSED CHANGES

Subject: Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations in Units 1 and 3, and Administrative Changes for Unit 2

Affected Operating License Paragraph: 2.C.(1)
Technical Specification Sections: 1.1, 3.3.1, 3.3.2, 3.3.5, 3.4.1, 3.7.1 and 5.5.16

1.0 DESCRIPTION

This letter is a request from Arizona Public Service Company (APS) to amend Operating Licenses NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 respectively. The proposed amendments would allow operation of PVNGS Units 1 and 3 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94% above the current licensed power level of 3876 MWt. The proposed amendments would also make administrative changes to the Unit 2 Technical Specifications so that the changed pages would apply to the three PVNGS units. Operation at the uprated power level with replacement steam generators has been approved for Unit 2 (Reference 1).

These proposed amendments are requested to improve the economic performance of PVNGS Units 1 and 3. Increasing the rated thermal power (RTP) limit of PVNGS Units 1 and 3 from 3876 MWt to 3990 MWt would result in an increase in electrical output of approximately 55 megawatts electric (MWe) in each unit.

2.0 Proposed Changes

The proposed amendments would make the following changes.

1. Revise paragraph 2.C.(1) of the Unit 1 Facility Operating License (NPF-41) to increase the authorized 100% reactor core power (rated thermal power) from 3876 MWt to 3990 MWt, an increase of 2.94%, after operating cycle 12. The new power level of 3990 MWt represents an increase of 5% above the originally licensed power level of 3800 MWt. The increase to 3876 MWt was authorized by the NRC in a letter dated May 23, 1996, Amendment No. 108 for Unit 1 and Amendment No. 80 for Unit 3.

2. Revise paragraph 2.C.(1) of the Unit 3 Facility Operating License (NPF-74) to increase the authorized 100% reactor core power (rated thermal power) from 3876 MWt to 3990 MWt, an increase of 2.94%, after operating cycle 13. The new power level of 3990 MWt represents an increase of 5% above the originally licensed power level of 3800 MWt. The increase to 3876 MWt was authorized by the NRC in a letter dated May 23, 1996, Amendment No. 108 for Unit 1 and Amendment No. 80 for Unit 3.

3. Revise TS Section 1.1, Definition of Rated Thermal Power, for Units 1 and 3, from 3876 MWt to 3990 MWt after operating cycle 12 for Unit 1 and operating cycle 13 for Unit 3.

4. Revise Table 3.3.1-1, Reactor Protective System Instrumentation (referenced in LCO 3.3.1), item 6, Steam Generator #1 Pressure - Low and item 7, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Units licensed to operate at 3990 MWt RTP. The Table would be revised to provide the values for 3876 MWt RTP and 3990 MWt RTP. This increase in the allowable value is proportional to the increase in steam generator pressure during normal operation and

will ensure a comparable reactor protection system response. Both the power uprate and the replacement steam generators (RSGs) affect this specification.

5. Revise Table 3.3.2-1, Reactor Protective System Instrumentation - Shutdown (referenced in LCO 3.3.2), item 2, Steam Generator #1 Pressure - Low and item 3, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for units licensed to operate at 3990 MWt RTP. The Table would be revised to indicate the values for 3876 MWt RTP and 3990 MWt RTP. This increase in the allowable value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable reactor protection system response. Both the power uprate and the RSGs affect this specification.

6. Revise Table 3.3.5-1, Engineered Safety Features Actuation System Instrumentation (referenced in LCO 3.3.5), item 4.a, Steam Generator #1 Pressure - Low and item 4.b, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for units licensed to operate at 3990 MWt RTP. The Table would be revised to indicate the values for 3876 MWt RTP and 3990 MWt RTP. This increase in the allowable value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable engineered safety features system response. Both the power uprate and the RSGs affect this specification.

7. Revise Figure 3.4.1-1 (Page 1 of 2 and Page 2 of 2), Reactor Coolant Cold Leg Temperature vs. Core Power Level, to change the upper limit in the area of acceptable operation for units licensed to operate at 3990 MWt RTP. Page 1 of 2 would apply to units operating at 3876 MWt RTP, and page 2 of 2 would apply to units operating at 3990 MWt RTP. The new upper limit line would allow a cold leg temperature of 570 °F at 0% power, decreasing linearly to 564 °F at 100% power. The upper limit line of Figure 3.4.1-1, in the current TS, decreases linearly from 570 °F at 0% power to 568 °F at 30% power. At 30% power the current figure then decreases linearly from 568 °F to 560 °F at 100% power. The increase in cold leg temperature at 100% power will allow a more optimum main steam pressure for turbine operation. Both the power uprate and RSGs affect this specification.

8. Revise Table 3.7.1-1, Variable Overpower Trip (VOPT) Setpoint versus Operable Main Steam Safety Valves for units licensed to operate at 3990MWt RTP, to decrease the Maximum Power and the Maximum Allowable VOPT Setpoint when the Minimum Number of Main Steam Safety Valves (MSSVs) per Steam Generator Required Operable is less than ten. Columns currently labeled Units 1 and 3 would be labeled 3876 MWT RTP, and columns currently labeled Unit 2 would be labeled 3990 MWT RTP. The reduction in allowable power levels and VOPT setpoints for Units 1 and 3 are required to offset the impacts of increased core power and increased cold leg temperature on the consequences of the UFSAR Chapter 15 design basis events. The power uprate affects this specification.

9. Revise TS 5.5.16, Containment Leakage Rate Testing Program, to increase the peak calculated containment internal pressure for the design basis loss of coolant

accident (P_a) for units licensed to operate at 3990 MWt RTP from 52.0 psig to 58.0 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig. The calculated peak containment pressure remains below the containment internal design pressure of 60.0 psig. Both the power uprate and the RSGs affect this specification.

The bases for TSs 3.6.1, 3.6.2, 3.6.4 and 3.6.6 would also be revised to reflect these changes and are included in Attachment 3 of this submittal.

3.0 BACKGROUND

The proposed amendments are requested to improve the economic performance of PVNGS Units 1 and 3. Increasing the rated thermal power limit of PVNGS Units 1 and 3 from 3876 MWt RTP to 3990 MWt RTP would result in an increase in electrical output of approximately 55 megawatts electric (MWe) in each unit.

The original full power operating licenses for Unit 1, issued in June 1985 and for Unit 3, issued in November 1987, authorized a rated thermal power (RTP) of 3800 MWt. In May 1996, the NRC issued Amendment Nos. 108, 100 and 80 to Units 1, 2 and 3, respectively, to increase the authorized RTP to 3876 MWt. This amendment request to increase RTP to 3990 MWt would be a 2.94% increase above that authorized in Amendments 108 and 80, and represents a 5% increase from the original RTP.

On September 29, 2003, the NRC approved a similar amendment request for PVNGS Unit 2 to operate at 3990 MWt RTP with replacement steam generators (Reference 1).

3.1 System Description

Paragraph 2.C.(1) of the Facility Operating Licenses specifies, as a license condition, the maximum reactor core thermal power level at which the unit is authorized to operate. The maximum authorized reactor thermal power level is specified as a license condition in order to limit thermal power to the value used in the safety analyses. The maximum reactor core thermal power specified in the operating license is also known as the rated thermal power (RTP). Regulatory Guide 1.49 recommends a 2% uncertainty be included in the power level used in the safety analysis, as appropriate. Thus, the safety analysis supporting this amendment uses a reactor core thermal power of 4070 MWt, which is 102% of 3990 MWt, the proposed new RTP. The definition of Rated Thermal Power in TS 1.1 identifies the licensed limit of the total reactor core heat transfer rate to the reactor coolant.

LCO 3.3.1, Reactor Protective System Instrumentation - Operating and Table 3.3.1-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 6 and 7 in Table

3.3.1-1) provides protection against an excessive rate of heat extraction from the steam generators and the resulting rapid, uncontrolled cooldown of the Reactor Coolant System (RCS). This trip is needed to shut down the reactor and assist the Engineered Safety Features (ESF) system in the event of a Main Steam Line Break (MSLB) or Main Feedwater Line Break (MFWLB) accident. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.[1]

LCO 3.3.2, Reactor Protective System Instrumentation - Shutdown and Table 3.3.2-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 2 and 3 in Table 3.3.2-1) provides shutdown margin to prevent or minimize the return to power following a large MSLB in Mode 3.[2]

LCO 3.3.5, Engineered Safety Features Actuation System Instrumentation and Table 3.3.5-1, which it references specify the required number of channels operable for each reactor trip function, the applicable modes for each function, and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low signal actuates a MSIS to prevent an excessive rate of heat extraction and subsequent cooldown of the RCS in the event of a MSLB or MFWLB.[3]

Figure 3.4.1-1, Reactor Coolant Cold Leg temperature vs. Core Power Level, referenced in LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, provides parametric limits to ensure that the actual value of the reactor coolant cold leg temperature is maintained within the range of values used in the safety analysis. The safety analysis supporting this requested amendment uses the proposed new allowable cold leg temperature range (550 °F to 570 °F), and this proposed change maintains the basis for the cold leg temperature limits.[4]

Table 3.7.1-1, Variable Overpower Trip Setpoint versus Operable Main Steam Safety Valves, referenced in LCO 3.7.1, Main Steam Safety Valves (MSSVs), specifies maximum power levels and overpower reactor trip setpoints for specified numbers of OPERABLE MSSVs. An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power in accordance with Table 3.7.1-1. These reduced power levels, derived from the transient analysis, compensate for degraded relieving capacity and ensure that the results of the transient analysis are acceptable.[5]

Technical Specification 5.5.16, Containment Leakage Rate Testing Program, provides the requirements for the Containment Leakage Rate Testing Program. The calculated

[1] TS Bases B.3.3.1, Applicable Safety Analysis
[2] TS Bases 3.3.2, Applicable Safety Analysis
[3] TS Bases 3.3.5, Applicable Safety Analysis
[4] TS Bases 3.4.1, Background
[5] TS Bases 3.7.1, Actions

peak containment internal pressure for the design basis LOCA (P_a) is the basis for the containment leakage rate in the testing program.[6]

3.2 Need for the Proposed Changes

In March 1993 PVNGS Unit 2 experienced a Steam Generator (SG) tube rupture. The primary contributor to the event was determined to be IGA/IGSCC which occurred as a result of tube-to-tube crevice formation. The SG tubes in the Unit 2 SGs, as well as the tubes in the Units 1 and 3 SGs were manufactured using Alloy 600. Since elevated temperature is a contributing factor of IGA/IGSCC, APS reduced the RCS operating temperature and derated the three PVNGS units. Although RCS temperature was reduced, SG tube degradation continued to occur. Based on the continued degradation, APS decided to replace the SGs in Unit 2 and to also request a license amendment to increase power coincident with the replacement of the SGs (Reference 2). The request was approved by the NRC (Reference 1), allowing APS to replace the SGs with new ones containing better materials. The new SGs should be more reliable, and will allow APS to regain the power lost when RCS temperature was reduced, as well increase the economic performance of the unit.

The tubes in the Units 1 and 3 SGs have also continued to degrade, and the SGs will be replaced. APS is requesting similar amendments for Units 1 and 3. The reliability and economic performance of Units 1 and 3 will be enhanced with the implementation of SG replacement and power uprate.

4.0 TECHNICAL ANALYSIS

Refer to Attachment 4 (Power Uprate Licensing Report).

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

APS has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

a. Evaluation of the Probability of Previously Evaluated Accidents

[6] 10 CFR 50, Appendix J, Option B

Plant Structures, Systems and Components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, a small number of minor modifications will be made prior to implementation of uprated power operations so that surveillance test acceptance criteria continues to be met. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current technical specification (TS) surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated within current operating requirements at uprated conditions. Therefore, no new structure, system or component interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

b. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (i.e., both LOCA and non-LOCA accidents) previously analyzed in the UFSAR. The analysis showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remain either unchanged or have not significantly increased due to operation at uprated power conditions. The radiological consequences of all DBAs continue to meet established regulatory limits.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The configuration, operation and accident response of the PVNGS Units 1 and 3 structures, systems, and components are unchanged by operation at uprated power conditions or by the associated proposed TS changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident or different scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing SSCs. The basic design function of all SSCs remains unchanged and no new equipment or systems have been installed that could potentially introduce new failure modes or accident sequences.

Based on this analysis, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not have an adverse effect on any safety-related system or design basis function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

A comprehensive analysis was performed to evaluate the effects of power uprate on PVNGS Units 1 and 3. This analysis identified and defined the major input parameters to the NSSS, reviewed NSSS design transients, and reviewed the capabilities of the NSSS and BOP fluid systems, NSSS/BOP interfaces, NSSS and BOP control systems, and NSSS and BOP SSCs. All appropriate NSSS accident analyses were re-performed to confirm that acceptable results were maintained and that the radiological consequences remained within regulatory and Standard Review Plan (SRP) limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analyses confirmed that all NSSS and BOP SSCs are capable, some with minor modifications, to safely support operations at uprated power conditions.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicates that the calculated peak containment pressure (P_a) increases from 52.0 psig to 58.0 psig, but remains less than the containment internal design pressure of 60 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig.

Radiological consequences of the following accidents were reviewed: Main Steam Line Break, Locked Reactor Coolant Pump (RCP) Rotor, CEA Ejection, Small Steam Line Break Outside Containment, Steam Generator Tube Rupture, LBLOCA, SBLOCA, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents did not show a significant change due to uprated power conditions and 10 CFR 100 and SRP limits continue to be met.

The analyses supporting operation at power uprate conditions have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

Based upon the above, APS concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

On December 21, 2001, APS requested a license amendment to support replacement of steam generators and uprated power operations in PVNGS Unit 2 (Reference 2). The NRC issued Amendment 149 on September 29, 2003 (Reference 1) approving the request, and concluded that the facility will operate in conformity with the application, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission.

The request for Units 1 and 3 is similar to the request for Unit 2, and uses the same justification for NRC approval of this request. If approved, Units 1 and 3 will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission.

6.0 ENVIRONMENTAL CONSIDERATION

APS has determined that the proposed amendment does not involve an unreviewed environmental question, in accordance with Section 3.1 of Appendix B of the Technical Specifications. A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or (2) a significant change in the effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) above, which may have a significant adverse environmental impact. Based on the following, this amendment request does not constitute an unreviewed environmental question:

1) A matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board.

APS reviewed the FES and determined that this amendment request does not significantly increase any adverse environmental impact. The plant is not being modified in any way which would significantly increase or change the type of effluents currently produced. The 2.94% increase in RTP is within the 4100 MWt design stretch power evaluated in the FES - Construction Permit Stage (FES-CP). Thus the environmental effects as a result of the uprate are bounded by those previously evaluated during FES-CP phase.

Radiological releases are controlled in accordance with PVNGS Offsite Dose Calculation Manual and the results are reported annually to the NRC. Design Basis Event radiological releases have been demonstrated, in the safety analysis provided with this amendment request, to not significantly increase offsite exposure and remain within regulatory limits. The radiological exposure to plant workers is controlled under the PVNGS As Low As Reasonably Achievable (ALARA) Program and will not significantly change.

2) A significant change in the effluents or power level.

A 2.94% increase in RTP is not a significant increase in power level. The Final Environmental Statement (NUREG 0841) recognized in the Summary and Conclusions Section that the maximum design thermal output for each unit is 4100 MWt. The proposed increase is less than the FES-CP evaluated maximum design thermal output of the units. Thus the environmental effects previously evaluated for land and water usage are bounded by those previously evaluated. The increase in RTP does not change any of the conclusions of NUREG 0841.

Effluents as discussed above will not be significantly increased and are controlled by PVNGS programs and applicable regulations.

3) A matter not previously reviewed and evaluated in the documents specified in (1) above which may have a significant adverse environmental impact.

The increase in RTP does not change the processes, plant equipment, types of effluents, or significantly affect operation of the units. The changes are within the design basis of the NSSS and BOP SSCs at the increased RTP conditions. Safety analyses of design basis events affected by the increase have been reviewed or reanalyzed and the consequences found to be bounded by current UFSAR consequences or within regulatory requirements. The FES-CP, FES-OL, and NUREG-0841 all evaluated the environmental impact assuming the maximum design thermal output of 4100 MWt for each unit. Thus the proposed increase in rated thermal power is within the scope of the previous reviews performed to assess the environmental impact associated with the operation of each unit.

Based on the above, no unreviewed environmental question exists concerning this amendment request for increased RTP and associated TS changes.

7.0 References

1. Letter dated September 29, 2003 from B. M. Pham, USNRC to G. R. Overbeck, "Palo Verde Nuclear Generating Station, Unit 2 (PVNGS-2) – Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations" (Amendment 149) ML032720538 & ML032730666

2. 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"
ML013650362 and ML013650419

ATTACHMENT 1

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATION (mark-up)

Marked-up Operating License and Technical Specification Pages
(marked-up on current TS pages unless otherwise noted)

<u>Unit 1 Operating License</u>

Page 4

<u>Unit 3 Operating License</u>

Page 4

<u>Technical Specifications</u>

Page 1.1-6
Page 3.3.1-8
Page 3.3.1-17
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Page 3.4.1-4
Page 3.7.1-3
Page 3.7.1-4 (marked-up on page submitted in letter 102-05043, dated February 4, 2004 requesting revision concerning MSSVs)
Page 5.5-24

PVNGS Unit 1 Operating License Page 4
Marked-up with Proposed Power Uprate Changes

(6)(a) Pursuant to an Order of the Nuclear Regulatory Commission dated December 12, 1985, the Public Service Company of New Mexico (PNM) was authorized to transfer a portion of its ownership share in Palo Verde, Unit 1 to certain institutional investors on December 31, 1985, and at the same time has leased back from such purchasers the same interest in the Palo Verde, Unit 1 facility. The term of the lease is to January 15, 2015, subject to a right of renewal. Additional sale and leaseback transactions (for a term expiring on January 15, 2015) of all or a portion of PNM's remaining ownership share in Palo Verde Unit 1 are hereby authorized until June 30, 1987. Any such sale and leaseback transaction is subject to the representations and conditions set forth in the aforementioned applications of October 19, 1985, February 5, 1986, October 16, 1986 and November 26, 1986, and the Commission's Order of December 12, 1985, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 1. For purposes of this condition, the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for Palo Verde, Unit 1 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 1 as specified in license counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3876 megawatts thermal (100% power)<u> through operating cycle 12, and 3990 megawatts thermal (100% power) after operating cycle 12,</u> in accordance with the conditions specified herein.

PVNGS Unit 3 Operating License Page 4
Marked-up with Proposed Power Uprate Changes

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3876 megawatts thermal (100% power) **through operating cycle 13, and 3990 megawatts thermal (100% power) after operating cycle 13,** in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(5) Additional Conditions

Deleted

D. APS has previously been granted an exemption from Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50. This exemption was previously granted in Facility Operating License NPF-65 pursuant to 10 CFR 50.12.

With the granting of this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensees shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Safeguard Contingency Plan is incorporated into the Physical Security Plan. The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Palo

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt for Unit~~s~~ 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 3990 MWt for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full strength CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length or part strength CEA position.

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	3[a], 4[a], 5[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia
3. Steam Generator #2 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure - High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure - Low[(a)]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure - High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure - High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure - Low[(a)]		≥ 1821 psia
4. Main Steam Isolation Signal[(c)]		
a. Steam Generator #1 Pressure-Low[(b)]	1,2,3	~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia
b. Steam Generator #2 Pressure-Low[(b)]		~~Units 1 and 3~~ 3876 MWt RTP: ≥ 890 psia ~~Unit 2~~ 3990 MWt RTP: ≥ 955 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level-Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level-Low	1,2,3	≥ 25.3%
b. SG Pressure Difference-High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level-Low	1,2,3	≥ 25.3%
b. SG Pressure Difference-High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

Figure 3.4.1-1, (Page 1 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

~~Units 1 and 3~~ 3876 MWt RTP

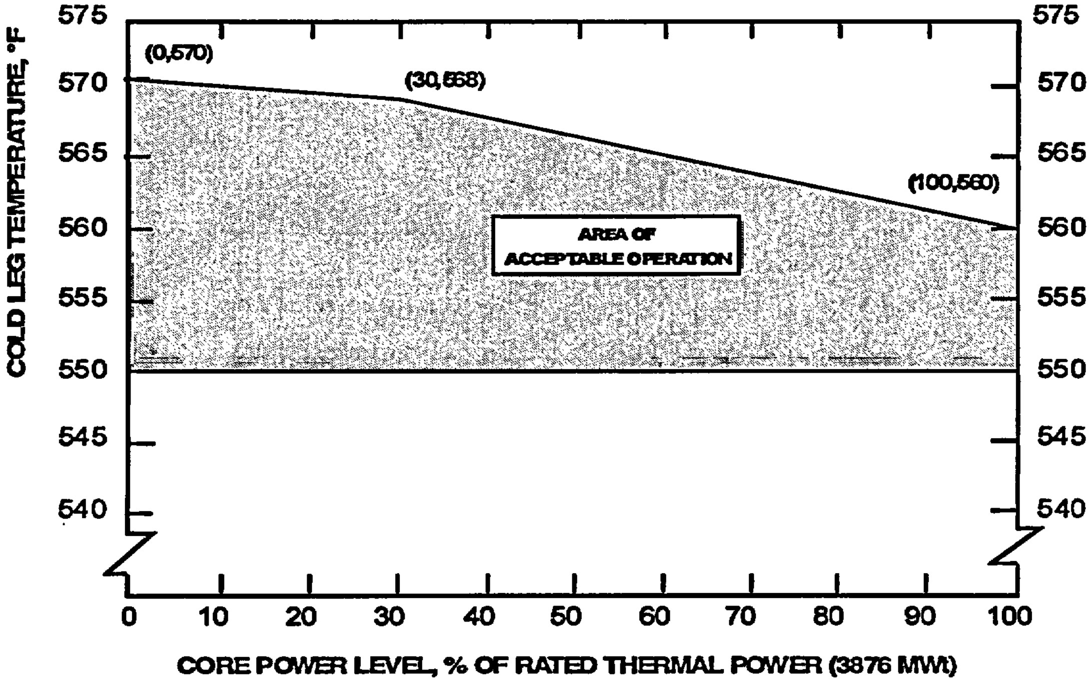

Figure 3.4.1-1, (Page 2 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

~~Unit 2~~ 3990 MWt RTP

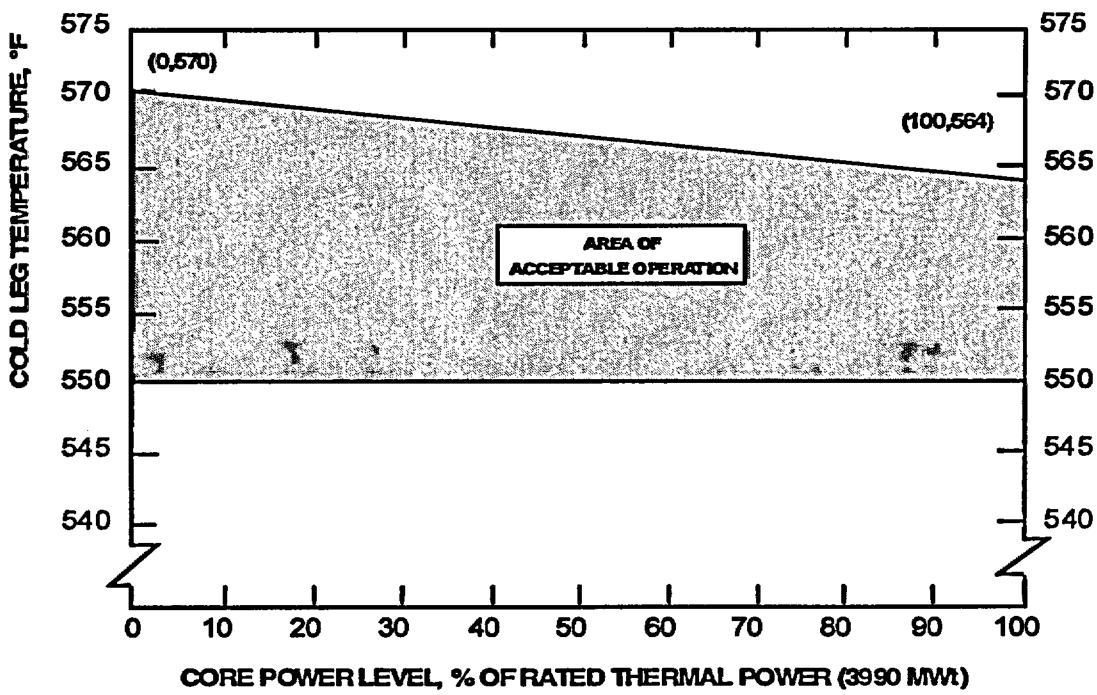

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	~~Units 1 and 3~~ 3876 MWt RTP	~~Unit 2~~ 3990 MWt RTP	~~Units 1 and 3~~ 3876 MWt RTP	~~Unit 2~~ 3990 MWt RTP
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM POWER (% RTP) or HIGHEST MODE		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT[a] (% RTP)	
		~~Units 1 and 3~~ 3876 MWt RTP	~~Unit 2~~ 3990 MWt RTP	~~Units 1 and 3~~ 3876 MWt RTP	~~Unit 2~~ 3990MWt RTP
10	0	100.0	100.0	111.0	111.0
9	1	98.2	90.0	108.0	99.7
8	2	87.3	80.0	97.1	89.7
7	3	76.4	68.0	86.2	77.7
6	4	65.5	56.0	75.3	65.7
5	5	MODE 3	MODE 3	NA	NA
4	6	MODE 3	MODE 3	NA	NA
3	7	MODE 3	MODE 3	NA	NA
2	8	MODE 3	MODE 3	NA	NA

(a) The VOPT setpoint is not required to be reset in MODE 3.

NOTE: This mark-up is on a page containing a proposed MSSV amendment that was requested in letter no. 102-05043, dated February 4, 2004.

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Unit~~s~~ 1 **through operating cycle 12** and **Unit 3 through operating cycle 13**, and 58.0 psig for **Unit 1 after operating cycle 12**, Unit 2**, and Unit 3 after operating cycle 13**. The containment design pressure is 60 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

ATTACHMENT 2

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES (retyped)

Attachment 2

Retyped Operating License and Technical Specification Pages

Unit 1 Operating License

Page 4

Unit 3 Operating License

Page 4

Technical Specifications

Page 1.1-6
Page 3.3.1-8
Page 3.3.1-17
Page 3.3.2-5
Page 3.3.5-4
Page 3.4.1-3
Page 3.4.1-4
Page 3.7.1-3
Page 5.5-24

PVNGS Unit 1 Operating License Page 4
Retyped with Proposed Power Uprate Changes

(6)(a) Pursuant to an Order of the Nuclear Regulatory Commission dated December 12, 1985, the Public Service Company of New Mexico (PNM) was authorized to transfer a portion of its ownership share in Palo Verde, Unit 1 to certain institutional investors on December 31, 1985, and at the same time has leased back from such purchasers the same interest in the Palo Verde, Unit 1 facility. The term of the lease is to January 15, 2015, subject to a right of renewal. Additional sale and leaseback transactions (for a term expiring on January 15, 2015) of all or a portion of PNM's remaining ownership share in Palo Verde Unit 1 are hereby authorized until June 30, 1987. Any such sale and leaseback transaction is subject to the representations and conditions set forth in the aforementioned applications of October 19, 1985, February 5, 1986, October 16, 1986 and November 26, 1986, and the Commission's Order of December 12, 1985, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 1. For purposes of this condition, the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for Palo Verde, Unit 1 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 1 as specified in license counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3876 megawatts thermal (100% power) through operating cycle 12, and 3990 megawatts thermal (100% power) after operating cycle 12, in accordance with the conditions specified herein.

PVNGS Unit 3 Operating License Page 4
Retyped with Proposed Power Uprate Changes

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3876 megawatts thermal (100% power) through operating cycle 13, and 3990 megawatts thermal (100% power) after operating cycle 13, in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This license is subject to the antitrust conditions delineated in Appendix C to this license.

(4) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(5) Additional Conditions

Deleted

D. APS has previously been granted an exemption from Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50. This exemption was previously granted in Facility Operating License NPF-65 pursuant to 10 CFR 50.12.

With the granting of this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensees shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Safeguard Contingency Plan is incorporated into the Physical Security Plan. The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Palo

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 3990 MWt for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13.

REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All full strength CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and

b. There is no change in part length or part strength CEA position.

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level - High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia
7. Steam Generator #2 Pressure - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
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7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High(d)	3(a), 4(a), 5(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP(c)
2. Steam Generator #1 Pressure-Low(b)	3(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia
3. Steam Generator #2 Pressure-Low(b)	3(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
4. Main Steam Isolation Signal[c]		
a. Steam Generator #1 Pressure–Low[b]	1,2,3	3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia
b. Steam Generator #2 Pressure–Low[b]		3876 MWt RTP: ≥ 890 psia 3990 MWt RTP: ≥ 955 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level–Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

Figure 3.4.1-1, (Page 1 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

3876 MWt RTP

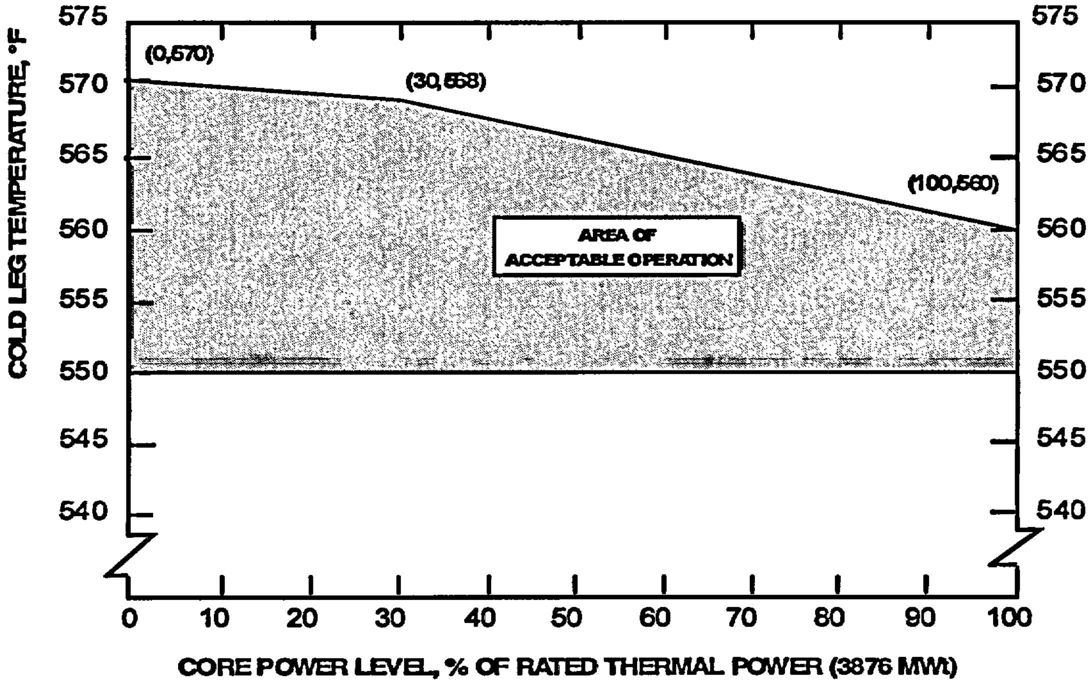

Figure 3.4.1-1, (Page 2 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

3990 MWt RTP

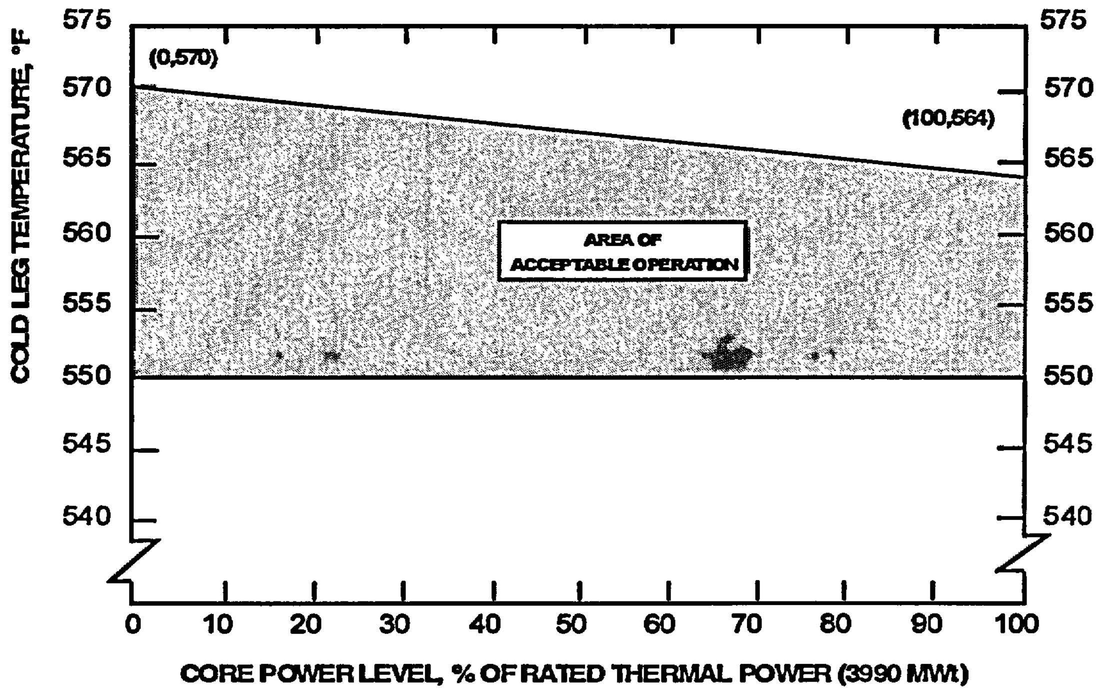

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	3876 MWt RTP	3990 MWt RTP	3876 MWt RTP	3990 MWt RTP
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Unit 1 through operating cycle 12 and Unit 3 through operating cycle 13, and 58.0 psig for Unit 1 after operating cycle 12, Unit 2, and Unit 3 after operating cycle 13. The containment design pressure is 60 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

ATTACHMENT 3

CHANGES TO TS BASES PAGES
(for information only)

Attachment 3

Associated Changes to Technical Specification Bases

Bases

Page B 3.6.1-2
Page B 3.6.2-2
Page B 3.6.4-1
Page B 3.6.6-3

BASES (continued)

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of 52.0 psig for units operating at 3876 MWt RTP ~~1 and 3~~, 58.0 psig for units operating at 3990 MWt RTP ~~2~~, which results from the limiting design basis LOCA.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a [52.0 psig for units **operating at 3876 MWt RTP** ~~1 and 3~~, and 58.0 psig for units **operating at 3990 MWt RTP** ~~2~~], following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. A double ended discharge line break LOCA with maximum ECCS results in the highest calculated internal containment pressure of 52.0 psig for units **operating at 3876 MWt RTP**~~ 1 and 3~~, and 58.0 psig for units **operating at 3990 MWt RTP**~~ 2~~, which is below the internal design pressure of 60 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) Systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis bounds the containment pressure allowed during normal operation. The LCO limit of 2.5 psig ensures that, in the event of an accident, the maximum peak containment internal pressure, 52.0 psig for units **operating at 3876 MWt RTP**~~ 1 and 3~~, and 58.0 psig for units **operating at 3990**

(continued)

MWt RTP~~2~~, and the maximum accident design pressure for containment, 60 psig, are not exceeded.

(continued)

BASES

BACKGROUND (continued)

The Containment Spray System accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

APPLICABLE SAFETY ANALYSES

The Containment Spray System limits the temperature and pressure that could be experienced following a DBA. The Containment Spray System is required to be capable of reducing containment pressure to 1/2 the peak pressure within 24 hours following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the Loss Of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 52.0 psig for units operating at 3876 MWt RTP ~~1 and 3~~, and 58.0 psig for units operating at 3990 MWt RTP ~~2~~ (experienced during a LOCA). The analysis shows that the peak containment vapor temperature is 405.65°F (experienced during a MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 102% RTP, one containment spray train operating, and initial (pre-accident) conditions of 120°F and 16.7 psia (LOCA) and 13.22 psia (MSLB). The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to -2.6 psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specification 3.6.4.

(continued)

ATTACHMENT 4

POWER UPRATE LICENSING REPORT

Arizona Public Service Company
Palo Verde Nuclear Generating Station

Power Uprate Licensing Report for the Palo Verde Nuclear Generating Station Units 1 and 3

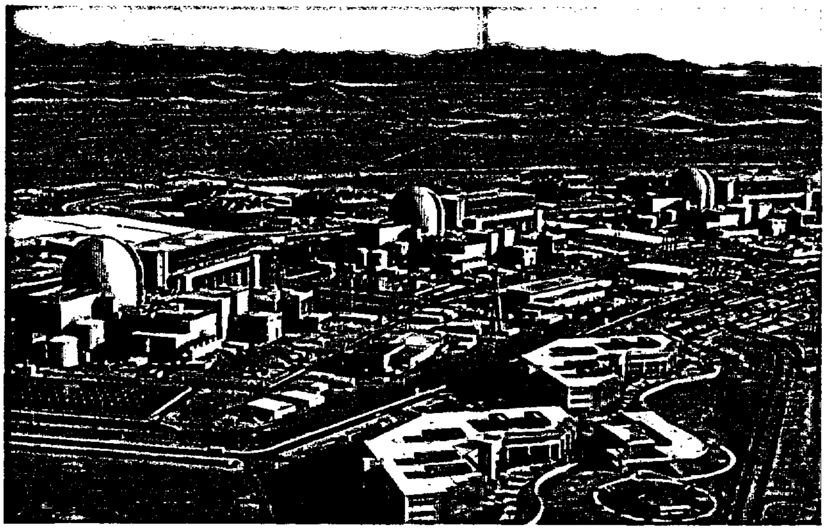

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TABLE OF TABLES

Note the above TABLE OF TABLES reflects only the tables that have been added or revised from the Unit 2 Power Uprate Licensing Report (PURLR). The Tables from the Unit 2 PURLR remain valid, Table 6.3-1 lists the Non-LOCA event assessment for Units 1 and 3 versus Table 6.3-1 provided the Unit 2 event assessment. Table 6.3-2 and Table 6.3-3 are added to replace Table 6.3-47 and Table 6.3-48 to discuss Double-Ended Break of the Letdown Line Outside Containment Upstream of the letdown line control valve (DBLLOCUS) reanalysis.

The TABLE OF FIGURES, from the Unit 2 PURLR remains valid.

POWER UPRATE LICENSING REPORT EXECUTIVE SUMMARY

The purpose of this license amendment request is to increase the electrical output (MW_e) of the Palo Verde Nuclear Generating Station (PVNGS) Units 1 and 3. This will be accomplished by requesting NRC approval to increase the licensed 100% reactor core power level from 3876 MW_t to 3990 MW_t, a 2.94% increase.

In support of the Power Uprate (PUR), Arizona Public Service (APS), the operator of PVNGS, and Westinghouse Electric Corporation (WEC) have performed analyses and evaluations for the Nuclear Steam Supply System (NSSS). These analyses demonstrate that APS complies with applicable licensing criteria and design requirements at the uprated reactor power of 3990 MW_t. The scope of the analyses and evaluations included the:

- NSSS and containment performance parameters,
    - design transients (used in stress analysis),
    - Structures, Systems, and Components (SSCs),
    - Design Basis Accidents (DBAs),
    - nuclear fuel design, and
- secondary side Balance of Plant (BOP).

PVNGS consists of three virtually identical units of Combustion Engineering System 80™ Pressurized Water Reactors (PWRs). Each unit consists of an independent reactor containment, ultimate heat sink, and turbine; auxiliary, fuel, radwaste, control/corridor, diesel generator, main steam support structure, and operations support buildings.

APS submitted a license amendment request to increase the rated thermal power and electrical output of PVNGS Unit 2 (Reference ES-1), and the NRC issued License Amendment 149 for Unit 2 (Reference ES-2). As discussed with the NRC in a meeting on November 18, 2003, this report provides a section-by-section summary of differences between the PVNGS Unit 2 Power Uprate Licensing Report (PURLR) and the proposed condition for Units 1 and 3. In addition, this report contains a new Section 9.2. This section provides justification for the APS request for a license condition to specify that the performance of the Appendix J-required Integrated Leak Rate Test (ILRT) using the new P_a will be due when the next ILRT is performed in accordance with the Appendix J schedule after implementation of PUR in Units 1 and 3.

The summary of differences includes references to APS responses to the NRC's request for additional information during the review and approval of the PUR license amendment for Unit 2. These references supplement the information provided in the Unit 2 PURLR. A summary for the APS letters in response to the NRC's requests for additional information and clarifications are presented in Table ES-1. In addition, the current docketed UFSAR (Reference ES-3) has not been fully updated to reflect the PUR condition of Unit 2. This PURLR considers the unincorporated changes to the UFSAR as well as all pending licensing actions.

Table ES-1
APS Letters in Response to the NRC's Requests for Additional Information and Clarifications Provided in Support of PUR
(Page 1 of 2)

Reference	Reference Number	Subject
Reference ES-4	102-04664	NRC Electrical and Instrumentation and Controls Branch Questions and APS Responses Summary of the PVNGS I & C Design Guide for Instrument Uncertainty and Setpoint Determination
Reference ES-5	102-04828	Plant Systems Branch Questions and APS Responses
Reference ES-6	102-04834	Materials an Chemical Engineering Branch Questions and APS Responses
Reference ES-7	102-04835	Probabilistic Safety Assessment Branch Questions and APS Responses
Reference ES-8	102-04837	Mechanical and Civil Engineering Branch Questions and APS Responses (contains Proprietary information)
Reference ES-9	102-04847	Reactor Systems Branch Questions and APS Responses Tables 31.a-1 through 31.a-9 SER Limitations/Constraints Associated with the LBLOCA and SBLOCA Evaluation Models Used for the PUR ECCS Performance Analysis and NRC Question 35 on the Description of the Long Term Cooling, Boron Precipitation Model (contains Proprietary information) Replacement pages for Section 6.3.0.3.1 Pressurizer Safety Valve Orifice Sizing Correction Factor
Reference ES-10	102-04866	Feedwater Line Break with Loss of Offsite Power - Long Term Cooling Event [(1)]
Reference ES-11	102-04872	Revised Response to Question 31.d from Reactor Systems Branch
Reference ES-12	102-04877	Additional Information for Question 1.b from Mechanical and Civil Engineering Branch
Reference ES-13	102-04899	Clarification of Responses to the Request for Additional Information from the Reactor Systems Branch
Reference ES-14	102-04936	Results of Review for Proprietary Information in Draft Safety Evaluation Report (SER)
Reference ES-15	102-04954	Results of Review for Factual Accuracy in Draft SER

Table ES-1
APS Letters in Response to the NRC's Requests for Additional Information and Clarifications Provided in Support of PUR
(Page 2 of 2)

Reference	Reference Number	Subject
Reference ES-16	102-04974	Response to Additional information Requested During the Meeting Held on June 26, 2003 Revised Comment 13 to Draft Safety Evaluation Report, Section 4.1 Additional information Requested in July 17, 2003 Conference Call
Reference ES-17	102-04989	Additional information Regarding Cladding Oxidation for Zircaloy-Clad Fuel

Note 1: The Feedwater Line Break (FWLB) with Loss of Offsite Power (LOP) - long term cooling event letter assumes that the plant is operated on program T_{ave}, and the Pressurizer Level Control System (PLCS) is in the automatic mode at the beginning of the event. This provided an additional methods/assumptions change in addition to those reported in Reference ES-1.

In addition, APS has reviewed processed/pending and approved any licensing actions that may impact this PUR. These actions are summarized below:

Table ES-2
Licensing Actions that Impact PUR
(Page 1 of 2)

Reference	Reference Number	Subject
Reference ES-18	102-04699	10 CFR 50.46 Thirty-Day Report for Changes to Palo Verde Nuclear Generating Station Units 1, 2, and 3 ECCS Performance Analysis for ZIRLO™
Reference ES-19	102-04700	APS' Response to the Requested Information requested by the NRC Regarding Reactor Vessel Material Surveillance Capsule Reports (TAC No. MB0396)
Reference ES-20	102-04836	Request for a License Amendment to Revise the Peak Linear Heat Rate Safety Limit, Technical Specification 2.1.l.2
Reference ES-21	102-04990	Request for Facility Licensing Amendment – Internal Fuel Pin Pressure Criteria for FHA Safety Analysis
Reference ES-22	102-05018	180-Day Response to NRC Generic Letter 2003-01: Control Room Habitability, dated December 5, 2003

Table ES-2
Licensing Actions that Impact PUR
(Page 2 of 2)

Reference	Reference Number	Subject
Reference ES-23	102-05043	Request for Amendment to Technical Specification 3.7.1, Main Steam Safety Valves
Reference ES-24	Technical Specification Amendment No. 136	Issuance of Amendment Re: Eliminate the Requirements for the Post Accident Sampling System (PASS) using the Consolidated Line Item Improvement Process (CLIP)
Reference ES-25	Technical Specification Amendment No. 140	Issuance of Amendment Re: Technical Specification 5.6.5b, Core Operating Limits Report (COLR) and use of ZIRLO™ Cladding Material
Reference ES-26	Technical Specification Amendment No. 145	Issuance of Amendment Re: Peak Fuel Centerline Temperature Safety Limit
Reference ES-27	Technical Specification Amendment No. 150	Issuance of Amendment Re: Core Protection Calculator System Upgrade
Reference ES-28	Technical Specification Amendment No. 152	Issuance of Amendment Re: Replacement of Part-Length Control Element Assemblies

The reference to the specific questions from the letters and the licensing actions provided in Tables ES-1 and Table ES-2 are identified in the appropriate sections of this report.

The focus of this report is on providing the information required by the NRC to approve the PUR for the PVNGS Units 1 and 3. As in Unit 2, APS is replacing Steam Generators (SGs) with larger generators in Units 1 and 3. The design and installation of the Replacement Steam Generators (RSGs) is being conducted under the provisions of 10 CFR 50.59. Evaluations and analyses supporting this PUR assume the installation of the RSGs.

The results of the engineering analyses and evaluations demonstrate that PVNGS Units 1 and 3 can safely operate at the increased rated thermal power and those applicable licensing criteria and requirements are satisfied. The evaluations and conclusions reached in this report do not change from the conclusions reached in the PURLR for

Unit 2. The SER issued for Unit 2 (Reference ES-2) is not affected by the changes/differences identified is this PUR request.

Executive Summary References

Reference ES-1	APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.
Reference ES-2	NRC letter to APS Unit 2 (PVNGS-2)-Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.
Reference ES-3	Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 12, August 2003.
Reference ES-4	APS letter 102-04664 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 13, 2002.
Reference ES-5	APS letter 102-04828 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 27, 2002.
Reference ES-6	APS letter 102-04834 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 29, 2002.
Reference ES-7	APS letter 102-04835 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 4, 2002.
Reference ES-8	APS letter 102-04837 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 6, 2002.
Reference ES-9	APS Letter 102-04847 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated October 11, 2002.

Reference ES-10 APS letter 102-04866 to the NRC, Supplement to Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated November 21, 2001.

Reference ES-11 APS letter 102-04872 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated December 10, 2002.

Reference ES-12 APS letter 102-04877 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated December 23, 2002.

Reference ES-13 APS letter 102-04899 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 11, 2003.

Reference ES-14 APS letter 102-04936 to the NRC, Results of Review for Proprietary Information in Draft Safety Evaluation Report (SER), dated May 1, 2003.

Reference ES-15 APS letter 102-04954 to the NRC, Results of Review for Factual Accuracy in Draft Safety Evaluation Report (SER), dated June 10, 2003.

Reference ES-16 APS letter 102-04974 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, July 25, 2003.

Reference ES-17 APS letter 102-04989 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, August 22, 2003.

Reference ES-18 APS letter 102-04699 to the NRC, 10 CFR 50.46(a)(3)(ii) 30-Day Report for Changes in LOCA/ECCS Performance Evaluation Models, May 3, 2002.

Reference ES-19 APS letter 102-04700 to the NRC, APS' Response to the Information Requested Regarding Reactor Vessel Material Surveillance Capsule Reports, dated May 8, 2002.

Reference ES-20 APS letter 102-04836 to the NRC, Request for a License Amendment to Revise the Peak Linear Heat Rate Safety Limit, Technical Specification 2.1.1.2, dated September 6, 2002.

Reference ES-21 APS letter 102-04990 to the NRC, Request for Facility Operating License Amendment – Internal Fuel Pin Pressure Criteria for Fuel Handling Accident Safety Analysis, dated August 22, 2003.

Reference ES-22 APS letter 102-05018 to the NRC, 180-Day Response to NRC Generic Letter 2003-01: Control Room Habitability, dated December 5, 2003.

Reference ES-23 APS letter 102-05043 to the NRC, Request for Amendment to Technical Specification 3.7.1, Main Steam Safety Valves, dated February 4, 2004.

Reference ES-24 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments Re: Eliminate the Requirements for the Post Accident Sampling System (PASS) using the Consolidated Line Item Improvement Process (CLIP) (TAC Nos. MB2291, MB2292, and MB2293), dated September 28, 2001.

Reference ES-25 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments RE: Technical Specification 5.6.5b, Core Operating Limits Report (COLR) and use of ZIRLO™ Cladding Material, (TAC Nos. MB3373, MB3374, and MB3375), dated March 12, 2002.

Reference ES-26 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on Peak Fuel Centerline Temperature Safety Limit (TAC Nos. MB6328, MB6329, and MB6330), dated December 2, 2002.

Reference ES-27 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on the Core Protection Calculator System Upgrade (TAC Nos. MB6726, MB6727, and MB6728), dated October 24, 2003.

Reference ES-28 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on Replacement of Part-Length Control Element Assemblies (TAC Nos. MC0870, MC0871, and MC0872), dated March 23, 2003.

Section 1 INTRODUCTION

Section 1.1 Purpose and Scope

Arizona Public Service (APS), Westinghouse Electric Corporation (WEC), and Ansaldo – Camozzi Energy Special Components (the Steam Generator (SG) fabricator) performed the various analyses/evaluations for the Power Uprate (PUR) for Unit 2. The scope included the Nuclear Steam Supply System (NSSS) and secondary side Balance of Plant (BOP) performance parameters. The analyses included design transients (used in stress analyses); safety analyses; Structures, Systems, and Component (SSCs) evaluations; and the response of nuclear fuel to the PUR condition. For Units 1 and 3, those analyses/evaluations were evaluated for applicability to those units. The majority of the evaluation/analyses for Unit 2 PUR were determined to be applicable to, and/or bounding for, Units 1 and 3, making reanalysis unnecessary. The analyses that were found not applicable to Units 1 and 3 were reanalyzed and the results are provided in the appropriate sections of this report.

APS has reviewed license amendments approved by the NRC since the issuance of License Amendment 149 (Reference 1-1). In addition, APS has reviewed submittals currently under review by the NRC that were submitted prior to this request. Those licensing action requests that were found applicable to this PUR are discussed in the appropriate sections of this report.

In addition, APS has reviewed all pending changes to the UFSAR. As stated in the Executive Summary, the current UFSAR (Reference 1-3) has not been updated to reflect the PUR condition in Unit 2. This PURLR considers the unincorporated changes to the UFSAR as well as all pending licensing actions.

Section 1.2 Methodology and Acceptance Criteria

Unless noted, the analyses were performed using methodologies that have been previously approved by the NRC, including those methodology changes that were approved for Unit 2 (Reference 1-1). The remaining information presented in this section as contained in Attachment 6 of Reference 1-2 is unchanged and applicable to Units 1 and 3.

Section 1.3 Technical Basis for No Significant Hazards Consideration Determination

This report provides the technical basis for the No Significant Hazards Consideration Determination associated with this license amendment request.

Section 1.4 Regulatory Guide Compliance

This PUR does not deviate from the regulatory guide compliance as listed in UFSAR Section 1.8 as modified by the pending license amendment request submitted to the NRC by Reference 1-4.

Section 1.5 Conclusions

The analyses and evaluations conclude that Units 1 and 3 can operate within licensed parameters at the PUR conditions.

Section 1.6 References

This reference section as presented in Reference 1-2, Attachment 6, Section 1.6, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 1-1 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 1-2 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 1-3 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 12, August 2003.

Reference 1-4 APS letter 102-04990 to the NRC, Request for Facility Operating License Amendment – Internal Fuel Pin Pressure Criteria for Fuel Handling Accident Safety Analysis, dated August 22, 2003.

Section 2 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

This section as contained in Reference 2-1, Attachment 6, is unchanged and applicable to Units 1 and 3.

Section 2.1 Performance Parameters

Section 2.1.1 Introduction and Background

There are no changes to this section.

Section 2.1.2 Input Parameters and Assumptions

There are no changes to this section.

Section 2.1.3 Acceptance Criteria for Determination of Parameters

There are no changes to this section.

Section 2.1.4 Discussion of Parameters

There are no changes to this section.

Section 2.2 References

This reference section as presented in Reference 2-1, Attachment 6, Section 2.2, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 2-1 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Section 3 DESIGN TRANSIENTS

This section as contained in Reference 3-1, Attachment 6, is unchanged and applicable to Units 1 and 3.

Section 3.1 Nuclear Steam Supply System Design Transients

Section 3.1.1 Introduction and Background

There are no changes to this section.

Section 3.1.2 Input Parameters and Assumptions

There are no changes to this section.

Section 3.1.3 Description of Analyses and Evaluation

There are no changes to this section.

Section 3.1.4 Results and Conclusions

There are no changes to this section.

Section 3.2 Non-Nuclear Steam Supply System Design Transients

Section 3.2.1 Introduction and Background

There are no changes to this section.

Section 3.2.2 Input Parameters and Assumptions

There are no changes to this section.

Section 3.2.3 Description of Analyses and Evaluation

There are no changes to this section.

Section 3.2.4 Results and Conclusions

There are no changes to this section.

Section 3.3 References

This reference section as presented in Reference 3-1, Attachment 6, Section 3.3, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 3-1 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Section 4 NUCLEAR STEAM SUPPLY SYSTEM

This section as contained in Reference 4-1, Attachment 6, is unchanged and applicable to Units 1 and 3.

Section 4.1 Nuclear Steam Supply System Fluid Systems

There are no changes to this section.

Section 4.1.1 Reactor Coolant System

There are no changes to this section.

Section 4.1.2 Chemical and Volume Control System

There are no changes to this section.

Section 4.1.3 Emergency Core Cooling System

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the Safety Evaluation Report (SER) issued for Unit 2 in Section 2.1 of Reference 4-2.

Section 4.1.4 Residual Heat Removal System

There are no changes to this section.

Section 4.1.5 Containment Heat Removal System

The section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 13 and 14 of Attachment 2, Reference 4-3. The plant modification described in Section 9.1 (Reference 4-1, Attachment 6) will be performed in Units 1 and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 2.2 of Reference 4-2.

Section 4.2 Nuclear Steam Supply System/Balance of Plant Fluid Systems Interfaces

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2, Reference 4-4.

Section 4.2.1 Main Steam System

There are no changes to this section.

Section 4.2.1.1 Main Steam Isolation Valves

There are no changes to this section.

Section 4.2.1.2 Main Steam Isolation Valve Bypass Valves

There are no changes to this section. The plant modification described in Section 9.1 (Reference 4-1, Attachment 6) will be performed for Units 1 and 3.

Section 4.2.1.3 Main Steam Safety Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1 of Attachment 2, Reference 4-3. A change to Technical Specification 3.7.1 is requested in this license amendment request submittal, as for Unit 2, to change the maximum allowable percent power level with one or more Main Steam Safety Valves (MSSVs) inoperable.

In addition to the changes above, Reference 4-5 has been submitted to the NRC for approval. Reference 4-5 proposes an amendment to the MSSV Technical Specification to permit operation in Mode 3 with five to eight inoperable MSSVs. The technical evaluation provided in Reference 4-5 for the proposed changes uses a thermal power of 3990 MW_t, which bounds this PUR.

Section 4.2.1.4 Atmospheric Dump Valves

There are no changes to this section.

Section 4.2.2 Condensate and Feedwater System

There are no changes to this section.

Section 4.2.2.1 Containment Feedwater Line Isolation Valves

There are no changes to this section.

Section 4.2.2.2 Condensate and Feedwater System Pumps

There are no changes to this section.

Section 4.2.2.3 Condensate and Feedwater System Conclusions

There are no changes to this section.

Section 4.2.3 Auxiliary Feedwater System and Condensate Storage Tank

There are no changes to this section.

Section 4.2.3.1 Auxiliary Feedwater System and Condensate Storage Tank Conclusions

There are no changes to this section.

Section 4.2.4 Secondary Chemistry and Steam Generator Blowdown Systems

There are no changes to this section.

Section 4.3 Instrumentation and Controls

Section 4.3.1 Introduction

This section is applicable to Units 1 and 3 as augmented by Attachment 3 of Reference 4-4. To address concerns associated with Instrument Society of America (ISA) recommended practice (Reference 4-6), instrument setpoint and uncertainty calculations demonstrate that the Allowable Value (AV) provides a large enough allowance with respect to the Analytical Limit (AL) to account for those uncertainties not measured during testing.

Section 4.3.2 Reactor Protection System

There are no changes to this section. The plant modification to change the low SG pressure trip setpoint described in Section 4.3.2 and Section 9.1 (Reference 4-1) will be performed in Units 1 and 3.

Section 4.3.3 Engineered Safety Feature Systems

There are no changes to this section.

Section 4.3.3.1 Balance of Plant Engineered Safety Feature Actuation Signal Setpoints and Regulatory Guide 1.97 Instrumentation

There are no changes to this section.

Section 4.3.3.2 Reactor Trip System/Nuclear Steam Supply System Engineered Safety Feature Actuation System Setpoints

There are no changes to this section. The plant modification to change the low SG pressure trip setpoint described in Section 4.3.2 and Section 9.1 (Reference 4-1) will be performed in Units 1 and 3.

Section 4.3.4 Systems Required For Safe Shutdown

There are no changes to this section.

Section 4.3.5 Safety-Related Display Instrumentation

There are no changes to this section.

Section 4.3.6 All Other Instrumentation Systems Required For Safety

There are no changes to this section.

Section 4.3.7 Control Systems Not Required for Safety

Section 4.3.7.1 Reactor Regulating System

There are no changes to this section.

Section 4.3.7.2 Pressurizer Pressure Control System

There are no changes to this section.

Section 4.3.7.3 Pressurizer Level Control System

There are no changes to this section.

Section 4.3.7.4 Digital Feedwater Control System

There are no changes to this section.

Section 4.3.7.4.1 Steam Generator Water Level Control System

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 3 of Attachment 2, Reference 4-4.

Section 4.3.7.5 Steam Bypass Control System

There are no changes to this section.

Section 4.3.7.6 Reactor Power Cutback System

There are no changes to this section.

Section 4.3.7.7 Boron Control System

There are no changes to this section.

Section 4.3.7.8 Loose Parts Monitoring System

There are no changes to this section.

Section 4.3.7.9 In-Core Instrumentation System

There are no changes to this section.

Section 4.3.7.10 Excore Neutron Flux Monitoring System (Non-Safety Channels)

There are no changes to this section.

Section 4.3.7.11 Boron Dilution Alarm System

There are no changes to this section.

Section 4.4 References

This reference section as presented in Reference 4-1, Attachment 6, Section 4.4, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 4-1	APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.
Reference 4-2	NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.
Reference 4-3	APS letter 102-04828 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 27, 2002.
Reference 4-4	APS letter 102-04664 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 13, 2002.
Reference 4-5	APS letter 102-05043 to the NRC, Request for Amendment to Technical Specification 3.7.1, Main Steam Safety Valves, dated February 4, 2004.
Reference 4-6	ISA-RP 67.04-2000 (equivalent to ISA-RP 67.04, Part II, 1994), Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.

Section 5 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

There is no change to this section that would affect the evaluation conducted by the NRC staff in the Safety Evaluation Report (SER) issued for Unit 2 in Section 3 of Reference 5-3. The specific Unit 2 information contained in the SER remains bounding for Units 1 and 3.

Section 5.1 Structural Evaluations of the Reactor Coolant System

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3 of Reference 5-3.

Section 5.1.1 Reactor Vessel Structural Evaluation

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.1 of Reference 5-3.

Section 5.1.1.1 Closure Head Flange Region

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.1.1.2 Reactor Vessel Inlet and Outlet Nozzles

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a, 1.b, and 2 of Attachment 2 and Enclosure 1 of Reference 5-4 and NRC Question 1.b of Attachment 2 of Reference 5-6.

Section 5.1.1.3 Reactor Vessel Nozzle Supports

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a and 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.1.1.4 Control Element Drive Mechanism Nozzles

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a, 1.b, and 5 of Attachment 2 and Enclosure 1, Reference 5-4 with the following amendment:

The as- built Outside Diameter (OD) of the Unit 3 Control Element Drive Mechanism (CEDM) nozzles is 4.275", compared to 4.050" for Units 1 and 2. The stiffer nozzle design results in lower seismic loads, but somewhat higher Loss-of-Coolant Accident (LOCA) loads. The calculated Faulted Loads of Unit 3 remain well below the Design Loads for the 4.050" OD nozzle design.

Section 5.1.1.5 In-Core Instrumentation Nozzles

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 3 of Attachment 2 and Enclosure 1, Reference 5-4.

Section 5.1.1.6 Reactor Vessel Support Columns

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.1.2 Reactor Vessel Integrity

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.1 and 3.1.1 of Reference 5-3.

Section 5.2 Reactor Vessel Internals

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.2 of Reference 5-3.

Section 5.2.1 Thermal/Hydraulic System Evaluations

Section 5.2.1.1 System Pressure Losses

There are no changes to this section.

Section 5.2.1.2 Core Bypass Flow Analysis

There are no changes to this section.

Section 5.2.1.3 Hydraulic Lift Forces

There are no changes to this section.

Section 5.2.1.4 Reactor Trip Performance Evaluation

There are no changes to this section.

Section 5.2.1.5 Control Element Assembly Structural Integrity

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2 and Enclosure 1 of Reference 5-4. The part-length, part-strength Control Element Assemblies (CEAs) are being modified per Reference 5-5. This new design is structurally similar and has been approved for use at 3990 MW_t.

Section 5.2.2 Mechanical System Evaluation

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a, 1.b, 7, and 8 of Attachment 2 and Enclosure 1 of Reference 5-4, NRC Question 1.b of Attachment 2 of Reference 5-6, and NRC Question 1.b of Attachment 2, Reference 5-7.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.2 of Reference 5-3.

Section 5.2.2.1 Loss-of-Coolant Loads

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.2.2.2 Flow Induced Vibrations

There are no changes to this section.

Section 5.2.3 Structural Evaluation of Reactor Vessel Internal Components

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 4 of Attachment 2 and Enclosure 1 of Reference 5-4. Tables 4-1 and 4-2 of the response to NRC Question 4 have been revised to reflect unit specific design changes, such as the different manufacturing processes used to manufacture the Unit 1 Upper Guide Structure (UGS) head flange and Unit 1 modifications that are associated the Unit 1 Precritical Vibration Monitoring Program (PVMP). The revised tables are provided in Table 5.2-1 and Table 5.2-3 for Unit 1 and Table 5.2-2 and Table 5.2-4 for Unit 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.2 of Reference 5-3.

Section 5.2.3.1 Introduction

There are no changes to this section.

Section 5.2.3.2 Methodology Used for the Reactor Vessel Internals Structural Evaluations

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 4 of Attachment 2 and Enclosure 1 of Reference 5-4. Tables 4-1 and 4-2 of the response to NRC Question 4 have been revised to reflect unit specific design changes, such as the different manufacturing processes used to manufacture the Unit 1 UGS head flange and Unit 1 modifications that are associated the Unit 1 PVMP. The revised tables for Unit 1 and 3 Reactor Vessel Internals (RVIs) stress summary for Replacement Steam Generators (RSGs)/Power Uprate (PUR) conditions are provided in Table 5.2-1 and Table 5.2-3 for Unit 1 and Table 5.2-2 and Table 5.2-4 for Unit 3.

Section 5.2.3.3 Additional Components

There are no changes to this section.

Section 5.2.3.4 Summary of Conclusions for Reactor Vessel Internal Components

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.2 of Reference 5-3.

Table 5.2-1
Unit 1 RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition
(Page 1 of 3)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (1)	CUF (6) (1.0 Allowable)
Core Support Barrel (CSB) Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	4,838 13,860 28,961	16,100 24,150 48,300	0.475
	Cylinders	P_m $P_m + P_b$ $P_m + P_b + Q$	5,278 9,214 22,128	16,100 24,150 48,300	< 0.475
	Lower Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	2,779 10,036 23,239	16,100 24,150 48,300	< 0.475
	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	< 0.475
	CSB to LSS Flexure Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	2,059 4,328 10,214	16,100 24,150 48,300	< 0.475

Table 5.2-1
Unit 1 RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition
(Page 2 of 3)

Major Assembly	Component	Stress Category [5]	Maximum Stress (psi)	Allowable (psi) [1]	CUF (1.0 Allowable)
Lower Support Structure (LSS) Components	Insert Pin	P_m $P_m + P_b$	2,520 11,514	43,300 64,950	< 0.074
	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	0.074
	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	< 0.074
	Cylinder	P_m $P_m + P_b$ $P_m + P_b + Q$	6,104 8,633 27,788	14,490 21,735 43,470	< 0.074
	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	< 0.074
UGS Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	7,045 23,718 31,770	16,100 24,150 48,300	0.258
	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	< 0.258
	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 FAP Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	421 4,166 4,166	12,075 18,113 36,225	0

Table 5.2-1
Unit 1 RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition
(Page 3 of 3)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (1)	CUF (1.0 Allowable)
UGS Components	Guide Tube @ PVMP Hole	P_m $P_m + P_b$ $P_m + P_b + Q$	1,277 17,016 30,383	16,100 24,150 48,300	< 0.003
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 5.2-2
Unit 3 RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition
(Page 1 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (1)	CUF (1.0 Allowable)
CSB Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	4,838 13,860 28,961	16,100 24,150 48,300	0.411
	Cylinders	P_m $P_m + P_b$ $P_m + P_b + Q$	5,356 9,418 21,694	16,100 24,150 48,300	< 0.411
	Lower Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	2,779 10,036 23,239	16,100 24,150 48,300	< 0.411
	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	< 0.411
	CSB to LSS Flexure Weld	P_m $P_m + P_b$ $P_m + P_b + Q$	2,779 3,896 17,875	14,490 21,735 43,470	< 0.411
LSS Components	Insert Pin	P_m $P_m + P_b$	2,520 11,377	43,300 64,950	< 0.075

Table 5.2-2
Unit 3 RVI Stress Summary for RSG and PUR - Normal Operation plus Upset Design Condition
(Page 2 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (1)	CUF (1.0 Allowable)
LSS Components	Main Support Beam	P_m $P_m + P_b$ $P_m + P_b + Q$	4,535 6,017 27,181	14,490 21,735 43,470	< 0.075
	Cylinder	P_m $P_m + P_b$ $P_m + P_b + Q$	6,104 8,633 27,788	14,490 21,735 43,470	< 0.075
	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	< 0.075
UGS Components	Upper Flange	P_m $P_m + P_b$ $P_m + P_b + Q$	9,160 19,557 27,821	16,100 24,150 48,300	0.125
	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	< 0.125
	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
UGS Components	Guide Tube to 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 $P_m + P_b$ $P_m + P_b + Q$	684 14,412 40,226	16,100 24,150 48,300	0.161
	Fuel Alignment Plate	P_m $P_m + P_b$ $P_m + P_b + Q$	715 14,599 25,508	16,100 24,150 48,300	0.918
	Guide Tube Extension	P_m $P_m + P_b$ $P_m + P_b + Q$	518 12,698 12,698	16,100 24,150 48,300	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 5.2-3
Unit 1 RVI Stress Summary for RSG and PUR - Faulted Design Condition
(Page 1 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
CSB Components	Upper Flange	P_m $P_m + P_b$	15,530 53,952	38,640 57,960
	Cylinders	P_m $P_m + P_b$	32,800 42,814	38,640 57,960
	Lower Flange	P_m $P_m + P_b$	14,637 34,178	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 $P_m + P_b$	14,637 30,761	38,690 57,960
LSS Components	Insert Pin	P_m $P_m + P_b$	6,152 28,019	91,000 136,500
	Main Beam to Short Beam Weld	P_m $P_m + P_b$	21,994 46,124	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 $P_m + P_b$	24.146 50,416	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
	Fuel Alignment Plate	P_m $P_m + P_b$	2,418 44,891	38,640 57,960

Table 5.2-3
Unit 1 RVI Stress Summary for RSG and PUR - Faulted Design Condition
(Page 2 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
UGS Components	Tube to FAP Weld	P_m $P_m + P_b$	451 5,076	28,980 43,470
	Guide Tube @ PVMP Hole	P_m $P_m + P_b$	5,385 15,478	38,640 57,960
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)

Table 5.2-4
Unit 3 RVI Stress Summary for RSG and PUR - Faulted Design Condition
(Page 1 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
CSB Components	Upper Flange	P_m $P_m + P_b$	15,528 53,952	38,640 57,960
	Cylinders	P_m $P_m + P_b$	31,506 43,703	38,640 57,960
	Lower Flange	P_m $P_m + P_b$	14,637 34,178	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 $P_m + P_b$	14,637 30,761	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 $P_m + P_b$	21,994 46,124	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

Table 5.2-4
Unit 3 RVI Stress Summary for RSG and PUR - Faulted Design Condition
(Page 2 of 2)

Major Assembly	Component	Stress Category (5)	Maximum Stress (psi)	Allowable (psi) (2)
UGS Components	Upper Flange	P_m $P_m + P_b$	30,796 43,268	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 $P_m + P_b$	2,331 47,587	38,640 57,960
	Fuel Alignment Plate	P_m $P_m + P_b$	2,451 47,181	38,640 57,960
	Tube to FAP Weld	P_m $P_m + P_b$	541 15,504	38,640 57,960
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 Table 5.2-1, Table 5.2-2, Table 5.2-3, and Table 5.2-4:

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
6) CUF – Cumulative Usage Factor

Section 5.3 Additional Reactor Coolant System Items

There are no changes to this section.

Section 5.3.1 Control Element Drive Mechanisms

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2 and Enclosure 1 of Reference 5-4 with the following amendment:

> The Unit 3 CEDM nozzles with 4.275" OD are heavier than the Unit 1 and 2 nozzles with 4.050" OD. The stiffer nozzle shifts the first three natural frequencies slightly for Unit 3. The resultant dynamic Operational Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), and Branch Line Pipe Break (BLPB) loads, stresses, and deflections along the CEDM structures are less for Unit 3 than for Units 1 and 2. Consequently, the Unit 2 CEDM loads, stresses, and deflections are applicable to Unit 1 and bound the Unit 3 CEDM loads, stresses, and deflections.

The part-length, part-strength CEAs are being modified per Reference 5-5. This new design is structurally similar and has been approved for use at 3990 MW_t.

Section 5.3.1.1 Control Element Drive Mechanism Evaluations

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.3.1.2 Evaluation of Control Element Drive Mechanism Deflections

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2 and Enclosure 1 of Reference 5-4, with the exception that maximum CEDM top deflections for Unit 3 are less than those for Units 1 and 2.

Section 5.3.1.3 Ability to Trip the Reactor

There are no changes to this section.

Section 5.3.1.4 Reed Switch Position Transmitter Operability

There are no changes to this section.

Section 5.3.1.5 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.3 of Reference 5-3.

Section 5.3.2 Heated Junction Thermocouple Cables and Flange

There are no changes to this section.

Section 5.3.2.1 Heated Junction Thermocouple Cables Evaluation

This section is applicable to Units 1 and 3 with the following amendment:

> The Reactor Vessel Level Monitoring System (RVLMS) contains the Heated Junction Thermocouple Cables (HJTCs). The RVLMS head area nozzles of Unit 3 are heavier (4.275" OD) than those of Units 1 and 2 (4.050" OD). This results in an increase of the first two natural frequencies from 8 and 81 Hz to 9.2 and 85 Hz. Analysis of the Unit 3 RVLMS structure resulted in lower seismic loads but slightly higher faulted loads. These loads, however, were still well within the allowables of the 4.050" OD nozzle design. Dynamic deflections at the Unit 2 flange elevation are lower than those for Units 1 and 2. Consequently, use of the Unit 2 analysis is conservative.

Section 5.3.2.2 Heated Junction Thermocouple Instrumentation Flange Assembly

This section is applicable to Units 1 and 3 as augmented by Enclosure 1 of Reference 5-4 with the following amendment:

> The Unit 3 HJTC (RVLMS) nozzles are also stiffer. In Unit 3, the seismic loads are less than those for Units 1 and 2. The controlling Unit 3 faulted bending moment is slightly higher than that for Units 1 and 2, but remains well below the HJTC flange design value.

Section 5.3.2.3 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.3 of Reference 5-3.

Section 5.3.3 In-Core Instrumentation Tubes

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 3 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.3.3.1 Operating Basis Earthquake Evaluation

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 3 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.3.3.2 Safe Shutdown Earthquake Evaluation

There are no changes to this section.

Section 5.3.3.3 Branch Line Pipe Break Evaluation

There are no changes to this section.

Section 5.3.3.4 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.3 of Reference 5-3.

Section 5.3.4 Head Lift Rig

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.3 of Reference 5-3.

Section 5.4 Reactor Coolant Loop Major Components and Component Supports

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a, 1.b, and 2 of Attachment 2 and Enclosure 1 of Reference 5-4 and NRC Question 1.b of Attachment 2 of Reference 5-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.4 of Reference 5-3.

Section 5.4.1 Reactor Coolant System - Leak-Before-Break

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 1.a and 1.b of Attachment 2 and Enclosure 1 of Reference 5-4 and NRC Question 1.b of Attachment 2 of Reference 5-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.4.1 of Reference 5-3.

Section 5.4.2 Use of ANSYS Computer Code

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.0 of Reference 5-3.

Section 5.4.3 Reactor Coolant Model Changes

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.4 Reactor Coolant System Main Loop Piping and Tributary Nozzles

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.4.1 Main Loop Piping

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.4.1.1 Non-Faulted Conditions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.4.1.2 Faulted Conditions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.4.2 Tributary Lines and Nozzles

The design/routing of Unit 2 tributary piping and nozzles is similar to Units 1 and 3. A qualitative assessment has been conducted to ensure the Units 1 and 3 ASME Class 1 analyses are bounded by the AOR.

Section 5.4.4.2.1 Tributary Piping

There are no changes to this section.

Section 5.4.4.2.2 Safety Injection Nozzles

There are no changes to this section.

Section 5.4.4.2.3 Surge Line Nozzle

There are no changes to this section.

Section 5.4.4.2.4 Charging Nozzle

There are no changes to this section.

Section 5.4.4.2.5 Letdown/Drain Nozzles

There are no changes to this section.

Section 5.4.4.2.6 Shutdown Cooling Nozzles

There are no changes to this section.

Section 5.4.4.2.7 Spray Nozzles

There are no changes to this section.

Section 5.4.4.2.8 Partial Penetration Nozzles

There are no changes to this section.

Section 5.4.5 Reactor Coolant Pumps

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.3 of Reference 5-3.

Section 5.4.5.1 Reactor Coolant Pump Structural Evaluations

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.4.5.2 Reactor Coolant Pump Motor Structural Evaluations

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.5 Steam Generators

Due to minor manufacturing differences, the Palo Verde Unit 1 and 3 RSG design differs for material designation and by less than 1% in total weight and center of gravity from the Unit 2 RSG design. The Unit 2 primary and secondary nozzles were cast. The new RSGs will have forged nozzles. These slight differences have minimal impact on the analysis/evaluations for Units 1 and 3 PUR/RSG design.

In addition, this section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 1.a, 1.b, 1.c, 1.d, 1.e, 3, and 4 Attachment 2 of Reference 5-7 and NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Sections 3.5 and 3.5.1 of Reference 5-3.

Section 5.5.1 Steam Generator Supports

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1.a Attachment 2, Reference 5-7 and NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.5.1.1 Steam Generator Upper Supports

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1.a Attachment 2, Reference 5-7 and NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.5.1.2 Steam Generator Sliding Base and Skirt Studs

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1.a Attachment 2, Reference 5-7 and NRC Question 1.a of Attachment 2 and Enclosure 1 of Reference 5-4.

Section 5.5.2 Computer Codes Used in Steam Generator Structural Analysis

There are no changes to this section.

Section 5.6 Pressurizer

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.6 of Reference 5-3.

Section 5.7 Nuclear Steam Supply System Auxiliary Equipment

There are no changes to this section.

Section 5.8 Alloy 600 Material Evaluation

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 Attachment 2, Reference 5-7.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 3.5.1 of Reference 5-3.

Section 5.9 References

This reference section as presented in Reference 5-2, Attachment 6, Section 5.9, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 5-1 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 12, August 2003.

Reference 5-2 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 5-3 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 5-4 APS letter 102-04837 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 6, 2002.

Reference 5-5 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on Replacement of Part-Length Control Element Assemblies (TAC Nos. MC0870, MC0871, and MC0872), dated March 23, 2003.

Reference 5-6 APS letter 102-04877 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated December 23, 2002.

Reference 5-7 APS letter 102-04834 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 29, 2002.

Section 6 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSIS

Section 6.1 Emergency Core Cooling System Performance Analysis

The existing PVNGS Emergency Core Cooling System (ECCS) performance analysis of record (AOR) is applicable to Units 1 and 3 operating with Replacement Steam Generators (RSGs) at the uprated core power of 3990 MW_t (4070 MW_t after including a 2% power measurement uncertainty).

The Large Break Loss-of-Coolant Accident (LBLOCA) and the Small Break Loss-of-Coolant Accident (SBLOCA) parts of the existing ECCS performance analysis were performed after the submittal of the Unit 2 Power Uprate Licensing Report (PURLR) (Reference 6-1), and consequently, were not described in that submittal. However, Sections 6.1.2 and 6.1.3 of Reference 6-1, Attachment 6 were replaced by APS' response to NRC Question 32 in Attachment 2 of Reference 6-2.

There has been no change to the ECCS performance AOR since its review and evaluation conducted by the NRC staff as reported in the Safety Evaluation Report (SER) issued for Unit 2 in Section 4.1 of Reference 6-3.

The following sections restate the description and results of the existing ECCS performance AOR, and discuss topics that were the subject of NRC requests for additional information during the review of the Unit 2 PURLR.

Section 6.1.1 Introduction

The existing PVNGS ECCS performance AOR consists of three parts:

- LBLOCA,
- SBLOCA, and
- post-Loss-of-Coolant Accident (LOCA) long-term cooling.

The AOR was performed for a rated core power of 4070 MWt. The AOR was evaluated for the RSGs (with up to 10% tube plugging per Steam Generator (SG)) and was found applicable.

The LBLOCA and SBLOCA analyses were performed for ZIRLO™ cladding, which is now in use in Units 1, 2, and 3, and is described in topical report CENPD-404-P-A (Reference 6-4). These analyses implemented the 1999 Evaluation Method (EM) and S2M versions, for LBLOCA and SBLOCA respectively, of the Westinghouse ECCS performance evaluation models for Combustion Engineering designed PWRs (Reference 6-5 and Reference 6-6). These analyses were performed after the submittal of the Unit 2 PURLR (Reference 6-1, Attachment 6), and consequently, were not described in that submittal. However, Sections 6.1.2 and 6.1.3 of Reference 6-1, Attachment 6 were replaced by APS' response to NRC Question 32 in Attachment 2 of Reference 6-2, presenting the description and the results of those analyses. The NRC reviewed and approved these analyses for PVNGS Unit 2 Power Uprate (PUR) in Reference 6-3.

Section 6.1.2 Large Break Loss-of-Coolant Accident

The LBLOCA AOR, which was approved for the Unit 2 PUR in Reference 6-3, is applicable to Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. The results of the analysis are summarized in Table 6.1-1. The results conform to the criteria of 10 CFR 50.46(b) (1)-(4) (Reference 6-7).

The LBLOCA AOR was performed in support of the introduction of ZIRLO™ cladding described in topical report CENPD-404-P-A (Reference 6-4), which is now in use in Units 1, 2, and 3. This analysis utilized the 1999 EM described in topical report CENPD-132, Supplement 4-P-A (Reference 6-5). APS notified the NRC of the new analysis pursuant to 10 CFR 50.46(a)(3)(ii) in Reference 6-8 stating that this analysis utilizing the 1999 EM was the new LBLOCA AOR for Units 1, 2, and 3. Subsequently, in response to NRC Question 32 in Attachment 2, Reference 6-2, APS re-affirmed that the ZIRLO™ analysis utilizing the 1999 EM was the AOR for LBLOCA ECCS performance for Unit 2 PUR, replacing the analysis that was described in Section 6.1.2 of Reference 6-1.

The responses to NRC Questions 3, 31.a, and 31.b in Attachment 2, Reference 6-2, provided a comprehensive list of codes and methods used in the LBLOCA analysis, their SERs, and limitations/constraints imposed by those SERs. Since this additional information was requested for the analysis presented in Reference 6-1, the responses provided the information for the codes and methodology utilized in that analysis, namely, the 1985 EM. Therefore, this submittal presents an update of the same information for the current AOR, which utilizes the 1999 EM and analyzes ZIRLO™ cladding. The proprietary update to this information is provided in Attachment 5. The non-proprietary update to this information is presented in Table 6.1-2. Specifically, Table 6.1-2 lists the topical reports and associated SERs that, in conjunction with those provided in the response to NRC Question 3 of Reference 6-2, comprise the 1999 EM.

The responses to NRC Questions 31.a, 31.b, and 31.c in Attachment 2, Reference 6-2, provided additional information regarding the LBLOCA evaluation model and its applicability to Unit 2 at the uprated power. These responses are applicable to Units 1 and 3. However, it is noted that the discussion of an error in the decay heat energy redistribution factor in response to Question 31.b is no longer pertinent since, as described above, the current LBLOCA analysis uses the 1999 EM, which was not impacted by the error.

The response to NRC Question 31.d in Attachment 2, Reference 6-2, and, subsequently in Attachment 2, Reference 6-9, provided additional information on processes in place to assure that input values to the LBLOCA analysis for peak cladding temperature-sensitive parameters bound the as-operated plant values for the parameters. These responses are applicable to Units 1 and 3. The core reload process continues to assure that plant configuration values are bounded by the values used in the AOR.

In addition, APS provided the NRC with information related to pre-accident oxidation in Attachment 4 of Reference 6-10 and in Reference 6-11 that allowed the staff to conclude that the AOR results for maximum cladding oxidation meet the maximum cladding oxidation criterion of 10 CFR 50.46(b)(2). The comparable information for ZIRLO™ cladding that was identified for Zircaloy cladding in Reference 6-11 is documented in Figure 4.5.2-1 of Reference 6-4.

The minimum containment pressure analysis associated with the LBLOCA analysis is described in Section 6.2.1.5 of the PVNGS UFSAR (Reference 6-12). The UFSAR had not been updated to include a description of the analysis prior to the submittal of the Unit 2 PURLR. This prompted APS' response to the NRC Plant Systems Branch request for additional information (Question 12 of Reference 6-13). The UFSAR has been revised. The UFSAR explicitly describes the minimum containment pressure analysis that is applicable to PUR.

Table 6.1-1
Summary of Results of the LBLOCA ECCS Performance Analysis

Parameter	ZIRLO™ Cladding	Zircaloy Cladding
Core power, MW_t	4070	4070
Peak Linear Heat Generation Rate (PLHGR), kW/ft	13.1	13.1
Limiting break size	0.8 DEG/PD(1)	0.6 DEG/PD(1)
Peak cladding temperature, °F	2087	2110
Time of peak cladding temperature, sec	232	266
Maximum cladding oxidation, %	12.0	7.6
Maximum core-wide cladding oxidation, %	<0.73	<0.57
Time of cladding rupture, sec	26	48

Note (1) DEG/PD = Double-Ended Guillotine Break in Reactor Coolant Pump Discharge Leg.

Table 6.1-2
1999 EM LBLOCA Evaluation Model Topical Reports and SERs

Subject(1)	Topical Report Reference	SER Reference
1999 EM topical report, Supplement 4-P-A to CENPD-132	Reference 6-5	Reference 6-14
Implementation of ZIRLO™ cladding in CE fuel assembly designs	Reference 6-4	Reference 6-15

Note (1) See Table 3-1 of Reference 6-2 for the topical reports and associated SERs that, by reference, are part of the 1999 EM.

Section 6.1.3 Small Break Loss-of-Coolant Accident

The current SBLOCA analysis, which was approved for the Unit 2 PUR in Reference 6-3, is applicable to Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. The results of the analysis are summarized in Table 6.1-3. The results conform to the criteria of 10 CFR 50.46(b) (1)-(4) (Reference 6-7).

The current SBLOCA AOR was performed in support of the introduction of ZIRLO™ cladding described in topical report CENPD-404-P-A (Reference 6-4), which is now in use in Units 1, 2, and 3. The analysis utilized the SBLOCA methodology (S2M) described in topical report CENPD-137, Supplement 2-P-A (Reference 6-6). APS notified the NRC of the new analysis pursuant to 10 CFR 50.46(a)(3)(ii) in Reference 6-8 stating that this analysis utilizing the S2M was the new SBLOCA AOR for Units 1, 2, and 3. Subsequently, in response to NRC Question 32 in Attachment 2, Reference 6-2, APS stated that the ZIRLO™ analysis utilizing the S2M was the new AOR for SBLOCA ECCS performance for Unit 2 PUR, replacing the analysis that was described in Section 6.1.3 of Reference 6-1.

The responses to NRC Questions 3, 31.a, and 31.b in Attachment 2, Reference 6-2, provided a comprehensive list of codes and methods used in the SBLOCA analysis, their SERs, and limitations/constraints imposed by those SERs. Since this additional information was requested for the analysis presented in Reference 6-1, the responses provided the information for the codes and methodology utilized in that analysis, namely, the S1M. Therefore, this submittal presents an update of the same information for the current AOR, which utilizes the S2M and analyzes ZIRLO™ cladding. The proprietary update to this information is provided in Attachment 5. Table 6.1-4 updates the non-proprietary information for the S2M. There are no additional limitations/constraints imposed by the SER for the S2M topical report.

The responses to NRC Questions 31.a, 31.b, 31.c, and 33 in Attachment 2, Reference 6-2, provided additional information regarding the SBLOCA evaluation model and its applicability to Unit 2 for PUR. With the exception of the discussion of the error in the CEFLASH-4AS computer code, the responses to these questions (Reference 6-2) are applicable to Units 1 and 3. The discussion of the error in the CEFLASH-4AS computer code is no longer pertinent since the corrected version of the code was used in the SBLOCA analysis described above.

The response to NRC Question 31.d in Attachment 2, Reference 6-2, and, subsequently in Attachment 2, Reference 6-9, provided additional information on processes in place to assure that input values to the SBLOCA analysis for peak cladding temperature-sensitive parameters bound the as-operated plant values for the parameters. These responses are applicable to Units 1 and 3. The core reload process continues to assure that plant configuration values are bounded by the values used in the AOR.

Attachment 2 to Reference 6-10 described a study that was requested to support the applicability of the S2M to Unit 2 at the uprated core power of 3990 MW_t. The study repeated a similar study, documented in the S2M topical report (Reference 6-6), which was performed for a 3400 MW_t Combustion Engineering designed Pressurized Water Reactor (PWR). The study described in Reference 6-10 is applicable to Units 1 and 3.

As shown in Table 6.1-3, the SBLOCA analysis was performed at a PLHGR of 13.5 kW/ft. This is different from the LBLOCA analysis, which was performed at a PLHGR of 13.1 kW/ft (Table 6.1-1). The response to NRC Question 3 in Attachment 2, Reference 6-2, explained the difference between the two values describing the analysis history that led to the difference. The same two values were maintained in the SBLOCA and the LBLOCA analyses described in this submittal.

Table 6.1-3
Summary of Results of the SBLOCA ECCS Performance Analysis

Parameter	Value
Core power, MW_t	4070
PLHGR, kW/ft	13.5
Limiting break size	0.05 ft^2/PD[1]
Peak cladding temperature, °F	1618
Time of peak cladding temperature, sec	1592
Maximum cladding oxidation, %	1.28
Maximum core-wide cladding oxidation, %	<0.2
Time of cladding rupture, sec	No rupture

Note (1) PD = Reactor Coolant Pump Discharge Leg.

Table 6.1-4
S2M SBLOCA Evaluation Model Topical Reports and SERs

Subject[1]	Topical Report Reference	SER Reference
S2M topical report, Supplement 2-P-A to CENPD-137	Reference 6-6	Reference 6-16
Implementation of ZIRLO™ cladding in CE fuel assembly designs	Reference 6-4	Reference 6-15

Note (1) See Table 3-2 of Reference 6-2 for the topical reports and associated SERs that, by reference, are part of the S2M.

Section 6.1.4 Post-Loss-of-Coolant Accident Long-Term Cooling

The current post-LOCA long-term cooling analysis, which was approved for the Unit 2 PUR in Reference 6-3, is applicable to Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. As described in Section 4.1.1 of the SER for the Unit 2 PURLR (Reference 6-3), the NRC concluded that the results of the analysis, in conjunction with the NRC staff evaluation described in Section 4.1.1 of Reference 6-3, meet the regulatory requirements for long-term cooling under 10 CFR 50.46(b)(5) for PUR.

The analysis was performed with the post-LOCA long-term cooling evaluation model described in topical report CENPD-254-P-A (Reference 6-17). The SER for the evaluation model is documented in Reference 6-18. There are no limitations/constraints imposed by the SER.

Information regarding the boric acid precipitation model used in the Unit 2 PUR long-term cooling analysis was provided in response to NRC Question 35 in Attachment 2, Enclosure 1 of Reference 6-2. The response is applicable to Units 1 and 3.

The response to NRC Question 31.d in Attachment 2, Reference 6-2, and, subsequently in Attachment 2, Reference 6-9, provided additional information on processes in place to assure that input values to the long-term cooling analysis for significant parameters bound the as-operated plant values for the parameters. These responses are applicable to Units 1 and 3. The core reload process continues to assure that plant configuration values are bounded by the values used in the AOR.

The response to NRC Question 34 in Attachment 2, Reference 6-2 provided information on design of ECCS switchover from the injection mode to the ECCS sump recirculation mode, and the impact of PUR on assumed heat source and timing of the switchover. In addition, APS provided procedural requirements for the time of initiation of hot leg injection in response to NRC Question 10 in Attachment 2, Reference 6-19. These responses are applicable to Units 1 and 3.

Section 6.1.5 Summary

The existing PVNGS ECCS performance analyses for LBLOCA, SBLOCA, and post-LOCA long-term cooling demonstrate conformance to the ECCS acceptance criteria of 10 CFR 50.46(b)(1)-(5). These analyses are applicable to Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. There has been no change to the AOR for Units 1 and 3 PUR with RSGs since its review and evaluation conducted by the NRC staff as approved in the SER issued for Unit 2 in Section 4.1 of Reference 6-3.

Section 6.2 Containment Response Analysis

The containment response analysis is performed per requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 16 and 50, to demonstrate that the design pressure and temperature conditions for the containment structure are not exceeded. These analyses also define environmental envelopes for mechanical/electrical equipment located within the containment. As for Unit 2, the containment is designed to a pressure of 60 psig and maximum liner temperature of 300 °F and is described in detail in UFSAR Section 6.2.1. The containment response analysis section contained in Reference 6-1, Attachment 6, is unchanged and applicable to Units 1 and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Sections 4.2 and 4.2.1 of Reference 6-3.

Section 6.2.1 Introduction and Background

The postulated accidents considered in determining design containment peak pressure (and temperature) and external pressures are summarized in UFSAR Table 6.2.1-1. Containment analyses were performed at 102% of requested licensed power of 3990 MW_t (4070 MW_t core power).

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Sections 4.2 and 4.2.1 of Reference 6-3.

Section 6.2.2 Loss-of-Coolant Accident Containment Analysis

A LOCA is characterized by the rapid discharge of the Reactor Coolant System (RCS) inventory into the containment. Three break types are investigated:

- Double-Ended Discharge (DEDLSB),
    - Suction Leg Slot Break (DESLSB), and
- the Double-Ended Hot Leg Slot Break (DEHLSB).

All three-break locations are analyzed assuming both minimum and maximum Safety Injection (SI) pump flows. Limiting single failure for these analyses is a loss of one train of Containment Spray System (CSS).

Section 6.2.2.1 Introduction and Background

There are no changes to this section.

Section 6.2.2.2 Description of Loss-of-Coolant Accident Containment Analysis

There are no changes to this section.

Section 6.2.2.3 Methodology Used for Loss-of-Coolant Accident Containment Analysis

Section 6.2.2.3.1 Loss-of-Coolant Accident Mass and Energy Release Calculations

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 6.a, 6.b, and 11 of Attachment 2, Reference 6-13.

Section 6.2.2.3.2 Loss-of-Coolant Accident Long-Term Response

There are no changes to this section.

Section 6.2.2.4 Results of Loss-of-Coolant Accident Containment Analysis

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 6.c of Attachment 2, Reference 6-13 and NRC Questions 10 and 11 of Attachment 2, Reference 6-20.

Section 6.2.2.4.1 Long-Term Containment Performance

There are no changes to this section.

Section 6.2.2.5 Conclusion

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Sections 4.2.2 and 4.2.5 of Reference 6-3.

Section 6.2.3 Main Steam Line Break Containment Analysis

There are no changes to this section.

Section 6.2.3.1 Introduction and Background

There are no changes to this section.

Section 6.2.3.2 Description of Main Steam Line Break Containment Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 7 of Attachment 2, Reference 6-13.

Section 6.2.3.3 Methodology Used for Main Steam Line Break Containment Analysis

Section 6.2.3.3.1 Main Steam Line Break Mass and Energy Release Calculations

There are no changes to this section.

Section 6.2.3.3.2 Containment Response Analysis

There are no changes to this section.

Section 6.2.3.4 Results of Main Steam Line Break Containment Analysis

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 4 of Attachment 2, Reference 6-24 and NRC Question 11 of Attachment 2, Reference 6-20.

Section 6.2.3.5 Conclusion

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.2.3 of Reference 6-3.

Section 6.2.4 Main Steam Line Break Outside Containment Analysis

Section 6.2.4.1 Introduction and Background

There are no changes to this section.

Section 6.2.4.2 Description of Main Steam Line Break Outside Containment Analysis

There are no changes to this section.

Section 6.2.4.3 Methodology Used for 1 ft^2 Main Steam Line Break Outside Containment Equipment Qualification Analysis

Section 6.2.4.3.1 Mass and Energy Release Calculations

There are no changes to this section.

Section 6.2.4.3.2 Change in SGNIII Code Methodology

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 8 and 9 of Attachment 2, Reference 6-13.

Section 6.2.4.3.3 Main Steam Support Structure MSLB Pressure and Temperature Response Analysis

There are no changes to this section.

Section 6.2.4.4 Results of this Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 10 of Attachment 2, Reference 6-13.

Section 6.2.4.5 Conclusion

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.2.4 of Reference 6-3.

Section 6.3 Non-Loss-of-Coolant Accident Transient Analysis

Summary

UFSAR Chapter 15 Non-LOCA transient events that were performed in support of Unit 2 PURLR (Reference 6-1) were evaluated for Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. Safety analyses were performed at 102% of requested licensed power of 3990 MW_t (4070 MW_t core power). The results of the evaluation demonstrate that the Non-LOCA transients analyzed for Unit 2 PUR with RSGs are applicable to Units 1 and 3, and continue to be within the acceptance criteria. Thus, no reanalysis of Chapter 15 transient events was required for Units 1 and 3.

The following sections present the results of the evaluations and discuss topics that were the subject of NRC requests for additional information during the review of the Unit 2 PURLR.

Section 6.3.0 Introduction

All UFSAR Chapter 15 non-LOCA transient analyses that were performed in support of Unit 2 PURLR (Reference 6-1) were evaluated for Units 1 and 3 operating with RSGs at an uprated license power of 3990 MW_t. Table 6.3-1 of this submittal lists the non-LOCA transient events by category and defines the level of evaluation included for this PUR request. All UFSAR events received some level of evaluation to ensure applicability to Units 1 and 3. The levels of evaluation are:

- The analysis "remains bounded" by an existing UFSAR analysis.
- The analysis has been "evaluated" for the Units 1 and 3 PUR, the results of analyses listed in Section 6.3 of Unit 2 PURLR (Reference 6-1) remain bounding, and "reanalysis" was not required.
- The analysis had to be "reanalyzed" as part of this submittal. As shown in Table 6.3-1, no analysis was found in this level.

Note that the NRC has approved a license amendment for replacement of part-length Control Element Assemblies (CEAs) with part-strength CEA (Reference 6-21). The replacement of the part-length/part-strength CEAs will not affect any of the Chapter 15 analysis adversely.

Table 6.3-1
Non-LOCA Transient Events
(Page 1 of 3)

PURLR	UFSAR	Transient Event	Category	Units 1 and 3 Assessment
6.3.1	15.1	Increase in Heat Removal by the Secondary System		
6.3.1.1	15.1.1	Decrease in Feedwater (FW) Temperature	Anticipated Operational Occurrence (AOO)	Remains bounded by the Increase in main steam flow event.
6.3.1.2	15.1.2	Increase in FW Flow	AOO	Remains bounded by the Increase in main steam flow event.
6.3.1.3	15.1.3	Increase in Main Steam Flow	AOO	Evaluated as part of the Core Protection Calculator (CPC) signal filters analysis. No reanalysis required. The Core Protection Calculator System (CPCS) Upgrade is described in Section 6.3.0.3.
6.3.1.3		Steam Bypass Control System (SBCS) Malfunction	AOO	Evaluated. No reanalysis required.
6.3.1.4	15.1.4	Inadvertent Opening of an Atmospheric Dump Valve (ADV) (IOSGADV) with a Loss of Offsite Power (LOP)	Infrequent Event	Evaluated. No reanalysis required.
6.3.1.5 6.3.1.7	15.1.5	MSLB - Modes 1 and 2, Post-Trip Return-to-Power (R-t-P), Pre-Trip Power Excursion	Limiting Fault	Evaluated. No reanalysis required.
6.3.1.6	15.1.6	MSLB - Mode 3 Post-Trip R-t-P	Limiting Fault	Evaluated. No reanalysis required.
6.3.2	15.2	Decrease in Heat Removal by the Secondary System		
6.3.2.1	15.2.1	Loss of External Load	AOO	Evaluated as part of the CPC signal filters analysis. No reanalysis required. The CPCS Upgrade is described in Section 6.3.0.3.
6.3.2.2	15.2.2	Turbine Trip	AOO	Remains bounded by the Loss of Condenser Vacuum (LOCV) Event.
6.3.2.3	15.2.3	LOCV	AOO	Evaluated. No reanalysis required. Note that Reference 6-2 augmented Section 6.3.2.3 of Reference 6-1.
6.3.2.4	15.2.4	Main Steam Isolation Valves (MSIVs) Closure	AOO	Remains bounded by the LOCV Event.
6.3.2.5	15.2.5	Steam Pressure Regulator Failure	AOO	N/A
6.3.2.6	15.2.6	Loss of Non-Emergency AC Power	AOO	Remains bounded by the LOCV Event.
6.3.2.7	15.2.7	Loss of Normal FW Flow	AOO	Remains bounded by the LOCV Event.
6.3.2.8	15.2.8	Feedwater Line Breaks (FWLBs)	Limiting Fault	Evaluated. No reanalysis required. Note that Reference 6-2 and Reference 6-22 augmented Section 6.3.2.8 of Reference 6-1.

Table 6.3-1
Non-LOCA Transient Events

PURLR Section	UFSAR	Transient Event	Category	Units 1 and 3 Assessment
6.3.3	15.3	Decrease in RCS Flowrate		
6.3.3.1	15.3.1	Total Loss of RCS Flowrate	AOO	Evaluated. No reanalysis required. Note that Reference 6-2 augmented Section 6.3.3.1 of Reference 6-1.
6.3.3.2	15.3.2	Flow Controller Malfunction	AOO	N/A
6.3.3.3	15.3.3	Single Reactor Coolant Pump (RCP) Rotor Seizure with a LOP	Limiting Fault	Remains bounded by the Single RCP Shaft Break Event.
6.3.3.4	15.3.4	Single RCP Shaft Break with LOP	Limiting Fault	Evaluated. No reanalysis required.
6.3.4	15.4	Reactivity and Power Distribution Anomalies		
6.3.4.1	15.4.1	Uncontrolled Control Element Assembly Withdrawal (CEAW) - Subcritical and Hot Zero Power (HZP)	AOO	Evaluated. No reanalysis required.
6.3.4.2	15.4.2	Uncontrolled CEAW at Power	AOO	Evaluated as part of CPC signal filters analysis. No reanalysis required. The CPCS Upgrade is described in Section 6.3.0.3.
6.3.4.3	15.4.3	Single Full-Length CEA Drop	AOO	Evaluated. No reanalysis required.
6.3.4.4	15.4.4	Startup of an Inactive RCP	AOO	Evaluated. No reanalysis required.
6.3.4.5	15.4.5	Flow Controller Malfunction Causing an Increase in Boiling Water Reactor (BWR) Core Flow	AOO	N/A
6.3.4.6	15.4.6	Inadvertent Deboration (ID)	AOO	Evaluated. No reanalysis required.
6.3.4.7	15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Location	AOO	Evaluated. No reanalysis required.
6.3.4.8	15.4.8	CEA Ejection	Limiting Fault	Evaluated. No reanalysis required. Note that Reference 6-2 augmented Section 6.3.4.8 of Reference 6-1. Fuel failure is evaluated on a cycle-by-cycle basis.
6.3.5	15.5	Increase in RCS Inventory		
6.3.5.1	15.5.1	Inadvertent Operation of ECCS	AOO	Evaluated. No reanalysis required.
6.3.5.2	15.5.2	Chemical and Volume Control System (CVCS) Malfunction - Pressurizer Level Control System (PLCS) Malfunction with LOP	Infrequent Event	Evaluated. No reanalysis required.

Table 6.3-1
Non-LOCA Transient Events
(Page 3 of 3)

PURLR Section	UFSAR	Transient Event	Category	Units 1 and 3 Assessment
6.3.6	15.6	Decrease in RCS Inventory		
6.3.6.1	15.6.1	Inadvertent Opening of a Pressurizer Safety Valve (PSV)	AOO	Evaluated as part of ECCS performance analysis in Section 6.1 and Section 6.3.6.1. No reanalysis required.
6.3.6.2	15.6.2	Double-Ended Break of a Letdown Line Outside Containment of the letdown line control valve (DBLLOCUS)	Limiting Fault	Evaluated. No reanalysis is required. Note that the transient portion of this event was reanalyzed for all PVNGS units after Reference 6-3 was issued. See Section 6.3.6.2.
6.3.6.3	15.6.3	Steam Generator Tube Rupture (SGTR) with LOP (SGTRLOP)	Limiting Fault	Evaluated for steam generator overfill. No reanalysis is required.
		SGTRLOP and single failure	Limiting Fault	Evaluated with respect to radiological consequences described in Section 6.4.6.2. No reanalysis required.
6.3.6.4	15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Limiting Fault	N/A
6.3.6.5	15.6.5	LOCAs	Limiting Fault	Evaluated as part of ECCS performance analysis Section 6.1 and Section 6.3.6.5.
6.3.7	15.7	Radioactive Material Release from a Subsystem or Component		
6.3.7.1	15.7.1	Waste Gas System Failure	Limiting Fault	Evaluated. No reanalysis required.
6.3.7.2	15.7.2	Radioactive Liquid Waste System Leak or Failure	Limiting Fault	Evaluated. No reanalysis required.
6.3.7.3	15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Limiting Fault	Evaluated. No reanalysis required.
6.3.7.4	15.7.4	Radioactive Consequences of Fuel Handling Accidents	Limiting Fault	Evaluated. Note that there is a pending license amendment request (Reference 6-23). See Section 6.4.7.3. No reanalysis is required for Units 1 and 3 PUR.

Section 6.3.0.1 Methodology and Computer Codes

This section contained in Attachment 6 of Reference 6-1 is applicable to Units 1 and 3 as augmented by response to NRC Questions 4, 5, 8, and 10 of Attachment 2, of Reference 6-2, NRC Question 13 of Attachment 2, of Reference 6-19, and as supplemented by Enclosure 1, of Reference 6-22.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3 of Reference 6-3.

Section 6.3.0.2 Initial Conditions

This section contained in Reference 6-1, Attachment 6 is applicable to Units 1 and 3 as augmented by the response to NRC Questions 18.a, 18.b, and 18.c of Attachment 2, Reference 6-2, NRC Question 1 of Attachment 2, Reference 6-19, and NRC Question 2 of Attachment 2, Reference 6-24.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3 of Reference 6-3.

Section 6.3.0.3 Reactor Protection Systems

This section contained in Reference 6-1, Attachment 6 is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1 of Attachment 2, Reference 6-24 and NRC Question 6 of Attachment 2, Reference 6-2.

APS submitted a request for CPCS Upgrade after the submittal of Reference 6-1. Therefore, the upgraded CPCS was not discussed in Section 6.3.0.3 of Reference 6-1. The NRC issued the license amendment for CPCS upgrade (Reference 6-25). Enclosure 4 of Reference 6-25, Section 3.3.3, "Impact on UFSAR Chapter 15 Transients and Accidents," states: "The NRC staff has reviewed the information provided and finds that the upgraded CPCS will have no impact on the UFSAR Chapter 15 transients and accidents because the CPCS response times and accuracy assumed in the UFSAR Chapter 15 analyses for the Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) trip functions remain bounding." This conclusion was applicable to Unit 2 PUR, and is applicable to Units 1 and 3 at uprated power and RSGs.

Therefore, the upgrade of the CPCS does not make changes to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.1 of Reference 6-3.

Section 6.3.0.3.1 CPC Dynamic Signal Filter Coefficients

This section as contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 to provide a clearer and more thorough discussion of the CPC Dynamic Filter Coefficients.

These sections as described in Attachment 4 of Reference 6-2 are applicable to Units 1 and 3. The transients examined for increasing power for Unit 2 PUR were evaluated and determined to be applicable to Units 1 and 3. Therefore, no reanalysis is required.

Section 6.3.0.3.1.1 Increasing Power Signal Filters

This section contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 and is applicable to Units 1 and 3.

Section 6.3.0.3.1.2 Increasing Reactor Coolant System Temperature Signal Filters

This section contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 and is applicable to Units 1 and 3.

Section 6.3.0.3.1.3 Decreasing Reactor Coolant System Temperature Signal Filters

This section contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 and is applicable to Units 1 and 3.

Section 6.3.0.3.1.4 Decreasing Pressure Penalty

This section contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 and is applicable to Units 1 and 3.

Section 6.3.0.3.1.5 Results

This section contained in Attachment 6 of Reference 6-1 was replaced in its entirety by Attachment 4 of Reference 6-2 and is applicable to Units 1 and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3 of Reference 6-3.

Section 6.3.0.4 Engineered Safety Features

This section contained in Reference 6-1, Attachment 6 is applicable to Units 1 and 3 as augmented by the response to NRC Question 1 in Attachment 2, Reference 6-24.

The plant modification to change Main Steam Isolation Signal (MSIS) on Low Steam Generator Pressure (LSGP) setpoint as explained in the footnote to Table 6.3-4 of Reference 6-1 is applicable to Units 1 and 3. The LSGP setpoint, currently at 890 psia, will be changed to 955 psia with the implementation of the PUR to Units 1 and 3.

Section 6.3.1 Increase In Heat Removal By The Secondary System

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.1.1 Decrease in Feedwater Temperature

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1 of Reference 6-3.

Section 6.3.1.2 Increase in Feedwater Flow

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1 of Reference 6-3.

Section 6.3.1.3 Increased Main Steam Flow

Section 6.3.1.3.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.1.3.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.1.3.3 Description of Analysis

There are no changes to this section.

Section 6.3.1.3.3.1 Transient Simulation

This section contained in Reference 6-1, Attachment 6 is applicable to Units 1 and 3 as augmented by the response to NRC Question 7 of Attachment 2, Reference 6-2.

Section 6.3.1.3.4 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.1.3.5 Results

There are no changes to this section.

Section 6.3.1.3.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1.1 of Reference 6-3.

Section 6.3.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Section 6.3.1.4.1 Identification of Event and Causes

This section contained in Reference 6-1, Attachment 6, is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2, Reference 6-2.

Section 6.3.1.4.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.1.4.3 Description of Analysis

There are no changes to this section.

Section 6.3.1.4.3.1 Transient Simulation

There are no changes to this section.

Section 6.3.1.4.4 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.1.4.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 12 of Attachment 2, Reference 6-19.

Section 6.3.1.4.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1.2 of Reference 6-3.

Section 6.3.1.5 Steam System Piping Failures Inside and Outside Containment - Mode 1 Operation

Section 6.3.1.5.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.1.5.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.1.5.3 Description of Analysis

There are no changes to this section.

Section 6.3.1.5.3.1 Change in Method of Evaluation

This section is applicable to Units 1 and 3 as augmented by responses to NRC Question 13 of Attachment 2, Reference 6-19 and NRC Question 5 of Attachment 2, Reference 6-2.

The changes in methodology approved for Unit 2 (Reference 6-3) are being implemented in Units 1 and 3 in accordance with 10 CFR 50.59.

Section 6.3.1.5.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 9, and 10 of Attachment 2, Reference 6-2.

Section 6.3.1.5.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 14 and 16 of Attachment 2, Reference 6-19.

Section 6.3.1.5.5.1 Change in Methodology Reactivity Credit - Moderator Density Feedback

The changes in methodology approved for Unit 2 (Reference 6-3) are being implemented in Units 1 and 3 in accordance with 10 CFR 50.59.

Section 6.3.1.5.6 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 15 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1.3.1 of Reference 6-3.

Section 6.3.1.6 Steam System Piping Failures Inside and Outside Containment - Mode 3 Operation

Section 6.3.1.6.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.1.6.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.1.6.3 Description of Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 9 and 10 of Attachment 2, Reference 6-2.

Section 6.3.1.6.4 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.1.6.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 17 of Attachment 2, Reference 6-19.

Section 6.3.1.6.6 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 15 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1.3.2 of Reference 6-3.

Section 6.3.1.7 Pre-Trip Main Steam Line Break Power Excursion

Section 6.3.1.7.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.1.7.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.1.7.3 Description of Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 10 of Attachment 2, Reference 6-2.

Section 6.3.1.7.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 10 of Attachment 2, Reference 6-2.

Section 6.3.1.7.5 Results

There are no changes to this section.

Section 6.3.1.7.6 Conclusions

The conclusions section is contained in Reference 6-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 15 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.1.4 of Reference 6-3.

Section 6.3.2 Decrease in Heat Removal By The Secondary System

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.2.1 Loss of External Load

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2 of Reference 6-3.

Section 6.3.2.2 Turbine Trip

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2 of Reference 6-3.

Section 6.3.2.3 Loss of Condenser Vacuum

Section 6.3.2.3.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.2.3.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.2.3.3 Description of Analysis

There are no changes to this section.

Section 6.3.2.3.3.1 Transient Simulation

There are no changes to this section.

Section 6.3.2.3.4 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.2.3.5 Results

This section is applicable to Units 1 and 3 as modified by Attachment 5 and Enclosure 2 of Reference 6-2. Following the submittal of Reference 6-1, Westinghouse informed APS that during the conversion from CESEC code to CENTS code, a correction factor was inadvertently omitted from the calculation for PSV sizing. Subsequently, LOCV event was reanalyzed using the corrected PSV sizing calculation and the revised results were presented in Attachment 5 and Enclosure 2 of Reference 6-2.

Section 6.3.2.3.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2 of Reference 6-3.

Section 6.3.2.4 Main Steam Isolation Valve Closure

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2.1 of Reference 6-3.

Section 6.3.2.5 Steam Pressure Regulator Failure

As described in UFSAR Section 15.2.5, this event does not apply to the CE SYSTEM 80 design and therefore is not presented.

Section 6.3.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2.2 of Reference 6-3.

Section 6.3.2.7 Loss of Normal Feedwater Flow

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2.3 of Reference 6-3.

Section 6.3.2.8 Feedwater System Pipe Breaks

As modified by Attachment 2, Enclosure 1 of Reference 6-22, Section 6.3.2.8.1, the FWLB -Long Term Cooling event assumes that the plant is operated on program T_{ave}, and the PLCS is in the automatic mode of operation at the beginning of the event. There are no additional changes to this section.

Section 6.3.2.8.1 Feedwater Line Break Event with Concurrent Loss of Offsite Power

Section 6.3.2.8.1.1 Identification of Event and Causes

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 18 of Attachment 2, Reference 6-19, and as modified by Attachment 2, Enclosure 1 of Reference 6-22.

Section 6.3.2.8.1.2 Acceptance Criteria

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 19 of Attachment 2, Reference 6-19.

Section 6.3.2.8.1.3 Description of Analysis

This section is applicable to Units 1 and 3 as revised by Attachment 2, Enclosure 1 of Reference 6-22.

Section 6.3.2.8.1.3.1 Transient Simulation

This section is applicable to Units 1 and 3 as revised by Attachment 2, Enclosure 1 of Reference 6-22.

Section 6.3.2.8.1.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as revised by Attachment 2, Enclosure 1 of Reference 6-22, and as augmented by NRC Question 11 of Attachment 2, Reference 6-2 and NRC Question 20 of Attachment 2, Reference 6-19.

Section 6.3.2.8.1.5 Results

This section is applicable to Units 1 and 3 as revised by Attachment 5 and Enclosure 2 of Reference 6-2 and Attachment 2, Enclosure 1 of Reference 6-22.

Section 6.3.2.8.1.6 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 19 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2.4 of Reference 6-3.

Section 6.3.2.8.2 Small Feedwater Line Break Event

Section 6.3.2.8.2.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.2.8.2.2 Acceptance Criteria

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 19 of Attachment 2, Reference 6-19.

Section 6.3.2.8.2.3 Description of Analysis

There are no changes to this section.

Section 6.3.2.8.2.3.1 Transient Simulation

There are no changes to this section.

Section 6.3.2.8.2.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 11 of Attachment 2, Reference 6-2.

Section 6.3.2.8.2.5 Results

This section is applicable to Units 1 and 3 as revised by Attachment 5 and Enclosure 2 of Reference 6-2. Following the submittal of Reference 6-1, Westinghouse informed APS that during the conversion from CESEC code to CENTS code, a correction factor was inadvertently omitted from the calculation for PSV sizing. Subsequently, the Small Feedwater Line Break (SFWLB) event was reanalyzed with the corrected PSV sizing calculation and the revised results were presented in Attachment 5 and Enclosure 2 of Reference 6-2.

Section 6.3.2.8.2.6 Conclusions

This section is applicable to Units 1 and 3 as augmented by NRC Question 19 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.2.4 of Reference 6-3.

Section 6.3.3 Decrease in Reactor Coolant Flowrate

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.3.1 Total Loss of Reactor Coolant Flow

Section 6.3.3.1.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.3.1.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.3.1.3 Description of Analysis

There are no changes to this section.

Section 6.3.3.1.3.1 Transient Simulation

There are no changes to this section.

Section 6.3.3.1.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 21 of Attachment 2, Reference 6-19.

Section 6.3.3.1.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 12 of Attachment 2, Reference 6-2, and as revised by Attachment 5 and Enclosure 2 of Reference 6-2. Following the submittal of Reference 6-1, Westinghouse informed APS that during the conversion from CESEC code to CENTS code, a correction factor was inadvertently omitted from the calculation for PSV sizing. Subsequently, Loss of Flow (LOF) event was reanalyzed with the corrected PSV sizing calculation and the revised results were presented in Attachment 5 and Enclosure 2 of Reference 6-2.

Section 6.3.3.1.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.3.1 of Reference 6-3.

Section 6.3.3.2 Flow Controller Malfunction Causing a Flow Coastdown

As described in UFSAR Section 15.3.2 classifies the flow controller malfunction event as pertaining to BWRs. This event is not applicable and is not analyzed.

Section 6.3.3.3 Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power

As described in UFSAR Section 15.3.3, the rotor seizure event would be no more adverse than the RCP shaft break event as discussed in Section 6.3.3.4.

Section 6.3.3.4 Reactor Coolant Pump Shaft Break with Loss of Offsite Power

Section 6.3.3.4.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.3.4.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.3.4.3 Description of Analysis

There are no changes to this section.

Section 6.3.3.4.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 4, and 13 of Attachment 2, Reference 6-2 and NRC Question 21 of Attachment 2, Reference 6-19.

Section 6.3.3.4.5 Results

There are no changes to this section.

Section 6.3.3.4.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.3.2 of Reference 6-3.

Section 6.3.4 Reactivity and Power Distribution Anomalies

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.4.1 Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition

Section 6.3.4.1.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.4.1.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.4.1.3 Uncontrolled Control Element Assembly Withdrawal from a Subcritical Condition

Section 6.3.4.1.3.1 Description of Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 23.a of Attachment 2, Reference 6-2 and NRC Question 3 of Attachment 2, Reference 6-19.

Section 6.3.4.1.3.1.1 Transient Simulation

There are no changes to this section.

Section 6.3.4.1.3.2 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 23.a and 23.b of Attachment 2, Reference 6-2 and NRC Questions 3 and 4 of Attachment 2, Reference 6-19.

Section 6.3.4.1.3.3 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 24, 25, 26, and 30 of Attachment 2, Reference 6-2 and NRC Question 5 of Attachment 2, Reference 6-19.

Section 6.3.4.1.3.4 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 30 of Attachment 2, Reference 6-2.

As a result of the NRC review of Unit 2 PURLR (Reference 6-1), APS submitted a license amendment request (Reference 6-27) to change the Technical Specification safety limit from PLHGR to a fuel centerline melt temperature safety limit. Technical Specification Amendment No. 145 (Reference 6-28) changed the safety limit for Units 1, 2, and 3. The changed safety limit is applicable to Units 1 and 3 operating at uprated power with RSGs.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.1 of Reference 6-3.

Section 6.3.4.1.4 Uncontrolled Control Element Assembly Withdrawal from a Low Power Condition

Section 6.3.4.1.4.1 Description of Analysis

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 23.a of Attachment 2, Reference 6-2 and NRC Question 3 of Attachment 2, Reference 6-19.

Section 6.3.4.1.4.1.1 Transient Simulation

There are no changes to this section.

Section 6.3.4.1.4.2 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 23.a, and 23.b of Attachment 2, Reference 6-2 and NRC Questions 3 and 4 of Attachment 2, Reference 6-19.

Section 6.3.4.1.4.3 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 26 and 30 of Attachment 2, Reference 6-2 and NRC Question 5 of Attachment 2, Reference 6-19.

Section 6.3.4.1.4.4 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 30 of Attachment 2, Reference 6-2.

As a result of the NRC review of Unit 2 PURLR (Reference 6-1), APS submitted a license amendment request (Reference 6-27) to change the Technical Specification safety limit from PLHGR to a fuel centerline melt temperature safety limit. Technical Specification Amendment No. 145 (Reference 6-28) changed the safety limit for Units 1, 2, and 3. The changed safety limit is applicable to Units 1 and 3 operating at uprated power with RSGs.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.1 of Reference 6-3.

Section 6.3.4.2 Uncontrolled Control Element Assembly Withdrawal at Power

Section 6.3.4.2.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.4.2.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.4.2.3 Description of Analysis

There are no changes to this section.

Section 6.3.4.2.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 27 of Attachment 2, Reference 6-2, and NRC Question 6 of Attachment 2, Reference 6-19.

Section 6.3.4.2.5 Results

There are no changes to this section.

Section 6.3.4.2.6 Conclusions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 30 of Attachment 2, Reference 6-2.

As a result of the NRC review of Unit 2 PURLR (Reference 6-1), APS submitted a license amendment request (Reference 6-27) to change the Technical Specification safety limit from PLHGR to a fuel centerline melt temperature safety limit. Technical Specification Amendment No. 145 (Reference 6-28) changed the safety limit for Units 1, 2, and 3. The changed safety limit is applicable to Units 1 and 3 operating at uprated power with RSGs.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.2 of Reference 6-3.

Section 6.3.4.3 Single Full Length Control Element Assembly Drop

Section 6.3.4.3.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.4.3.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.4.3.3 Description of Analysis

There are no changes to this section.

Section 6.3.4.3.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 28.a, and 28.b of Attachment 2, Reference 6-2 and NRC Questions 6, and 7 of Attachment 2, Reference 6-19.

Section 6.3.4.3.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 28.c of Attachment 2, Reference 6-2.

Section 6.3.4.3.6 Conclusions

As a result of the NRC review of Unit 2 PURLR (Reference 6-1), APS submitted a license amendment request (Reference 6-27) to change the Technical Specification safety limit from PLHGR to a fuel centerline melt temperature safety limit. Technical Specification Amendment No. 145 (Reference 6-28) changed the safety limit for Units 1, 2, and 3. The changed safety limit is applicable to Units 1 and 3 operating at uprated power with RSGs.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.5 of Reference 6-3.

Note that the NRC has approved a license amendment for replacement of part-length Control Element Assemblies (CEAs) with part-strength CEA (Reference 6-21). The replacement of the part-length/part-strength CEAs will remain bounding for all CEA drop type of events.

Section 6.3.4.4 Startup of an Inactive Reactor Coolant Pump

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.4 of Reference 6-3.

Section 6.3.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow

As described in UFSAR Section 15.4.5, this event is not applicable to PWRs and, therefore, is not included in this submittal.

Section 6.3.4.6 Inadvertent Deboration

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 23 of Attachment 2, Reference 6-19.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.5 of Reference 6-3.

Section 6.3.4.7 Inadvertent Loading of a Fuel Assembly into the Improper Position

There are no changes to this section.

Section 6.3.4.8 Control Element Assembly Ejection

Section 6.3.4.8.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.4.8.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.4.8.3 Description of Analysis

Section 6.3.4.8.3.1 Fuel Performance Case

There are no changes to this section.

Section 6.3.4.8.3.2 Peak Reactor Coolant System Pressure Case

There are no changes to this section.

Section 6.3.4.8.3.3 Transient Simulation

There are no changes to this section.

Section 6.3.4.8.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 29.a, 29.b, 29.c, 29.d, and 29.e of Attachment 2, Reference 6-2 and NRC Question 8 of Attachment 2, Reference 6-19.

Section 6.3.4.8.5 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 29.e of Attachment 2, Reference 6-2, and as revised by Attachment 5 and Enclosure 2 of Reference 6-2.

Following the submittal of Reference 6-1, Westinghouse informed APS that during the conversion from CESEC code to CENTS code, a correction factor was inadvertently omitted from the calculation for PSV sizing. Subsequently, CEA ejection primary peak pressure event was reanalyzed with the corrected PSV sizing calculation and the revised results were presented in Attachment 5 and Enclosure 2 of Reference 6-2.

Section 6.3.4.8.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.4.6 of Reference 6-3.

Section 6.3.5 Increase in Reactor Coolant System Inventory

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.5.1 Inadvertent Operation of the Emergency Core Cooling System

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.5 of Reference 6-3.

Section 6.3.5.2 Chemical and Volume Control System Malfunction - Pressurizer Level Control System Malfunction with a Concurrent Loss of Offsite Power

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.5 of Reference 6-3.

Section 6.3.6 Decrease in Reactor Coolant System Inventory

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.6.1 Inadvertent Opening of a Pressurizer Safety/Relief Valve

As described in UFSAR Section 15.6.1, this event is evaluated in the ECCS analyses (Section 6.1).

Section 6.3.6.2 Double-Ended Break of a Letdown Line Outside Containment

Section 6.3.6.2.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.6.2.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.6.2.3 Description of Analysis

There are no changes to this section.

Section 6.3.6.2.4 Input Parameters, Initial Conditions, and Assumptions

The assumptions made for DBLLOCUS event listed in Section 6.3.6.2.4 of Reference 6-1 remain unchanged. However, The DBLLOCUS event was reanalyzed for 3876 MW_t and 3990 MW_t configurations after the issuance of PVNGS Unit 2 PUR Amendment (Reference 6-3) to incorporate system components in discharge path. That reanalysis performed a parametric study on initial core inlet temperature to determine the combination of the leak flow and flashing fraction that produces the most limiting release to be used in dose consequences. Therefore, the value presented in Table 6.3-47 of Reference 6-1 are changed, see Table 6.3-2.

Section 6.3.6.2.5 Results

The DBLLOCUS event was reanalyzed for 3876 MW_t and 3990 MW_t configurations after the issuance of PVNGS Unit 2 PUR Amendment (Reference 6-3) to incorporate system components in discharge path. That reanalysis performed a parametric study on initial core inlet temperature to determine the combination of the leak flow and flashing fraction that produces the most adverse dose consequences. As a result of this parametric analysis, the sequence of events that occur following the DBLLOCUS event has changed. Therefore, the value presented in Table 6.3-48 of Reference 6-1 are changed, see Table 6.3-3.

Section 6.3.6.2.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.6.1 of Reference 6-3.

Table 6.3-2
Parameters Used for DBLLOCUS Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temp (°F)	548 & 568	548 & 568
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	23.3	23.3
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-4.2E-04	-4.2E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip–WRSO (%$\Delta\rho$)	no trip	no trip
Fuel rod gap conductance (Btu/hr-ft^2-°F)	5755	5755
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Table 6.3-3
Sequence of Events for DBLLOCUS Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	DBLLOCUS occurs.	---	---
87.2	88.0	Pressurizer backup and proportional heaters on (psia).	2275	2275
243.9	245.4	Third charging pump starts (ft).	18.1	18.1
489.8	492.4	Pressurizer backup and proportional heaters off on low pressurizer level (ft).	12.7	12.7
600	600	RCS inventory release (lb_m).	30936	31035
---	---	Minimum DF	3.14	3.14
600	600	Operator isolates the DBLLOCUS and takes steps for a controlled shutdown.	---	---

Section 6.3.6.3 Steam Generator Tube Rupture

Section 6.3.6.3.1 Steam Generator Tube Rupture without a Concurrent Loss of Offsite Power

There are no changes to this section.

Section 6.3.6.3.2 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power and a Single Failure

Section 6.3.6.3.2.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.6.3.2.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.6.3.2.3 Description of Analysis

There are no changes to this section.

Section 6.3.6.3.2.3.1 Transient Simulation

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 14 and 15 of Attachment 2, Reference 6-2.

Section 6.3.6.3.2.4 Input Parameters, Initial Conditions, and Assumptions

This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 14 and 15 of Attachment 2, Reference 6-2.

Section 6.3.6.3.2.5 Results

There are no changes to this section.

Section 6.3.6.3.2.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Sections 4.3.6.2 and 4.3.6.2.1 of Reference 6-3.

Section 6.3.6.3.3 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

Section 6.3.6.3.3.1 Identification of Event and Causes

There are no changes to this section.

Section 6.3.6.3.3.2 Acceptance Criteria

There are no changes to this section.

Section 6.3.6.3.3.3 Description of Analysis

There are no changes to this section.

Section 6.3.6.3.3.3.1 Transient Simulation

There are no changes to this section.

Section 6.3.6.3.3.4 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.6.3.3.5 Results

There are no changes to this section.

Section 6.3.6.3.3.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.6.2 of Reference 6-3.

Section 6.3.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

As described in UFSAR Section 15.6.4, this event is applicable to BWRs only.

Section 6.3.6.5 Loss-of-Coolant Accidents

ECCS performance and LOCA are discussed in Section 6.1. Radiological consequences of this event are described in Section 6.4.6.3.

Section 6.3.7 Radioactive Material Release from a Subsystem or Component

This section is contained in Reference 6-1, Attachment 6. There are no changes to this section.

Section 6.3.8 Limiting Infrequent Events

This section is contained in Reference 6-1, Attachment 6.

Section 6.3.8.1 Anticipated Operational Occurrences in Combination with a Single Active Failure

There are no changes to this section.

Section 6.3.8.1.1 Acceptance Criteria

There are no changes to this section.

Section 6.3.8.1.2 Description of Analysis

There are no changes to this section.

Section 6.3.8.1.2.1 Transient Simulation

There are no changes to this section.

Section 6.3.8.1.3 Input Parameters, Initial Conditions, and Assumptions

There are no changes to this section.

Section 6.3.8.1.4 Results

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 16 of Attachment 2, Reference 6-2 and NRC Question 22 of Attachment 2, Reference 6-19.

Section 6.3.8.1.5 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.7.1 of Reference 6-3.

Section 6.3.8.2 Anticipated Transient Without Scram

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.7.2 of Reference 6-3.

Section 6.3.8.3 Station Blackout

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.3.7.3 of Reference 6-3.

Section 6.3.8.3.1 Auxiliary Feedwater and Steam Release

There are no changes to this section.

Section 6.3.8.3.2 Loss of Ventilation

There are no changes to this section.

Section 6.3.8.3.3 Condensate Storage Tank Inventory

There are no changes to this section.

Section 6.3.8.3.4 Reactor Coolant System Inventory

There are no changes to this section. Sufficient RCS inventory exists to keep the core covered, and natural circulation, through reflux boiling, will keep the core cooled.

Section 6.4 Radiological Accident Evaluations

The radiological consequences of a fuel handling accident for PUR has an exemption as requested in Reference 6-23 and SRP Section 15.7.4. There are no additional changes to this section.

Section 6.4.0 Methodology Used for Radiological Assessment Analyses

This section is contained in Section 6.4 of Reference 6-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by responses to NRC Questions 4 (sections a through d) and 5 (sections a through j) of Attachment 2, Reference 6-29. Response to NRC question 5 (sections a through j) provided event specific input parameters, and assumptions. There are no changes to the methodology, input parameters, and/or assumptions to affect the conclusions reached by the NRC staff in Section 4.4.1 of Reference 6-3. Refer to Section 9.10.1 for control room doses.

Table 6.4-1 of Reference 6-1, Attachment 6, incorrectly states the value of Kr-88 as 1.30E+07 Ci. Table 6.5-1 of Reference 6-1, Attachment 6 correctly states the value as 1.30E+08 Ci. APS used the correct value for Kr-88 for the dose consequences reported in Reference 6-1. In Reference 6-3, Table 1 the NRC staff used this incorrect value of Kr-88 from Table 6.4-1 for an independent evaluation. If the NRC staff used this value for an independent evauation, the results of this evaluation would yield lower doses than the APS reported values. The higher dose rate in Exclusion Area Boundary (EAB)/Low

Population Zone (LPZ) doses, calculated by APS, continue to meet regulatory requirements.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.1 of Reference 6-3.

Section 6.4.1 Radiological Consequences of Increase in Heat Removal by the Secondary System

Section 6.4.1.1 Radiological Consequences of Inadvertent Opening of a Steam Generator Relief or Safety Valve

The EAB and LPZ offsite radiological consequences of the IOSGADV section were calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by responses to NRC Questions 4 (sections a through d) and 5 (sections a through j) of Attachment 2, Reference 6-29 and NRC Question 5 of Attachment 2, Reference 6-2. The NSSS and core responses to this transient are detailed in Section 6.3.1.4.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.2 of Reference 6-3. The consequences at EAB and LPZ doses for this event were found to be within the acceptance criteria of the regulatory requirements. The control room doses due to this were bounded by those estimated for other DBAs.

Section 6.4.1.2 Radiological Consequences of Main Steam Line Break Outside Containment with a Concurrent Loss of Power

The EAB and LPZ offsite radiological consequences of a MSLB with a LOP section were calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. The dose assessment calculations are described in Reference 6-3 and are augmented by APS responses to NRC Questions 4, 5, and 6 of Attachment 2, Reference 6-29. The NSSS and core responses to this transient are detailed in Section 6.3.1.5.

The evaluations and conclusion documented by the NRC staff in the SER issued for Unit 2 Section 4.4.3 of Reference 6-3, the EAB and LPZ doses for this event were found to be within the acceptance criteria of the regulatory requirements. The control room doses due to this were bounded by those estimated for other DBAs.

Section 6.4.2 Radiological Consequences of Decrease in Heat Removal By The Secondary System

Section 6.4.2.1 Radiological Consequences of Feedwater System Pipe Breaks

The EAB and LPZ offsite radiological consequences of the FW system pipe breaks section was calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. The dose assessment calculations are augmented by APS responses to NRC Questions 4 (sections a through d) and 5 (sections a through j) of Attachment 2, Reference 6-29. The NSSS and core responses to this transient are detailed in Section 6.3.2.8.

The FWLB with a LOP-Long Term Cooling Event was reanalyzed and submitted for approval in Reference 6-22, note that Reference 6-22 supplemented the FWLB event and concluded no change in the event consequences.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.4 of Reference 6-3. The consequences at EAB and LPZ doses for this event were found to be within the acceptance criteria of the regulatory requirements. The control room doses due to this were bounded by those estimated for other DBAs.

Section 6.4.3 Radiological Consequences of Decrease in Reactor Coolant Flowrate

Section 6.4.3.1 Radiological Consequences of Single Reactor Coolant Pump Sheared Shaft with Loss of Offsite Power

The EAB and LPZ offsite radiological consequences of the single RCP rotor seizure with a LOP section were calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. The maximum source term was calculated based on the maximum fuel failure at a maximum radial peaking factor (Reference 6-29, Attachment 2, NRC Question 4d and Reference 6-30, Attachment 2, Page 13). Since the source term is proportional to fuel failure and peaking factor, the product of the two parameters are used as limiting parameter for the event. Therefore, for this event the calculated EAB and LPZ doses are less than 10 CFR Part 100 guidelines (i.e., 300 REM thyroid, 25 REM whole body) as long as:

Fuel Failure % X Radial peaking Factor $\leq$ 29

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.5 of Reference 6-3. The NSSS and core responses to this transient are detailed in Section 6.3.3.3. The EAB, LPZ, and control room doses for this event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.4 Radiological Consequences of Reactivity and Power Distribution Anomalies

Section 6.4.4.1 Radiological Consequences of Control Element Assembly Ejection

The methodology and regulatory criteria applicable for evaluating the radiological consequences of a CEA ejection accident are identical to those described in UFSAR Section 15.4.8. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 6-29.

The reload design process verifies that the consequences of the event are limiting for all core designs. Allowable doses for CEA ejection events with PUR remain within the values presented in UFSAR section 15.4.8. Total offsite doses from all sources remain within the applicable guidelines of 10 CFR Part 100. Therefore, there are no changes to the consequences of the CEA ejection event for operation at PUR. The NSSS and core responses to this transient are detailed in Section 6.3.4.1.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.6 of Reference 6-3. The EAB, LPZ, and control room doses for this event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.5 Radiological Consequences of Increase in Reactor Coolant System Inventory

No event in this category was reanalyzed for radiological consequences.

Section 6.4.6 Radiological Consequences of Decrease in Reactor Coolant System Inventory

Section 6.4.6.1 Radiological Consequences of Double - Ended Break of a Letdown Line Outside Containment

The methodology and regulatory criteria for evaluating the radiological consequences of a DBLLOCUS are identical to those described in UFSAR Section 15.6.2.3.2. Assumptions are consistent with SRP Section 15.6.2.

Analysis parameters and the NSSS response that were affected by the reanalysis of DBLLOCUS event as described in Section 6.3.6.2 of this report have been reviewed for impacts and new consequences are bounded by those reported in the UFSAR. The total 2-hour EAB thyroid dose remains bounded by existing UFSAR analysis and it continues to be within the SRP Section 15.6.2 criteria. The NSSS and core responses to this transient are detailed in Section 6.3.6.2.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.7 of Reference 6-3. The consequences of this event are within small fraction of (less than 10%) of applicable guidelines of 10 CFR Part 100.

Section 6.4.6.2 Radiological Consequences of Steam Generator Tube Rupture

Section 6.4.6.2.1 Radiological Consequences of Steam Generator Tube Rupture with a Concurrent Loss of Power and Single Failure

The EAB and LPZ offsite radiological consequences of the SGTR with a LOP and single failure section were calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 4.c and 5 of Attachment 2, Reference 6-29 and page 11 of Reference 6-30. The NSSS and core responses to this transient are detailed in Section 6.3.6.3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.8 of Reference 6-3. The EAB, LPZ, and control room doses for this event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.6.2.2 Radiological Consequences of Steam Generator Tube Rupture with a Concurrent Loss of Power

The radiological consequence of this event is bounded by event described in Section 6.4.6.2.1. There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.9 of Reference 6-3. The EAB and LPZ doses for this event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.6.3 Radiological Consequences of Loss-of-Coolant Accidents

The NSSS response to this transient is detailed in Section 6.1.

Section 6.4.6.3.1 Radiological Consequences of Small Break Loss-of-Coolant Accident

The methodology and the regulatory criteria applicable to the evaluation of SBLOCA radiological consequences are identical to those described in UFSAR Section 15.6.5.2. Assumptions are consistent with NRC Regulatory Guides 1.4 (Reference 6-31) and 1.77 (Reference 6-32) are employed along with guidelines from SRP Section 15.6.5 and changes as a result of ZIRLO™ cladding implementation (Reference 6-4 and Reference 6-6). Dose contributions from containment leakage, the power access purge, Engineered Safety Features (ESF) containment sump leakage, secondary releases from primary to secondary leakage, and initial secondary system inventory release are evaluated. The NSSS and core responses to this transient are detailed in Section 6.1.3.

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2, Reference 6-29.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.10 of Reference 6-3. The EAB and LPZ and control room doses for a SBLOCA were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.6.3.2 Radiological Consequences of Large Break Loss-of-Coolant Accident

The methodology and the regulatory criteria applicable to the evaluation of LBLOCA radiological consequences are identical to those described in UFSAR Section 15.6.5.6. Assumptions consistent with NRC Regulatory Guides 1.4 are employed along with guidelines from SRP Section 15.6.5. Dose contributions from containment leakage, the power access purge, ESF containment sump leakage, and back leakage to the Refueling Water Tank (RWT) are evaluated. Control room doses also include shine from the containment, accumulation on ESF filters, and direct cloud doses. The NSSS and core responses to this transient are detailed in Section 6.1.2.

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 1 of Attachment 2, Reference 6-29. As stated in Section 6.4.0, the value for Kr-88 in Table 6.4-1 of Reference 6-1, Attachment 6, should be 1.30E+08.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.10 of Reference 6-3. The EAB, LPZ, and control room doses for a LBLOCA were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

Section 6.4.7 Radiological Consequences of Radioactive Material Release from a Subsystem or Component

Section 6.4.7.1 Radiological Consequences of Waste Gas System Failure

The methodology and regulatory criteria for evaluating the radiological consequences of a waste gas system failure are identical to those described in UFSAR Section 15.7.1.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.11 of Reference 6-3. The radiological consequences are less than 1% of 10 CFR Part 100 guidelines.

Section 6.4.7.2 Radiological Consequences of Postulated Radioactivity Release Due to Liquid Containing Tank Failure

The methodology and regulatory criteria for evaluating the radiological consequences of a liquid-containing tank failure are identical to those described in UFSAR Sections 15.7.3 and 2.4.13.3. Assumptions are consistent with NUREG/CR-3332 (Reference 6-33) and SRP Section 15.7.3. The hypothetical event is characterized as a rapid release of the contents of a RWT to the environment. It is postulated that the tank contains its maximum inventory of 60 Ci per the Technical Requirements Manual (TRM, Reference 6-34) T3.10.200 and that no action is taken to mitigate the consequences of the event

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.12 of Reference 6-3. The existing analysis source term in the UFSAR is bounding for PUR. The most limiting spill in the perched zone that reaches the exclusion boundary will be below 1% of any Maximum Permissible Concentration in Water (MPC_w) listed in 10 CFR Part 20, Appendix B, Table II as required by the SRP.

Section 6.4.7.3 Radiological Consequences of Fuel Handling Accidents

The methodology and regulatory criteria for evaluating the radiological consequences of a fuel handling accident for PUR are identical to those described in UFSAR Section 15.7.4.1.3. Assumptions are consistent with NRC Regulatory Guide 1.25 and exemption as requested in Reference 6-23 and SRP Section 15.7.4. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 3 of Attachment 2, Reference 6-29.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.13 of Reference 6-3. The radiological consequences of a fuel handling accident inside and outside the containment are less than one-third of 10 CFR Part 100 guidelines as required by SRP 15.7.4.

Section 6.4.8 Radiological Consequences of Limiting Infrequent Events

The EAB and LPZ offsite radiological consequences of limiting infrequent events section were calculated using the methods and inputs described in Section 6.4.0 of this report and Reference 6-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by the response to NRC Questions 4.d and 5 (sections a through j) of Attachment 2, Reference 6-2. The NSSS and core responses to this transient are detailed in Section 6.3.8.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 4.4.13 of Reference 6-3. The EAB and LPZ doses for this event were found to be within the acceptance criteria of the regulatory requirements. The control room doses due to this were bounded by those estimated for other DBAs.

Section 6.5 Accident Source Term

LBLOCA source terms as result of the PUR core thermal power increase are addressed in this section. Non-LOCA accident source terms are described in Section 7.6 of this report and Reference 6-1.

Source Terms for evaluating the radiological consequences of postulated Design Basis Accidents (DBAs) (LBLOCA) have been developed in accordance with the recommendations of Regulatory Guide 1.4, TID-14844 (Reference 6-35), and NUREG-0737 (Reference 6-36). With the exception of long-lived isotopes, (e.g., Kr-85 and solids) the accident source term was generated using TID-14844 methodology. Because of relatively long half-life and high fuel burnup, the core inventory for long-lived isotopes was calculated using the code ORIGEN-S (Reference 6-37, SCAL 4.4 package). Activities for long-lived isotopes are conservatively based on assumed end of life at a cumulative value of 70,000 MWD/MTU fuel burnup and 5% enrichment.

Section 6.5.1 Large Break Loss-of-Coolant Accident Source Term

There are no changes to this section.

Section 6.5.2 Other Accidents Source Term

There are no changes to this section.

Section 6.6 References

This reference section as presented in Reference 6-1, Attachment 6, Section 6.6, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 6-1 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 6-2 APS letter 102-04847 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated October 11, 2002.

Reference 6-3 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 6-4 CENPD-404-P-A, Rev. 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November, 2001.

Reference 6-5 CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

Reference 6-6 CENPD-137, Supplement 2-P-A, Calculative Methods for the ABB CE Small Break LOCA Evaluation Model, April 1998.

Reference 6-7 Code of Federal Regulations, Title 10, Part 20 (old), Appendix B, Concentration in Air and Water Above Natural Background.
Code of Federal Regulations, Title 10, Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.
Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants
Code of Federal Regulations, Title 10, Part 100, Reactor Site Criteria.

Reference 6-8 APS letter 102-04699 to the NRC, 10 CFR 50.46(a)(3)(ii) 30-Day Report for Changes in LOCA/ECCS Performance Evaluation Models, May 3, 2002.

Reference 6-9 APS letter 102-04872 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated December 10, 2002.

Reference 6-10 APS letter 102-04974 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, July 25, 2003.

Reference 6-11 APS letter 102-04989 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, August 22, 2003.

Reference 6-12 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 12, August 2003.

Reference 6-13 APS letter 102-04828 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 27, 2002.

Reference 6-14 NRC letter, S.A. Richards (NRC) to P.W. Richardson (Westinghouse), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model' (TAC MA5660)," December 15, 2000.

Reference 6-15 NRC letter to Westinghouse, S.A. Richards (NRC) to P.W. Richardson (Westinghouse), Safety Evaluation of Topical Report CENPD-404-P, Revision 0, 'Implementation of ZIRLO™ Material Cladding in CE Nuclear Power Fuel Assembly Designs' (TAC No. MB1035), September 12, 2001.

Reference 6-16 NRC letter to Westinghouse, T.H. Essig (NRC) to I.C. Rickard (ABB CE), Acceptance for Referencing of the Topical Report CENPD-137(P), Supplement 2, 'Calculative Methods for the C-E Small Break LOCA Evaluation Model' (TAC No. M95687), December 16, 1997.

Reference 6-17 CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model," June 1980.

Reference 6-18 NRC letter, R.L. Baer (NRC) to A.E. Scherer (C-E), Staff Evaluation of Topical Report CENPD-254-P, July 30, 1979.

Reference 6-19 APS letter 102-04899 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 11, 2003.

Reference 6-20 APS letter 102-04837 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 6, 2002.

Reference 6-21 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on Replacement of Part-Length Control Element Assemblies (TAC Nos. MC0870, MC0871, and MC0872), dated March 23, 2003

Reference 6-22 APS letter 102-04866 to the NRC, Supplement to Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated November 21, 2001.

Reference 6-23 APS letter 102-04990 to the NRC, Request for Facility Operating License Amendment – Internal Fuel Pin Pressure Criteria for Fuel Handling Accident Safety Analysis, dated August 22, 2003.

Reference 6-24 APS letter 102-04664 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 13, 2002.

Reference 6-25 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on the Core Protection Calculator System Upgrade (TAC Nos. MB6726, MB6727, and MB6728), dated October 24, 2003.

Reference 6-26 U. S. Nuclear Regulatory Commissions Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

Reference 6-27 APS letter 102-04836 to the NRC, Request for a License Amendment to Revise the Peak Linear Heat Rate Safety Limit, Technical Specification 2.1.1.2, dated September 6, 2002.

Reference 6-28 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments on Peak Fuel Centerline Temperature Safety Limit (TAC Nos. MB6328, MB6329, and MB6330), dated December 2, 2002.

Reference 6-29 APS letter 102-04835 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 4, 2002.

Reference 6-30 APS letter 102-04954 to the NRC, Results of Review for Factual Accuracy in Draft Safety Evaluation Report (SER), dated June 10, 2003.

Reference 6-31 Regulatory Guide 1.4, "Assumptions Used for Evaluating the Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974.

Reference 6-32 Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Revision 0, May 1974.

Reference 6-33 NUREG/CR 3332, "Radiological Assessment, A Textbook on Environmental Dose Analysis, Parts 1 and 2," dated September 1, 1983.

Reference 6-34 Technical Requirements Manual (TRM), for Palo Verde Nuclear Generating Station, Units 1, 2, 3, Revision 24, July 25, 2003.

Reference 6-35 TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

Reference 6-36 NUREG-0737, November 1980, Clarification of TMI Action Plan Requirements," and Supplement 1 to NUREG-0737, December 17, 1982, "Requirements for Emergency Response Capability," (Generic Letter 82-33).

Reference 6-37 Oak Ridge National Laboratory, "ORIGEN2 Isotope Generation and Depletion Code" CCC-371, July 1980.
Approved for St. Lucie Plant, Unit No. 2 (Docket 50-389), in the safety evaluation report related to Amendment 21 dated May 29, 1987.

Section 7 NUCLEAR FUEL

This section is contained in Reference 7-2, Attachment 6. Response to NRC Question 11 of Attachment 2, Reference 7-1, provided information on corresponding sections of the UFSAR and SRP to subsections of this section.

Section 7.1 Core Thermal-Hydraulic Design

This section is applicable to Units 1 and 3 as augmented by response to NRC Question 19 of Attachment 2, Reference 7-3. Note that the current reload methods include consideration of ZIRLO™ cladding material, consistent with Reference 7-4, for Units 1, 2, and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the Safety Evaluation Report (SER) issued for Unit 2 in Section 5.1 of Reference 7-5.

Section 7.1.1 Departure from Nucleate Boiling Analysis

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 20, 21, and 22, Reference 7-3, and NRC Question 2 of Attachment 2, Reference 7-1.

Section 7.1.2 Effects of Fuel Rod Bowing on Departure from Nucleate Boiling Ratio Margin

There are no changes to this section.

Section 7.2 Core Design

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 19, Reference 7-3 and NRC Question 11, Reference 7-1.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.2 of Reference 7-5.

Section 7.3 Fuel Rod Design and Performance

Note that the current reload methods include consideration of ZIRLO™ cladding material, consistent with Reference 7-4 for Units 1, 2, and 3. Reference 7-6 evaluated fuel design and performance with ZIRLO™ cladding.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3 of Reference 7-5.

Section 7.3.1 Rod Cladding Collapse

This section is applicable to Units 1 and 3 as amended by the cladding collapse evaluation for ZIRLO™ provided in Reference 7-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.1 of Reference 7-5.

Section 7.3.2 Clad Fatigue

This section is applicable to Units 1 and 3 as amended by the clad fatigue evaluation for ZIRLO™ provided in Reference 7-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.2 of Reference 7-5.

Section 7.3.3 Clad Stress and Strain

This section is applicable to Units 1 and 3 as amended by the clad stress and strain evaluation for ZIRLO™ provided in Reference 7-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.3 of Reference 7-5.

Section 7.3.4 Rod Maximum Internal Pressure

This section is applicable to Units 1 and 3 as amended by the rod maximum internal pressure evaluation for ZIRLO™ provided in Reference 7-6. Note that the fuel handling accident evaluation in Section 6.4.7.3 provides additional information on maximum internal pressure considerations.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.4 of Reference 7-5.

Section 7.3.5 Cladding Waterside Corrosion

This section is applicable to Units 1 and 3 as amended by the cladding waterside corrosion evaluation for ZIRLO™ provided in Reference 7-6.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.5 of Reference 7-5.

Section 7.3.6 Conclusions

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.3.6 of Reference 7-5. Reference 7-6, remains applicable to Units 1 and 3 for both Zircaloy and ZIRLO™ fuel.

Section 7.4 Heat Generation Rates

There are no changes to this section.

Section 7.5 Neutron Fluence

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 17, Reference 7-3, and NRC Question 15, Reference 7-1, and information provided in Reference 7-7.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 5.4 of Reference 7-5.

Section 7.6 Source Terms

Section 7.6.1 Expected Source Term

There are no changes to this section.

Section 7.6.2 Design Source Term (1% Failed Fuel Condition Equilibrium Activities)

There are no changes to this section.

Section 7.6.3 Other Isotopic Source Terms

There are no changes to this section.

Section 7.6.4 Conclusions

In summary, the source term reported in UFSAR Section 11.1 remains bounding and conservative for Units 1 and 3 PUR. There are no changes to this section.

Section 7.7 References

This reference section as presented in Reference 7-2, Attachment 6, Section 7.7, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 7-1 APS letter 102-04899 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 11, 2003.

Reference 7-2 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 7-3 APS Letter 102-04847 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated October 11, 2002.

Reference 7-4 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments RE: Technical Specification 5.6.5b, Core Operating Limits Report (COLR) and use of ZIRLO™ Cladding Material, (TAC Nos. MB3373, MB3374, and MB3375), dated March 12, 2002.

Reference 7-5 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 7-6 CENPD-404-P-A, Rev. 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November, 2001.

Reference 7-7 APS letter 102-04700 to the NRC, APS' Response to the Information Requested Regarding Reactor Vessel Material Surveillance Capsule Reports, dated May 8, 2002.

Reference 7-8 NUREG-0017, Revision 1, dated April 1, 1985, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, PWR Gale-Code.

Section 8 BALANCE OF PLANT DESCRIPTION

This section is contained in Reference 8-2, Attachment 6. There is no change to this section that would affect the evaluation conducted by the NRC staff in the Safety Evaluation Report (SER) issued for Unit 2 in Section 6.1 of Reference 8-1.

Section 8.1 Balance of Plant Program Overview

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 8 of Attachment 2, Reference 8-3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.1 of Reference 8-1.

Section 8.2 Auxiliary Feedwater System

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.2 of Reference 8-1.

Section 8.3 Condensate and Feedwater

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.3 of Reference 8-1.

Section 8.3.1 System Description

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.3.2 Condensate and Feedwater Pumps

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.3.3 Heater Drain Pumps

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.3.4 Low Pressure Feedwater Heaters

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.3.5 High Pressure Feedwater Heaters

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.4 Circulating Water

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.4 of Reference 8-1.

Section 8.5 Main Turbine

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.5 of Reference 8-1.

Section 8.6 Main Turbine Auxiliaries

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.6 of Reference 8-1.

Section 8.7 Main Generator and Auxiliaries

There are no changes to this section.

Section 8.8 Main Steam

This section describes the evaluation of the main steam system from the outlet of the SG nozzle to the turbine stop valves. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.7 of Reference 8-1.

Section 8.8.1 Main Steam Safety Valves

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 1 of Attachment 2, Reference 8-5, NRC Question 8 of Attachment 2, Reference 8-3 and NRC Question 12 of Attachment 2, Reference 8-4.

Section 8.8.2 Atmospheric Dump Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.8.3 Main Steam Isolation Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.8.4 Main Steam Isolation Valve Bypass Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3. The plant modification described in Section 9.1 (Reference 8-2) will be performed for Units 1 and 3.

Section 8.8.5 Turbine Bypass Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.8.6 Main Steam Traps

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 8 and 12 of Attachment 2, Reference 8-3.

Section 8.8.7 Feedwater Isolation Valves

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 8 of Attachment 2, Reference 8-3.

Section 8.8.8 Main Steam System Summary of Conclusions

There are no changes to this section.

Section 8.9 Miscellaneous Cooling Water Systems

Section 8.9.1 Plant Cooling Water

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.8.1 of Reference 8-1.

Section 8.9.2 Turbine Cooling Water

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.8.2 of Reference 8-1.

Section 8.9.3 Nuclear Cooling Water

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.8.3 of Reference 8-1.

Section 8.9.4 Essential Cooling Water

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.8.4 of Reference 8-1.

Section 8.9.5 Spent Fuel Pool Cooling and Cleanup System

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 2.a, 2.b, 2.c, 2.d, 2.e, 3, 4.a, 4.b, and 4.c of Attachment 2, Reference 8-5.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.8.5 of Reference 8-1.

Section 8.10 Miscellaneous Mechanical Reviews

Section 8.10.1 Heating, Ventilation, and Air Conditioning Systems

There are no changes to this section.

Section 8.10.1.1 Containment Heating, Ventilation, and Air Conditioning

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2, Reference 8-5.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.9.1.1 of Reference 8-1.

Section 8.10.1.2 Auxiliary Building Ventilation

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2, Reference 8-5.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.9.1.2 of Reference 8-1.

Section 8.10.1.3 Turbine Building Heating, Ventilation, and Air Conditioning System

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2, Reference 8-5.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.9.1.3 of Reference 8-1.

Section 8.10.1.4 Control Building Heating, Ventilation, and Air Conditioning System

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 5 of Attachment 2, Reference 8-5.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.9.1.4 of Reference 8-1.

Section 8.11 Water Chemistry

Section 8.11.1 Steam Generator Blowdown Processing Systems

There are no changes to this section.

Section 8.11.2 Primary and Secondary Water Chemistry

There are no changes to this section.

Section 8.12 Secondary System Piping and Valves

Re-analyses of secondary side piping and components confirm that requirements of the ASME Code have been met. This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 8 and 9 of Attachment 2, Reference 8-3.

Section 8.13 Low Temperature Overpressure Protection

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 15 of Attachment 2, Reference 8-6 and Reference 8-7.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.10 of Reference 8-1.

Section 8.13.1 Input Parameters and Assumptions

There are no changes to this section.

Section 8.13.2 Acceptance Criteria for Analyses/Evaluations

There are no changes to this section.

Section 8.13.3 Results and Conclusions

There are no changes to this section.

Section 8.14 Miscellaneous Electrical Reviews

Specific analysis for Unit 1 and 3 electrical systems was performed and found bounded by the system design. There are no changes to this section.

Section 8.14.1 Grid Stability

A study was performed to verify grid stability with increased generation capability. The study verified maximum generating levels for the generators in the PVNGS area from either a maximum bucking or a maximum boosting condition to ensure that no stability problems would be encountered. A transmission system operating procedure controls the level of power generation in the PVNGS area to ensure that the safe levels are not exceeded. When operated within the procedural limits, all single-contingency disturbances proved to be stable and within the study's criteria, even with an additional 7% megawatt generation above the projected generation capability associated with PUR in Units 1, 2, and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.1 of Reference 8-1.

Section 8.14.2 Main Power Transformers

Passing air through banks of oil coolers dissipates the transformer generated heat. Although not required by the PUR analysis, the main transformers cooling will be modified to increase reliability.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.2 of Reference 8-1.

Section 8.14.3 Unit Auxiliary Transformer

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.2 of Reference 8-1.

Section 8.14.4 Startup Transformers

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.2 of Reference 8-1.

Section 8.14.5 Diesel Generators

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.2 of Reference 8-1.

Section 8.14.6 Station Blackout Turbines

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 8.11.4 of Reference 8-1.

Section 8.14.7 Isophase Bus

There are no changes to this section.

Section 8.14.8 Reactor Coolant Pump Motors

There are no changes to this section.

Section 8.14.9 Condensate Pump Motors

There are no changes to this section.

Section 8.14.10 Heater Drain Pump Motors

There are no changes to this section.

Section 8.14.11 Breaker Coordination and Relay Settings

There are no changes to this section.

Section 8.15 Miscellaneous Instrumentation and Control Reviews

There are no changes to this section.

Section 8.15.1 Condensate Pump Minimum Flow Control

There are no changes to this section.

Section 8.15.2 Steam Generator Feedwater Pump Minimum Flow Control

There are no changes to this section.

Section 8.15.3 Heater Drains Control

There are no changes to this section.

Section 8.15.4 Condenser Hotwell Level Control

There are no changes to this section.

Section 8.16 Essential Spray Pond System

Note that the original reference for the computer code COPATTA-PV is provided in Reference 8-8. The plant modification described that is in Section 9.1 (Reference 8-2) will be installed in Units 1 and 3.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.12 of Reference 8-1.

Section 8.17 Conclusion

Current plant components can accommodate changes to the key plant operating conditions (steam flow, pressure, and temperature) affecting the Balance of Plant (BOP) system performance characteristics for PUR. Changes will be made to the MT control logic, Main Steam Isolation Valve (MSIV) bypass valves, and Essential Spray Pond System (ESPS) temperatures indicators (Section 9.1).

Section 8.18 References

This reference section as presented in Reference 8-2, Attachment 6, Section 8.18, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 8-1 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 8-2 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 8-3 APS Letter 102-04837 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 6, 2002.

Reference 8-4 APS Letter 102-04847 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated October 11, 2002.

Reference 8-5 APS letter 102-04828 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 27, 2002.

Reference 8-6 APS letter 102-04899 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 11, 2003.

Reference 8-7 APS letter 102-04700 to the NRC, APS' Response to the Information Requested Regarding Reactor Vessel Material Surveillance Capsule Reports, dated May 8, 2002.

Reference 8-8 COPATTA - PV A Class B SQA software (NE100), Version A1 -1 (SUN) / Version A1 - 2 (Windows). Bechtel Corporation, August 1993.

Section 9 MISCELLANEOUS TOPICS

This section is contained in Reference 9-1, Attachment 6.

Section 9.1 Modifications Required to Implement Power Uprate

The modifications required to implement Power Uprate (PUR) section is contained in Reference 9-1, Attachment 6. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 13 of Attachment 2, Reference 9-2 for the Main Steam Isolation Valve (MSIV) bypass valves modification.

The evaluations and conclusions reached in this section do not change from the conclusions reached in Power Uprate Licensing Report (PURLR) for Unit 2, and there is no change to this section that would affect the evaluation conducted by the NRC staff in the Safety Evaluation Report (SER) issued for Unit 2 in Section 8.11.2 of Reference 9-3.

Section 9.2 Integrated Leakage Rate Testing

The program established to implement the Integrated Leakage Rate Testing (ILRT) of the containment is required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. Option B specifies that Type A, B, and C tests be conducted to verify the integrity of the primary reactor containments. These tests will be conducted under conditions representing the design basis Loss-of-Coolant Accident (LOCA) containment peak pressure (P_a). Replacement Steam Generators (RSGs) and PUR results in an increase in P_a from 52.0 to 58.0 psig. Palo Verde has proposed that for Units 1 and 3, all required Type B and C tests will be completed at the higher P_a prior to operation at the increased power but Type A testing at the increased P_a will be deferred until the next regularly scheduled Type A test.

Section 9.2.1 Background

One of the conditions of all operating licenses for water-cooled power reactors is that primary reactor containments meet the containment leakage test requirements in either Option A or B of Appendix J to 10 CFR 50. These test requirements ensure that:

(a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the Technical Specifications and
(b) integrity of the containment structure is maintained during its service life.

Option B of Appendix J identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing. Palo Verde has implemented Option B.

Appendix J requires the performance of three types of tests:

1) Type A or "overall integrated leakage rate" tests which establish an overall containment leak rate from a summation of leakage through all potential

leakage paths including containment welds, valves, fittings, and components which penetrate containment;

2) "Type B Tests" intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:
   a) Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.
   b) Air lock door seals, including door operating mechanism penetrations, which are part of the containment, pressure boundary.
   c) Doors with resilient seals or gaskets except for seal-welded doors.
3) "Type C Tests" intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:
   a) Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;
   b) Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;
   c) Are required to operate intermittently under post-accident conditions

Section 9.2.2 Justification for Deferral of Type A Test at Increased P_a

Palo Verde Unit 2 has been Type A tested at the increased P_a and its test results at the higher pressure compare favorably with earlier tests conducted at lower pressures. Since the Unit 1 and 3 containment structures are identical to Unit 2 and have been subjected to similar test and maintenance regimes, similar performance in the increased P_a Type A test is reasonably assured.

The Palo Verde Nuclear Generating Station (PVNGS) consists of three identical pressurized water reactor units with identical reinforced, post tensioned concrete containment structures having a design pressure of 60 psig. All three containments were subjected to a structural integrity test at or above the design pressure (i.e., 60 psig) during pre-operational testing and have been subjected to periodic Type A, B and C leakage testing as required by 10 CFR 50, Appendix J. Type A tests conducted to date were completed at the calculated design basis accident containment peak pressure (P_a). Calculated P_a was 49.5 psig for Type A tests conducted early in plant life. P_a was later revised to 52.0 psig for all three units and then to 57.85 psig for Unit 2 following RSG and PUR. The most recent type A tests were conducted at approximately 52.0 psig in Units 1 and 3 and 58.0 psig in Unit 2. The results of the Unit 2 Type A test conducted in November, 2000 at 58.4 psig compared favorably with previous testing conducted at 49.5 psig. For example, the Unit 2 Type A test conducted in June of 1988 at 49.5 psig produced a leakage rate at the 95% Upper Confidence Limit (UCL) of 0.599% of L_a. The November, 2000 test at 58.4 psig produced a lower integrated leakage at the 95% UCL of 0.415% of L_a. L_a is defined as the maximum allowable leakage rate at pressure P_a as specified in the Technical Specifications. (L_a) for Palo Verde is 0.1% of containment air weight per day.

All individual containment penetrations in Units 1 and 3 subject to Type B and C testing will be tested at the increased P_a prior to operation at the higher power rating. Since industry experience has demonstrated that Type A testing rarely identifies leakage paths that would not be detectable by Type B and C testing, this provides additional assurance that the results of Type A tests conducted at the higher Pa would be acceptable.

A survey was performed in early 1994 and represented the NEI (known as NUMARC at that time) input used in NUREG-1493 (Reference 9-4). In this survey, the data from 144 ILRT Type A tests was collected. Reported in NUREG-1493 were 23 ILRT failures. Of the twenty-three ILRT failures:

- 14 failures were due to addition of Type B and C (local leak rate testing identified) leakage penalties, and therefore did not represent leak paths detected by the Type A test.
- 4 failures were due to steam generator in-leakage. The steam generator leak paths are identifiable during startup and normal operation and therefore do not represent leak paths that would be revealed only by Type A testing.
- 2 failures were due to ILRT line-up errors, and did not constitute valid leak paths.
- 1 failure was due to a discrepancy in a verification test and did not constitute a valid leak path.
- 2 failures were due to failures, which should have been indicated by the local leak rate testing programs. It is likely that these discrepancies would have been corrected at the next local leak rate test, and therefore do not represent leak paths that would be revealed only by a Type A test.

A second Type A test survey was performed in the fall of 2001. In the second survey, data was collected from 58 plants (91 units), reporting 38 ILRT (Type A) tests performed with one failure. The one ILRT identified failure should have been indicated by the local leak rate testing program therefore does not represent a leak path that would be revealed only by a Type A test. The failure was caused by contamination of a penetration with construction debris during a modification, which passed the post-modification LLRT. However, because of the contamination the failure would have most likely been identified by subsequent LLRT's had the ILRT not been conducted.

There would be at most a negligible increase in risk resulting from deferral of Type A testing at the increased Pa. Per Table 9.2-1 Type A test history in all PVNGS units has been within regulatory requirements. Acceptable Type A tests have been conducted within the last 5 years at 52.0 psig (in Units 1 and 3) and 58.4 psig (in Unit 2). Testing conducted in November of 1999 in Unit 1 and April of 2000 in Unit 3 produced results of 0.554% and 0.513% of L_a respectively. It would require leakage rates many times this large to affect the Large Early Release Frequency (LERF) as calculated in the Level 2 Probabilistic Risk Assessment (PRA) (Reference 9-4).

Table 9.2-1
PVNGS ILRT History

		Test Results in % of L_a		
ILRT	Date	Unit 1	Unit 2	Unit 3
Pre-op	U1-12/82 U2- 2/85 U3- 9/86	0.183% @ 49.5 psig	0.292% @ 49.5 psig	0.521% @ 49.5 psig
1st Interval	U1-5/86 U2- 6/88 U3- 5/91	0.664% @ 49.5 psig	0.668% @ 49.5 psig	0.620% @ 49.5 psig
2nd Interval	U1-2/90 U2- 12/91 U3- 4/00	0.67% @ 49.5 psig	0.31% @ 49.5 psig	0.513% @ 52.8 psig
3rd Interval	U1-11/99 U2- 11/00 U3- 4/09	0.554% @ 52.12 psig	0.415% @ 58.4 psig	Not yet performed

Several one-time Type A test interval extension requests accepted by the NRC have used an estimated leakage rate of 35 L_a The leak rate is to conservatively represent the leakage rate associated with a large early release as calculated in the Level 2 PRA. However, the definition of LERF is generally given as the exchange of a single containment volume before the effective implementation of the offsite emergency response and public protective actions. In turn, public protective actions are generally assumed to be taken approximately 2 to 4 hours following a core damage event. The exchange of a single containment volume within a 4- hour period corresponds to a leakage rate of 600% of containment volume per day or 6000 times L_a (PVNGS L_a is 0.1% per day).

Section 9.3 Loss-of-Coolant Accident Hydrogen Generation

The post-LOCA hydrogen generation analysis was performed for Unit 1 and 3 PUR. The existing containment aluminum and zinc inventories remain unchanged by PUR. The PUR post- LOCA containment temperature profile is discussed in Section 6.2.2. A core wide oxidation rate of 1.0% was used to predict the quantity of hydrogen released because of the zirconium metal water reaction as a result of implementation of ZIRLO™ cladding refer to Section 6.1. Consistent with the licensing basis, the hydrogen recombiners are assumed to be placed into service at 100 hours. The analysis concluded that the peak bound hydrogen concentration remains less than 3.99% by volume.

As discussed in UFSAR Section 6.2.5 (Reference 9-5), and as accepted by the NRC as documented in the Standard Review Plan (SRP) (Reference 9-6), under postulated LOCA conditions, the Reactor Drain Tank (RDT) room may become an essentially closed room with the only venting occurring through an annular opening in the ceiling. The potential therefore exists for accumulation of hydrogen in the RDT room.

The maximum hydrogen concentration has been analyzed in the RDT room using the same NRC reviewed methodology as the original design (Reference 9-7). The results of the analysis show that the gas plume exiting the room remains well below the combustible limit utilizing a conservative post-LOCA pressure, temperature, and bulk hydrogen profiles that bound those predicted to occur at PUR conditions. The results are consistent with the original NRC established limit. Additionally, the RDT room has been designed to eliminate all potential ignition sources within the room.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.1 of Reference 9-3.

Section 9.4 Radiological Assessment

Section 9.4.1 Description

There are no changes to this section.

Section 9.4.2 Scope of Review

There are no changes to this section.

Section 9.4.3 Design Requirements

There are no changes to this section.

Section 9.4.4 Assumptions

There are no changes to this section.

Section 9.4.5 Method of Evaluation

There are no changes to this section.

Section 9.4.6 Summary of Evaluations

Section 9.4.6.1 Normal Plant Shielding

There are no changes to this section.

Section 9.4.6.2 Normal Offsite Releases

There are no changes to this section.

Section 9.4.6.3 Radioactive Waste Management Systems

There are no changes to this section.

Section 9.4.6.4 Post-Accident Shielding

Accident source terms consistent with the requirements of NUREG-0737 have been reviewed for PUR. There are no changes to this section.

Section 9.4.6.5 Post-Accident Vital Area Doses

There are no changes to this section.

Section 9.4.6.5.1 Control Room

Control room habitability requirements are addressed in Section 9.10 of this report and Reference 9-1, Attachment 6.

Section 9.4.6.5.2 Technical Support Center

There are no changes to this section.

Section 9.4.6.5.3 Emergency Operations Facility

There are no changes to this section.

Section 9.4.6.5.4 Hydrogen Recombiner Area

There are no changes to this section.

Section 9.4.6.5.5 Sampling System

APS has received a license amendment for the elimination of the Post Accident Sampling System (PASS) (Reference 9-8). Therefore, NUREG-0737 Section II.B.3 is no longer applicable.

Section 9.4.7 Summary of Conclusions

No changes to Structures, Systems, or Components (SSCs) are required to provide adequate radiation protection for operators or the public during normal and post-accident conditions. The plant shielding design remains bounding for PUR.

Section 9.5 Electrical Equipment Qualification

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 6.11.5 of Reference 9-3. The In-Core Instrument (ICI) connectors and non-standard Raychem splices have been qualified for PUR conditions.

Section 9.5.1 Scope of Review

There are no changes to this section.

Section 9.5.2 Summary of Evaluations

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 4 of Attachment 2, Reference 9-9.

Section 9.5.3 Summary of Conclusions

There are no changes to this section.

Section 9.6 Valve Program

There are no changes to this section.

Section 9.7 Fire Protection Program

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.2 of Reference 9-3.

Section 9.8 Probabilistic Risk Assessment

This license amendment request is not being submitted as a risk-informed request. There are no changes to this section.

Section 9.9 Environmental Impact Evaluations

An evaluation of the Final Environmental Statement (FES, Reference 9-10) was completed for PUR. There are no changes to this section.

Section 9.10 Control Room Habitability

Section 9.10.1 Control Room Radiological Design

There are no changes to this section.

Section 9.10.1.1 Essential System Parameters used in Radiological Analysis

There are no changes to this section.

Section 9.10.2 Single Failure Applied to Control Room Habitability Analysis

There are no changes to this section.

Section 9.10.3 Control Room Radiological Assessment

There are no changes to this section.

Section 9.10.3.1 Radiological Parameters used for Control Room Evaluation

This section is applicable to Units 1 and 3 as augmented by the responses to NRC Question 9.a of Attachment 2, Reference 9-11.

This section was modified for breathing rates. For calculating the exposure to control room personnel, occupancy factors and breathing rates are as follows:

0 to 24 hours	occupancy factor = 1	breathing rate = 3.47E-04 m^3/sec
1 to 4 days	occupancy factor = 0.6	breathing rate = 3.47E -04 m^3/sec
4 to 30 days	occupancy factor = 0.4	breathing rate = 3.47E -04 m^3/sec

Radioactivity concentration (Ci/m^3) in the radioactive cloud surrounding the control room is the product of the building leak rate (Ci/sec) and the control room atmospheric dispersion factor, X/Q (sec/m^3). X/Q values for the exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are presented in UFSAR Section 2.3. A tabulation of control room X/Q's is presented in UFSAR Table 15B-5.

Section 9.10.3.2 Results and Conclusions

As shown in Table 6.4.7-1 of UFSAR, the most limiting organ dose (thyroid) accident is the Control Element Assembly (CEA) Ejection event. For all events, the analyses assume a total of 61 scfm unfiltered air in leakage (Reference 9-13). For all events, the consequences to control room operators are less than the criteria provided in SRP Section 6.4 and GDC 19 of 10 CFR 50 Appendix A.

Section 9.10.4 Testing and Conformation of Design Bases Parameters

The control room essential filtration units are tested per the requirements of Technical Specification 3.7.11 (Reference 9-12). For Unit 2, a special, integrated pressure boundary leak test was performed to validate the total unfiltered in leakage assumption used in the control room habitability analysis. The results of this validation test demonstrated that the design assumption of 61 scfm unfiltered in leakage bounds the actual as-built plant condition. No additional testing will be conducted in Units 1 and 3 as stated in Reference 9-13.

Section 9.11 Natural Circulation Cooldown Analysis

There are no changes to this section.

Section 9.12 Impact of Increased Power on Operations

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.12.1 Impact on the Control Room

The Safety Parameter Display System (SPDS) will be modified for the larger Steam Generators (SGs) (i.e., larger RCS volume, larger SG volume, etc.). Operators will be trained on these plant changes before operation at PUR per the requirements of administrative control procedures. There are no changes required to the Qualified Safety Parameter Display System (QSPDS) as a result of PUR in any of the PVNGS Units.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.12.2 Impact on Operations Department Procedures

This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 10 of Attachment 2, Reference 9-14.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.12.3 Impact on the PVNGS Simulators

Currently one of the PVNGS simulators has been revised to reflect PUR conditions. There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.12.4 Impact on Training

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.13 Testing

The PUR and larger SGs result in design changes/modifications. In order to perform the required retests to verify the design basis parameters and to verify continued safe operation, an Integrated Startup Test Plan will be developed. This section is applicable to Units 1 and 3 as augmented by the response to NRC Question 2 of Attachment 2, Reference 9-15 and NRC Questions 9 and 12 of Reference 9-2.

Section 9.14 Human Factors

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.6 of Reference 9-3.

Section 9.15 High Energy Line Breaks

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.3 of Reference 9-3.

Section 9.16 Erosion/Corrosion Program

As for Unit 2, the existing Erosion/Corrosion Program inspection acceptance criteria will be maintained. This section is applicable to Units 1 and 3 as augmented by the responses to NRC Questions 5 and 6 of Attachment 2, Reference 9-16.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.4 of Reference 9-3.

Section 9.17 Flooding

There are no changes to this section.

Section 9.17.1 Containment Sump pH and Containment Flooding

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.5.1 of Reference 9-3.

Section 9.17.2 Outside Containment Flooding

There are no changes to this section.

There is no change to this section that would affect the evaluation conducted by the NRC staff in the SER issued for Unit 2 in Section 7.5.2 of Reference 9-3.

Section 9.18 Computer Code Applications

There are no computer code changes for Units 1 and 3

Section 9.19 References

This reference section as presented in Reference 9-1, Attachment 6, Section 9.18, is applicable to Units 1 and 3. The references are updated and augmented by the following:

Reference 9-1 APS letter 102-04641 to the NRC, Request for a License Amendment to Support Replacement of Steam Generators and

Uprated Power Operations for PVNGS Unit 2, dated December 21, 2001.

Reference 9-2 APS Letter 102-04837 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 6, 2002.

Reference 9-3 NRC letter to APS, Palo Verde Nuclear Generating Station, Unit 2 - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696), dated September 29, 2003.

Reference 9-4 EPRI TR-104285, Revision 1, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.

Reference 9-5 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 12, August 2003.

Reference 9-6 U. S. Nuclear Regulatory Commission Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

Reference 9-7 NUREG-0857, Safety Evaluation Report related to the operation of PVNGS Units 1, 2, and 3, Supplement 4.

Reference 9-8 NRC letter to APS, Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments Re: Eliminate the Requirements for the Post Accident Sampling System (PASS) using the Consolidated Line Item Improvement Process (CLIP) (TAC Nos. MB2291, MB2292, and MB2293), dated September 28, 2001.

Reference 9-9 APS letter 102-04664 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 13, 2002.

Reference 9-10 NUREG-0841, dated, February 1, 1982, "Final Environmental Statement Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2 and 3."
NUREG-0036, dated, February 1, 1978, "Final Environmental Statement Related to the Construction of Palo Verde Nuclear Generating Station, Units 1, 2, and 3."
NUREG-75/078, dated September, 1975, "Final Environmental Statement (FES)."

Reference 9-11 APS letter 102-04835 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated September 4, 2002.

Reference 9-12 Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Technical Specifications, through Amendment No. 152, March 23, 2004.

Reference 9-13 APS letter 102-05018 to the NRC, PVNGS Units 1, 2 and 3 Docket Nos. STN 50-528/529/530 180-Day Response to NRC Generic Letter 2003-01: Control Room Habitability, dated December 5, 2003.

Reference 9-14 APS letter 102-04899 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated March 11, 2003.

Reference 9-15 APS letter 102-04847 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated October 11, 2002.

Reference 9-16 APS letter 102-04834 to the NRC, Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request, dated August 29, 2002.

Section 10 ACRONYMS

ACRONYM	DEFINITION
X/Q	atmospheric dispersion factor
°F	degrees Fahrenheit
ΔP	change in pressure
AC	Alternating Current
ADV	Atmospheric Dump Valve
AL	Analytical Limit
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOR	Analyses of Record
APS	Arizona Public Service
ASME	American Society of Mechanical Engineers
AV	Allowable Value
BLPB	Branch Line Pipe Break
BOP	Balance of Plant
BWR	Boiling Water Reactor
CE	Combustion Engineering
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CENTS	Combustion Engineering Nuclear Transient Simulator
CFR	Code of Federal Regulations
Ci	Curie(s)
CLIP	Consolidated Line Item Improvement Process
COLR	Core Operating Limits Report
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CSB	Core Support Barrel
CSS	Containment Spray System
CVCS	Chemical And Volume Control System
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DBLLOCUS	Double-Ended Break of the Letdown Line Outside Containment Upstream of the letdown line control valve
DEDLSB	Double-Ended Discharge Leg Slot Break
DEG/PD	Double-Ended Guillotine Break in Reactor Coolant Pump Discharge Leg
DEHLSB	Double-Ended Hot Leg Slot Break
DESLSB	Double-Ended Suction Leg Slot Break
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EM	Evaluation Method
ESF	Engineered Safety Features
ESPS	Essential Spray Pond System

ACRONYM	DEFINITION
FES	Final Environmental Statement
ft	feet or foot
FW	Feedwater
FWLB	Feedwater Line Break
GDC	General Design Criterion
HJTC	Heated Junction Thermocouple
Hz	Hertz
HZP	Hot Zero Power
ICI	In-Core Instrument
ID	Inadvertent Deboration
ILRT	Integrated Leak Rate Test
In	Inch(es)
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
ISA	Instrument Society of America
kW	Kilowatt
LBLOCA	Large Break Loss-of-Coolant Accident
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOF	Loss of Flow
LOP	Loss of Offsite Power
LPD	Local Power Density
LPZ	Low Population Zone
LSS	Lower Support Structure
LSGP	Low Steam Generator Pressure
m	meter
min	minute
MPC	Maximum Permissible Concentration
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
MTU	Metric Ton of Uranium
MWD	Megawatt Days
MW_e	Megawatts Electrical
MW_t	Megawatts Thermal
N/A	Not Applicable
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
OD	Outside Diameter
PASS	Post Accident Sampling System
PD	Reactor Coolant Pump Discharge Leg
PLCS	Pressurizer Level Control System
PLHGR	Peak Linear Heat Generation Rate

ACRONYM	DEFINITION
PRA	Probabilistic Risk Assessment
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PSV	Pressurizer Safety Valve
PUR	Power Uprate
PURLR	Power Uprate Licensing Report
PVMP	Precritical Vibration Monitoring Program
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
QSPDS	Qualified Safety Parameter Display System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDT	Reactor Drain Tank
REM	Roentgen Equivalent Man
RSG	Replacement Steam Generator
R-t-P	Return to Power
RVI	Reactor Vessel Internals
RVLMS	Reactor Vessel Level Monitoring System
RWT	Refueling Water Tank
SBCS	Steam Bypass Control System
SBLOCA	Small Break Loss-of-Coolant Accident
scfm	standard cubic feet per minute
sec	second (time)
SER	Safety Evaluation Report
SFWLB	Small Feedwater Line Break
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SGTRLOP	Steam Generator Tube Rupture, with a Loss of AC Power
SI	Safety Injection
SPDS	Safety Parameter Display System
SQA	Software Quality Assurance
SRP	Standard Review Plan
SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
TRM	Technical Requirements Manual
UCL	Upper Confidence Limit
UFSAR	Updated Final Safety Analysis Report
UGS	Upper Guide Structure
UGSSP	Upper Guide Structure Support Plate
WEC	Westinghouse Electric Corporation

ATTACHMENT 5

WESTINGHOUSE ELECTRIC COMPANY PROPRIETARY INFORMATION IN SUPPORT OF PVNGS-1 AND 3 POWER UPRATE SUBMITTAL
(includes non-proprietary version)

Attachment 5

Non-Proprietary Version

Table 1

SER Limitations/Constraints for the 1999 EM Topical Report

Topical Report:	CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.
SER:	S.A. Richards (NRC) to P.W. Richardson (Westinghouse), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model' (TAC MA5660)," December 15, 2000.

(a,c)

No.	Limitation/Constraint	Applicability	Conformance [Yes/No]	Comments

Table 1

SER Limitations/Constraints for the 1999 EM Topical Report

Topical Report:	CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.
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(a,c)

No.	Limitation/Constraint	Applicability	Conformance [Yes/No]	Comments

Table 2

SER Limitations/Constraints for the ZIRLO™ Topical Report

(a,c)

<table>
<tr><td colspan="5">Topical Report: CENPD-404-P-A, Rev. 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.
SER: S.A. Richards (NRC) to P.W. Richardson (Westinghouse), "Safety Evaluation of Topical Report CENPD-404-P, Revision 0, 'Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs' (TAC No. MB1035)," September 12, 2001.</td></tr>
<tr><th>No.</th><th>Limitation/Constraint</th><th>Applicability</th><th>Conformance [Yes/No]</th><th>Comments</th></tr>
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</table>

ATTACHMENT 6

AFFIDAVIT FROM THE WESTINGHOUSE ELECTRIC COMPANY SUBMITTED IN ACCORDANCE WITH 10 CFR 2.390 TO CONSIDER ATTACHMENT 5 AS A PROPREITARY DOCUMENT

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

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

Direct tel: 412-374-4643
Direct fax: 412-374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-04-1858
June 24, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Attachment to Arizona Public Service Company Letter 102-05116

This Application for Withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of 10 CFR 2.390(b)(1), as amended, of the Commission's regulations. It pertains to proprietary information regarding compliance with NRC Safety Evaluation Report limitations and constraints for various Westinghouse methodologies as provided in the attachment to the subject letter.

In conformance with 10 CFR Section 2.390, Affidavit CAW-04-1858 accompanies this Application for Withholding and sets forth the basis on which the identified proprietary information may be withheld from public disclosure. The justification for claiming this information as proprietary is identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit. Accordingly, Westinghouse respectfully requests that the proprietary information contained in this transmittal be withheld from public disclosure.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse affidavit should reference this letter, CAW-04-1858, and should be addressed to the undersigned.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosure

bcc: J. S. Galembush (ECE 4-7A)
C. B. Brinkman, (Rockville, MD 20852)
M. Howard (Windsor)
RCPL Administrative Aide (ECE 4-7A)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA)

ss:

COUNTY OF ALLEGHENY)

Before me, the undersigned authority, personally appeared Norton L. Shapiro, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Norton L Shapiro

Norton L. Shapiro,
Chief Engineer
Westinghouse Nuclear Services

Sworn to and subscribed before me
this 24th day of June 2004.

Lorraine M. Piplica
Notary Public

My commission expires 12-14-07

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2007
Member, Pennsylvania Association Of Notaries

LORRAINE M. PIPLICA NOTARY PUBLIC COMMONWEALTH OF PENNSYLVANIA

(1) I, Norton L. Shapiro, depose and say that I am the Chief Engineer in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system for classification of proprietary information, which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is contained in the Attachment to Arizona Public Service Company letter 102-05116.

This information demonstrates compliance with NRC Safety Evaluation Report (SER) limitations/constraints for various Westinghouse methodologies, including:

(a) Identification of the applicable SER for the methodology,

(b) Identification of the SER limitations/constraints,

(c) Basis for compliance with each limitation/constraint.

(vii) Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation..

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology that was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide technical and licensing services without incurring commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

Compiling the information required considerable Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, a significant manpower effort, having the requisite talent, experience, and knowledge of Westinghouse methodologies would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

That information which is proprietary in the proprietary version is contained within brackets in order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the identification and protection of proprietary information voluntarily transmitted to the NRC. Such proprietary information has been deleted in the non-proprietary version, leaving only the brackets. The justification for claiming the information designated as proprietary is indicated in both versions by means of superscript letters (a) through (f) following the brackets enclosing each item identified as proprietary. These letters refer to the types of information Westinghouse customarily holds in confidence as identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).