November 2, 2005

James M. Levine
Executive Vice President, Generation
Mail Station 7602
Arizona Public Service Company
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Phoenix, Arizona 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - DRAFT SAFETY EVALUATION RE: REPLACEMENT OF STEAM GENERATORS AND UPRATED POWER OPERATIONS AND ASSOCIATED ADMINISTRATIVE CHANGES (TAC NOS. MC3777, MC3778, AND MC3779)

Dear Mr. Levine:

By application dated July 9, 2004, as supplemented by letters dated June 2, June 3 (two letters), June 17, July 9 (two letters), July 19, August 3, September 29, October 21, and November 1, 2005, Arizona Public Service Company (the licensee), requested to amend the Technical Specifications for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 in support of the replacement steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94 percent increase from the current 3876 MWt.

The licensee requested the opportunity to review the draft Safety Evaluation (SE) for technical accuracy. Since the amendment is complex and required comprehensive reviews in several technical disciplines, the NRC staff concurs with your request to verify that the factual information presented is accurate and complete.

Enclosed is the draft SE that the NRC staff has prepared to evaluate your amendment request. This draft SE has not had an editorial review at this time, as this will be done concurrently with your technical review. Please provide any comments that you may have, in writing, on the enclosed draft SE with regard to factual accuracy by November 11, 2005. This target date for your response has been discussed with Mr. Tom Weber of your staff. Should a situation occur which prevents you from meeting the aforementioned date, please contact me at (301) 415-3062.

Sincerely,

/RA/
Mel B. Fields, Senior Project Manager
Plant Licensing Branch G
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529, and STN 50-530

Enclosure: Draft SE

cc w/encl: See next page

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated July 9, 2004, as supplemented by letters dated June 2, June 3 (two letters), June 17, July 9 (two letters), July 19, August 3, September 29, October 21, and November 1, 2005, Arizona Public Service Company (the licensee), requested changes to the Technical Specifications (TSs) for Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3. The June 2, June 3 (two letters), June 17, July 9 (two letters), July 19, August 3, September 29, October 21, and November 1, 2005, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 28, 2004 (69 FR 57980).

The proposed changes support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94 percent increase from the current 3876 MWt for PVNGS Unit 1 and PVNGS Unit 3. The amendments also make administrative changes to the PVNGS Unit 2 TSs so that the changed pages would apply to the three PVNGS units. Based on its review of this application, the Nuclear Regulatory Commission (NRC) staff categorized the application as a stretch power uprate (PUR). After implementation of the amendments, the 3876 MWt rated thermal power (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. Some of the proposed changes in the licensee's submittal are being made to accomplish the PUR, while others are needed both to accomplish the PUR and replacement steam generators (RSGs).

By application dated December 21, 2001, the licensee requested a similar 2.94-percent PUR for PVNGS Unit 2. The NRC approved the 2.94-percent PUR for PVNGS Unit 2 by License Amendment No. 149, dated September 29, 2003. Given the many commonalities between the PVNGS Unit 1, 2, and 3 design and licensing bases, the licensee utilized a similar approach for assessing the proposed PVNGS Unit 1 and Unit 3 PUR as that which was previously approved by the NRC staff for the PVNGS Unit 2 PUR.

This Safety Evaluation (SE) documents the NRC staff's evaluation of the licensee's analyses mentioned above. In its review, the NRC staff sought to determine whether or not the licensee's results are acceptable and demonstrate that the applicable design basis acceptance criteria will continue to be met during the PUR conditions with the RSGs. This SE contains reviews by NRC staff members from various technical disciplines, and is arranged by subject matter similar to those outlined in Attachment 4, "Power Uprate Licensing Report (PURLR)," of the licensee's July 9, 2004, amendment request.

1.1 Description of Proposed Amendment

The proposed amendment would allow operation of PVNGS Unit 1 and Unit 3 up to a maximum reactor core power level of 3990 MWt, an increase of 2.94 percent 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. Specifically, the following Facility Operating Licenses and Technical Specification (TS) changes are requested to support the increased power operation:

<u>Facility Operating License No. NPF-41</u>
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.

<u>Facility Operating License No. NPF-74</u>

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 MIt, 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.

<u>TS 1.1 Definitions</u>
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.

<u>TS 3.3.1 Reactor Protective System (RPS) Instrumentation - Operating</u>
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 RSGs affect this specification.

TS 3.3.2 RPS Instrumentation - Shutdown

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 Mt 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.

TS 3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation

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.

TS 3.4.1 Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

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 EF at 0% power, decreasing linearly to 564 EF at 100% power. The upper limit line of Figure 3.4.1-1, in the current TS, decreases linearly from 570 EF at 0% power to 568 EF at 30% power. At 30% power the current figure then decreases linearly from 568 EF to 560 EF 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.

TS 3.7.1 Main Steam Safety Valves (MSSVs)

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.

TS 5.5.16 Containment Leakage Rate Testing Program

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

Bases for Technical Specifications 3.6.1, 3.6.2, 3.6.4, 3.6.6 and 3.7.1
Bases would be revised to reflect these changes described above.

2.0 NUCLEAR STEAM SUPPLY SYSTEM

2.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to provide core cooling in the event of a loss-of-coolant accident (LOCA). The objectives of the ECCS are: to maintain the core subcritical, to remove decay heat in order to maintain core coolable geometry, limit cladding water interaction, prevent fuel melting, and remove core decay heat for an extended period of time. In PVNGS Units 1 and 3, the ECCS consists of two high-pressure safety injection (HPSI) systems, two low-pressure safety injection (LPSI) systems and four safety injection tanks (SIT). The HPSI and LPSI are arranged as two active redundant trains. Each SIT injects into a cold leg and in the case of blowdown will provide borated cooling water until the other two (active) systems are energized.

The NRC staff reviewed the licensee's submittal to verify that the ECCS system is able to adequately perform its function for the PUR. As shown in the analyses and results of Section 4.1 of this SE, "Emergency Core Cooling System Performance Analysis," the NRC staff concludes that the licensee's ECCS system is of appropriate size and capacity to protect the reactor core during a LOCA event. Therefore, the ECCS system is acceptable at the PUR condition.

2.2 Containment Heat Removal System

The licensee uses the COPATTA containment analysis code to calculate the containment response to a LOCA or main steamline break (MSLB). COPATTA assumes a spray drop size larger (and therefore more conservative) than the actual drop size produced by PVNGS Units 1 and 3 containment spray system (CSS) nozzles. The drop size is affected by the containment pressure. The PUR results in a higher containment pressure. However, the drop size assumption in COPATTA remains bounding. The spray distribution and spray flow rate assumed in the analysis are bounding and therefore are unaffected by the PUR. For these reasons the NRC staff finds the PUR's effects on the CSS to be acceptable.

3.0 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

This section of the NRC staff's review focuses on verifying the adequacy of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes. The acceptance criteria are based on continued conformance with the requirements of the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, and 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2, "Design bases for protection against natural phenomena," as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- GDC 4, “Environmental and dynamic effects design bases,” as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

- GDC 10, “Reactor design,” as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 14, “Reactor coolant pressure boundary,” as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 15, “Reactor coolant system design,” as it relates to the reactor coolant system (RCS) being designed with a sufficient margin to ensure that design conditions are not exceeded.

The specific review areas are contained in the Standard Review Plan (SRP) Section 3.9. This review also includes the plant specific provisions of NRC Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves," as related to plant specific programs for motor-operated valves, GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,” as related to the pressure locking and thermal binding for safety-related gate valves, and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," as related to the over-pressurization of isolated piping segments.

The NRC staff reviewed the PVNGS Units 1 and 3 PURLR as it relates to the structural and pressure boundary integrity of the NSSS and BOP systems. Affected components in these systems include piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), reactor vessel internals (RVIs), RSGs, control element drive mechanisms (CEDMs), reactor coolant pumps (RCPs), and pressurizer.

The NSSS design at PVNGS Units 1 and 3 was approved by the NRC staff via NUREG-0852, “Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply

Reference System CESSAR System 80." The Combustion Engineering Standard Safety Analysis Report (CESSAR) describes the design of the RCS, its components, and their supports. The CESSAR describes the methodologies used to develop limiting loads and their locations, and also contains a description and analysis of the interfaces between the CE-supplied System 80 NSSS and the rest of the plant.

In evaluating the impact of the RSGs and PUR on affected components, the licensee evaluated component stresses using the original code of record in all components except the SGs. The code of record for the original steam generators (OSGs) is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Sections II, III, V, and XI, 1971 Edition, with addenda through the Winter 1973 Addenda. The RSGs were designed and fabricated to the requirements of the 1989 Edition (no addenda) of the ASME Code, Sections II, III, V, and XI.

Load evaluation consisted of establishing revised loads for the RSGs and PUR conditions, and comparing them with the Code allowables for each load case and component. The licensee applied leak-before-break (LBB) methodology to reduce the range of dynamic effects needing Code analysis for pipe breaks, and used a computer code different from that used in the original stress analyses.

With respect to LBB, the licensee applied the methodology to justify removing from consideration the dynamic effects of breaks of RCS main coolant loop (MCL) piping. The NRC staff has previously approved the application of LBB for CESSAR NSSSs and PVNGS Units 1 and 3. The licensee justified the continued applicability of the LBB analyses to PVNGS Units 1 and 3 for the RSGs and PUR conditions. The NRC staff's evaluation of the LBB methodology is discussed below in Section 3.4.1 of this SE.

With respect to the change in computer codes used in the stress analyses, the licensee stated that the revised analyses were performed using the ANSYS computer code. The original analyses used a group of computer codes including MEC-21, STRUDL DYNAL, and CE-DAGS. MEC-21 was used for static analyses; STRUDL DYNAL was used for dynamic seismic analyses; STRUDL DYNAL and CE-DAGS (dynamic analysis of gapped structure) were used for pipe break analyses. ANSYS is a general purpose, finite-element program with structural and heat transfer capabilities. ANSYS performs all of the pertinent STRUDL DYNAL, MEC-21, and CE-DAGS functions. As described in the PURLR, the licensee performed a benchmark of the new ANSYS computer models against the original STRUDL, MEC-21, and CE-DAGS models. The results (i.e., modal frequencies, loads, and motions) of this benchmark demonstrated the equivalence of the ANSYS analyses to the original analyses. Results from the licensee's bench marking of the ANSYS code are consistent with previous NRC staff bench marking of ANSYS. The NRC staff, therefore, finds the licensee's use of ANSYS acceptable for demonstrating compliance with the ASME Code limits for stress and cumulative usage factor (CUF).

3.1 Reactor Pressure Vessel Components

The structural analysis of the RPV is addressed in Section 5.1 of the PURLR. The stresses and CUFs were determined by analyzing the RCS at the combined RSGs and PUR conditions.

The RPV was evaluated against the provisions of the ASME Code, 1971 Edition, with addenda through the Winter 1973 Addenda, which is the Code of record. The analysis considered the new dead weight and thermal loads, as well as the revised seismic and LOCA loads. The evaluation of the RCS stresses showed that the RPV analysis of record (AOR) was bounding for stresses and CUFs.

For the RPV inlet and outlet nozzles, the normal operating loads increased, the operational basis earthquake (OBE) loads decreased, and the faulted loads did not change. The licensee determined that the design and faulted condition remained bounded by the AOR. The increase in the normal loads resulted in an increase in the CUF for both the inlet and outlet nozzles, however, the CUF remains below the Code allowable.

For the closure head flange region, the evaluation showed that all loads were less than the AOR, with the exception of the faulted condition vertical load on the closure head and flange region. The analysis with the new faulted loads demonstrated that the total load on the closure head due to operating pressure, dead weight, thermal loads, and revised faulted loads, remains less than the closure stud preload; therefore, the faulted condition is not a limiting condition for the vessel closure studs. Furthermore, the maximum stresses in the head and flange region are less than the Code allowables, therefore, the head and flange region are acceptable for operation under RSGs and PUR conditions.

For the RPV inlet and outlet nozzles, the licensee determined that the PUR and RSGs condition results in an increased dead weight plus thermal load. The licensee also determined that there was a decrease in OBE loads such that the upset condition in the AOR remains bounding, however, the fatigue analyses of the nozzles were affected. The licensee revised the fatigue analyses and determined that the CUFs remain below the ASME Code limits.

The licensee also evaluated the remaining RPV components for the RSGs and PUR conditions. For the RPV nozzle supports, the licensee determined that the increases in dead weight and thermal loads were offset by decreases in OBE loads; therefore the AOR remains bounding. Reanalysis also demonstrated that the AOR remains bounding for the CEDM nozzles and incore instrumentation (ICI) nozzles. For the RPV support columns, the licensee determined that the maximum stresses remain below the Code allowable limits.

The licensee has provided the stresses and CUFs at the limiting locations for the above components. The NRC staff verified that the stresses and CUFs are below the Code allowables.

For the reasons set forth above, the NRC staff concurs with the licensee’s analyses and conclusion that structural loads on the RPV and its support components are within operating limits, as defined in 10 CFR 50.55a, and 10 CFR Part 50 Appendix A, for operation under RSGs and PUR conditions.

3.1.1 Reactor Vessel Materials Surveillance Program

The Reactor Vessel (RV) material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring

the structural integrity of the ferritic components of the RV. 10 CFR Part 50, Appendix H, provides the staff's requirements for the design and implementation of the RV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed Power Uprate (PUR) on the licensee's RV surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14, which requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the RV beltline region; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of NRC Review Standard No. RS-001, Revision 0, *Review Standard for Extended Power Uprates* (December 2003).

Regarding the Palo Verde Nuclear Generating Station (PVNGS) Units 1 and 3 RV surveillance program and capsule withdrawal schedule, the licensee concluded,

> The schedule was established based on the original calculation of fluence that was shown to bound conditions for PUR (3990 MWt). The analysis of record (AOR) was performed for a power level of 4200 MWt. The proposed PUR is to 3990 MWt. The surveillance capsule withdrawal schedule was established based on the original calculation of fluence based on 4200 MWt that was shown to bound conditions for PUR (3990 MWt). The detailed surveillance schedule is discussed in UFSAR Section 5.3.1-6.6 and Table 5.3-19. Therefore, the existing surveillance capsule withdrawal schedule remains applicable under conditions for PUR.

The AOR was based on the initial out-in fuel loading while the actual refueling loadings were in-out to lower vessel irradiation and to support increased fuel cycle lengths. The methodology in WCAP-15589 adheres to the guidance in Regulatory Guide (RG) 1.190 regarding approximations, cross sections and source evaluations, therefore, it is acceptable. The NRC staff concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the reactor vessel material surveillance program.

3.1.2 Pressure-Temperature Limits and Upper Shelf Energy

10 CFR Part 50, Appendix G, provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) materials in the RCPB, including requirements on the upper shelf energy (USE) values used for assessing the remaining safety margins of the RV materials against ductile tearing and requirements for calculating pressure-temperature (P-T) limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The staff's review of the USE assessments

covered the impact of the PUR on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period for PVNGS units 1 and 3. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of the EFPY specified for the proposed PUR, considering neutron embrittlement effects and using linear elastic fracture mechanics.

The NRC's acceptance criteria for P-T limits and USE are based on (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of NRC Review Standard RS-001, Revision 0.

Regarding the topic of the RV P-T limits,

> The licensee concluded that there are no changes to the RT_{NDT} values that were used to establish the Appendix G normal operating limits. The AOR was performed for a power level of 4200 MWt. The proposed PUR is to 3990 MWt. The limits represent conditions for PUR (3990 MWt) given that the projected fluence at the end of license, 3.29×10^{19} n/cm^2 (calculated based 4200 MWt), E>1 MeV, is bounded by the AOR such that the predicted vessel material properties used to establish the heat-up and cool-down limits are unchanged.

The NRC staff has evaluated the information provided by the licensee's supplement letter dated June 3, 2005, as well as information contained in the staff's Reactor Vessel Integrity Database. The current P-T limits are based on AOR fluence and is bounded, therefore, the staff concludes that there will be no impact on the P-T limit curves.

Regarding the topic of the RV USE, the licensee concluded,

> The beltline materials were determined using the PUR fluence to have USE greater than 50 ft-lb. through the end of license, as required by 10 CFR 50, Appendix G. For Palo Verde units 1 and 3, the lowest USE value at the end of current license was determined to be 65.2 ft-lbs and 70.70 ft-lbs respectively. These USE values are based on AOR, 3.29×10^{19} n/cm^2(E>1 MeV) and is conservative.

The NRC staff has evaluated the information provided by the licensee's supplement letter dated June 3, 2005, as well as information contained in the staff's Reactor Vessel Integrity Database. Based on the revised PUR fluence, the staff independently confirmed that the PVNGS Units 1 and 3 RV materials would continue to meeting the USE criteria requirements of 10 CFR 50, Appendix G.

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed PUR on the P-T limits for the plant and USE values for the RV beltline materials. The staff concludes that the licensee has adequately addressed changes in neutron fluence and their impacts on the

P-T limits for the plant and USE values for the PVNGS Units 1 and 3 RVs. The NRC staff concludes that the licensee has demonstrated the validity of the current P-T limits for operation under the proposed PUR conditions. The staff also concludes that the Palo Verde RV beltline materials will continue to have acceptable USE, as mandated by 10 CFR Part 50, Appendix G, through the expiration of the current operation license for the facility. Based on this assessment, the NRC staff concludes that the Palo Verde Units 1 and 3 will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the P-T limits and USE.

3.1.3 Pressurized Thermal Shock

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to assure that adequate fracture toughness is provided for supporting reactor operation. The staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61. The NRC staff's review covered the PTS methodology and the calculations for the reference temperature, RT_{PTS}, at the expiration of the license, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS is based on (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (3) 10 CFR 50.61, which sets fracture toughness criteria for protection against PTS events. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of NRC Review Standard RS-001, Revision 0.

Regarding the topics of PTS analyses for the PVNGS Units 1 and 3 RV, the licensee provided RT_{PTS} values for the beltline materials of the PVNGS Units 1 and 3 vessels in its supplement letter dated June 3, 2005:

> The pressurized thermal shock calculations were performed for the PVNGS Units 1 and 3 beltline materials using the 10 CFR 50.61. Based on this evaluation, the pressurized thermal shock values were determined to remain below the NRC screening criteria through the end of license for the PVNGS Units 1 and 3 using the projected fluence and thus meet the requirements of 10 CFR 50.61.

The NRC staff has evaluated the information provided by the licensee as well as information contained in the staff's Reactor Vessel Integrity Database. The licensee's PTS evaluation is based on the AOR values, 3.29×10^{19} n/cm^2. This fluence value is very conservative when compared with the PUR fluence values. The staff independently confirmed that the PVNGS Units 1 and 3 RV materials would continue to meeting the PTS screening criteria requirements of 10 CFR 50.61.

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed PUR on the PTS for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on PTS. The NRC staff further concludes that the licensee

has demonstrated that the plant will continue to meet the requirements of GDC-14, GDC-31, and 10 CFR 50.61 following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to PTS.

3.1.4 Reactor Internal and Core Support Materials

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2 and other review criteria and guidance are provided in Matrix 1 of NRC Review Standard RS-001, Revision 0. Matrix 1 of NRC Review Standard RS-001, Revision 0, provides references to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Revision 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals* (March 2001) and BAW-2248A, *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals* (March 2000).

The RV internals of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking ! induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC)
- loss of fracture toughness properties ! induced by irradiation exposure for all stainless steel grades, or the synergistic effects of irradiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;
- stress relaxation in bolted, fastened, keyed or pinned RV internal components ! induced by irradiation exposure and/or exposure to elevated temperatures
- void swelling (induced by irradiation exposure).

Table Matrix-1 of NRC Review Standard RS-001, Revision 0, provides the staff's basis for evaluating the potential for extended power uprates to induce these aging effects. In Table Matrix-1, the staff states that, in addition to the SRP, guidance on the neutron irradiation-related threshold levels inducing IASCC in RV internal components are given in WCAP-14577, Revision 1-A. WCAP-14577, Revision 1-A, establishes, a threshold of $1 \text{ X } 10^{21}$ n/cm^2 (E $\$$ 0.1 MeV) for the initiation of IASCC, loss of fracture toughness, and/or void swelling in PWR RV internal components made from stainless steel (including cast austenitic stainless steels) or Alloy 600/82/182 materials.

In RAI # 3, the staff informed PVNGS that, consistent with Table Matrix-1 of NRC Review Standard RS-001, Revision 0, either an inspection plan would need to be established to manage the age related degradation in the PVNGS Units 1 and 3 RV internals, or that a commitment would be needed indicating that PVNGS would participate in the industry's initiatives on age-related degradation of PWR RV internal components. In its June 3, 2005, letter response to RAI #3, APS confirmed that they are currently an active participant in the

Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) research initiatives on aging related degradation of reactor vessel internals components. APS committed to:

a) Continue its active participation in the MRP initiative to determine appropriate reactor vessel internals degradation management programs,
b) Evaluate the recommendations resulting from this initiative and implement a reactor vessel internals degradation management program applicable to Palo Verde 1 and 3,
c) Incorporate the resulting reactor vessel internals inspections into the Palo Verde 1 and 3 augmented inspection program as appropriate and provide the internals inspection plan to the NRC staff for review and approval within 24 months of the EPRI MRP final recommendations or within five years from the date of issuance of the uprated license, whichever comes first.

The licensee's commitments to participate in the industry's research program of degradation of PWR RV internal components and to develop and submit for staff approval an inspection program for the RV internals that is based on the recommendations of the industry initiatives are consistent with Table Matrix-1 of NRC Review Standard RS-001, Revision 0, and are therefore acceptable. Based on this assessment, the staff concludes that PVNGS has established an acceptable course of action for managing age-related degradation in the PVNGS Units 1 and 3 RV internals under the PUR conditions for the units.

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed PUR on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to reactor internal and core support materials.

3.1.5 Conclusion

The staff has reviewed PVNGS proposed license amendment to increase the rated core thermal power for PVNGS Units 1 and 3 by 2.94-percent and has evaluated the impact that the PUR conditions will have on the structural integrity assessments for the RV and RV internals. The staff has determined that the proposed license amendment will not significantly impact the remaining safety margins required for following RCS-related structural integrity assessments: (1) RV Surveillance Program for the PVNGS Units 1 and 3, (2) USE assessment for the PVNGS RVs, (3) P-T limits for the PVNGS RVs, (4) PTS assessment for the PVNGS RV beltline materials, and (5) structural integrity assessment of the PVNGS RVs internal components, in that the licensee has committed to the establishment of a plant-specific inspection program for the RV internals.

Therefore, the NRC staff determined that the proposed power uprate will not significantly impact the operation of the RVs or the RV internals, and therefore, the staff finds the requested power uprate acceptable.

3.2 Reactor Pressure Vessel Internals

The RVIs are addressed in Section 5.2 of the PURLR. The RVIs were evaluated for a range of loadings, including normal operating pressure and temperature differences, flow loads, vibration loads, shock loads [including OBE and safe shutdown earthquake (SSE)], loads from anticipated transients, and LOCA loads. The evaluations were performed in accordance with ASME Code, Section III, 1974 Edition with no addenda, which is the code of record for the RVI.

The PUR conditions with respect to temperatures, pressures, and flows, are bounded by the original design conditions, as shown in Table 2.1-1 of the PURLR in the licensee's December 21, 2001, PVNGS Unit 2 PUR submittal. The licensee has stated that these conditions are bounding for Units 1 and 3. Since the RCS flow remains within the original design bases, the previous analysis for flow-induced vibration (FIV) remains valid. Also, the hydraulic lift forces are such that the RVIs will remain seated and stable.

The licensee's evaluation of the RVIs demonstrated that the Service Level A, B, and D stresses meet the criteria in Section III, Division 1, of the ASME Code. The licensee's evaluation also shows the deflections are such that the control element assemblies maintain their ability to scram for the RSGs and PUR conditions in accordance with the PVNGS Units 1 & 3 design bases.

The licensee provided a summary of the calculated stresses and CUFs for the RVIs. The NRC staff verified that the stresses and CUFs are within the Code allowables. Based on this review, the NRC staff concurs with the licensee's conclusion that the RVIs are acceptable for operation under RSGs and PUR conditions.

3.3 Other Equipment on Reactor Pressure Vessel

The heated junction thermocouples (HJTCs) are described in Section 5.3 of the PURLR. The HJTCs enter the reactor head through existing CEDM nozzles. The licensee evaluated the excitation of HJTC cables and flanges for seismic and LOCA events at the RSGs and PUR conditions. For the cables, the licensee determined that the accelerations used in the qualification tests bound the RSGs and PUR conditions. For the HJTC instrument flange assemblies, the licensee determined that the original design analyses envelop the results for operation under RSGs and PUR conditions.

The incore instrumentation (ICI) tubes are described in Section 5.3 of the PURLR. The ICI tubes are attached to the lower head of the RPV and terminate at the ICI seal table. The AOR analyzed dead weight, thermal expansion, pressure, seismic, mechanical, and LOCA loads to verify the structural integrity of the tubes. The analysis used a set of configuration spectra that were intended to bound the RPV spectra and the containment basemat spectra. The licensee identified that, for the vertical direction, the original configuration spectra did not bound the basemat spectra or the RPV spectra for the RSGs and PUR conditions. Also, the licensee clarified that the basemat spectra had not changed for RSGs and PUR conditions. The licensee reanalyzed the tubes for the new spectra and determined that the AOR remained bounding for the excitation of the ICI tubes.

As described in Section 5.3.4 of the PURLR, the licensee evaluated the permanent head lift rig (HLR) structure for seismic and LOCA loads for RSGs and PUR conditions. The licensee determined that the HLR stresses were within the allowable limits for all service level conditions.

Based on our review, for the reasons set forth above, the NRC staff concurs with the licensee's conclusion that these components remain acceptable for operation under RSGs and PUR conditions.

3.3.1 Control Element Drive Mechanisms

An assessment of the CEDMs is provided in Section 5.3 of the PURLR. The CEDMs were evaluated for normal conditions, upset conditions, and faulted conditions in accordance with the ASME Code, 1974 Edition with addenda through the Winter 1975 Addenda, which is the code of record.

The licensee changed the methodology by using a three-dimensional ANSYS model of the CEDMs, rather than the SAPIV code model (for response spectrum analyses) used for the AOR, and by considering LBB. The licensee benchmarked the ANSYS model by comparing the results to test data. The model was used for dead weight, seismic, and LOCA analyses. The calculated CUFs were bounded by the AOR. The stresses increased for some items, but remain less than the Code allowables.

The licensee also assessed the absolute deflections of the CEDMs. This evaluation demonstrated that (1) the control rods would trip as designed, (2) there was no impact between adjacent CEDMs, and (3) the reed switch position indicators remained qualified (e.g., deflections were less than those used in the equipment qualification tests) for the RSGs and PUR condition.

Based on the above reasoning, the NRC staff concurs with the licensee's conclusion that the CEDMs are acceptable for operation under RSGs and PUR conditions.

3.4 Reactor Coolant System Components

The RCS piping and supports are addressed in Section 5.4 of the PURLR. The licensee assessed the RCS piping and supports in accordance with the ASME Code, 1974 Edition, with addenda through the Summer 1974 Addenda, which is the code of record. The licensee assessed the RCS tributary piping and pressurizer surge line in accordance with ASME Section III, 1974 Edition, with addenda through the Winter 1975 Addenda (Summer 1979 Addenda for Subsections NB-3650 through 3680), which is the code of record for the tributary piping.

Installation of the RSGs, which are larger than the OSGs, will necessitate modification of some RCS cold leg piping. In addition, the RSGs and PUR conditions resulted in a new set of loads (i.e., pressure, thermal expansion, deadweight, seismic, and LOCA) for the RCS piping. The licensee determined that thermal design transients are bounded by the transients specified in the AOR. The seismic and LOCA loads were calculated using the ANSYS model described above. The licensee provided the calculated stresses and CUFs at the limiting locations. The stresses and CUFs are below the allowable limits, and therefore, RCS components were

acceptable under RSGs and PUR conditions.

3.4.1 Leak Before Break

During the review of CESSAR, Combustion Engineering (CE) submitted its basis for applying LBB to its System 80 NSSSs. By letter dated June 14, 1983, CE submitted a report entitled "Basis for Design of Plant Without Pipe Whip Restraints for RCS Main Loop Piping." In response to NRC staff concerns, by letter dated December 23, 1983, CE submitted a revision to the report. On the basis of deterministic fracture mechanics analyses, CE contended that postulated double-ended guillotine breaks of the NSSS MCL piping will not occur in CESSAR facilities and, therefore, did not need to be considered as a design basis for installing protective devices such as pipe whip restraints and jet impingement shields. No other changes in the design analyses were addressed. For example, no changes were proposed to the definition of a LOCA or its relationship to the regulations addressing the design requirements for the emergency core cooling systems (10 CFR 50.46), containment (GDC 16 and 50), other engineered safety features, and the conditions for environmental qualification of equipment (10 CFR 50.49).

By letter dated July 16, 1985, the licensee requested a partial exemption to GDC 4 to permit the design of PVNGS Unit 3 without pipe whip restraints or missile shields. The technical justification for the exemption request was provided by CE letters dated June 14, 1983, and December 23, 1983, which were submitted for CESSAR. By letter dated November 3, 1984, the licensee had clarified that the technical information in the CE letters was based on PVNGS Unit 2, which is the prototype for CESSAR (System 80) NSSSs. By letter dated November 11, 1984, the NRC staff accepted the application of LBB for removing pipe whip restraints and missile shields in CESSAR NSSSs. By letter dated November 29, 1985, the NRC staff granted the partial exemption to permit this application of LBB at PVNGS Unit 3. Since PVNGS Unit 1 had received its operating license, the same partial exemption was granted for Unit 1 in a separate letter, dated November 22, 1985.

In evaluating the acceptability of LBB for PVNGS Units 1 and 3, the NRC staff evaluated the MCL piping for the following: the location of maximum stresses in the piping, associated with the combined loads for normal operation and SSE; potential cracking mechanisms; size of throughwall cracks that would leak a detectable amount under normal loads and pressure; stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load; margin based on crack size; and the fracture toughness properties of piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in Enclosure 1 to GL 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with SSE. These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe-ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes systems operation procedures; system or component modification; water chemistry parameters, limits, and controls; resistence of material to various forms of stress corrosion, and performance under cyclic loadings.

(3) A throughwall crack should be postulated at the highest stressed locations determined from (1) above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(4) It should be demonstrated that the postulated leakage-size crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be determined by a crack stability analysis; i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and that the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size margin should be determined by comparing the leakage-size crack to critical-size crack. Under normal plus SSE loads, it should be demonstrated that there is an adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses and in the leakage detection capability. A limited-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

(6) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe-ends, the materials properties including the J-R curve used in the analyses, and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, maximum crack growth).

The NRC staff's November 11, 1984, letter contains the NRC staff's evaluation of LBB for CESSAR facilities. The evaluation is also reflected in Supplement 3 to NUREG-0852, dated December 1987. The NRC staff's acceptance was based on the following:

(1) The loads associated with the most highly stressed locations in the main loop primary system were provided and are within Code allowables.

(2) For CE plants, there is no history of cracking failure in reactor primary coolant system piping. CE reactor coolant system primary loops have an operating history which demonstrates their inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history includes several plants with many years of operation.

(3) The results of the leak rate calculations performed for CESSAR used initial postulated throughwall flaws that are equivalent in size to that in Enclosure 1 to GL 84-04.

CESSAR facilities are expected to have an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," so that they can detect leakage of one gpm in one hour. The calculated leak rate through the postulated flaw is large relative to the NRC staff's recommended sensitivity of plant leak detection systems. The margin is at least a factor of 10 on leakage.

(4) The expected margin in terms of load for the leakage-size crack under normal plus SSE loads is greater than a factor of three when compared to the limit load. In addition, the NRC staff found a significant margin in terms of loads larger than normal plus SSE loads.

(5) The margin between the leakage-size crack and the critical-size crack was calculated. Again, the results demonstrated that a crack size margin of at least a factor of three exists.

The NRC staff's November 11, 1984, letter also states that, in order for licensees with CESSAR facilities to use the above, they should confirm that their as-built facility design substantially agrees with the design described in CE's letters dated June 14 and December 23, 1983, and that the piping loads should be no greater than those cited in those documents. The licensees should also verify that the leakage detection system meets RG 1.45.

The licensee submitted the value-impact analysis in letter dated October 3, 1984. It states that the LBB analysis was performed on the PVNGS design (as the prototypical CESSAR plant) using pertinent PVNGS parameters. Therefore, the CE analysis envelopes the PVNGS design with respect to such parameters as loads, material properties, postulated crack leakage and size, seismicity, and leak detection system capabilities. In addition, the leak detection system for PVNGS Units 1 and 3 is consistent with RG 1.45 so that it can detect leakage of one gpm in one hour. Based on the above, the NRC staff found that LBB could be applied to PVNGS Unit 1 and Unit 3 and granted a partial exemption by letters dated November 22, 1985 and November 29, 1985.

For the PUR and RSGs conditions, the licensee proposes to use LBB to determine the faulted condition loads in the RCS. In accordance with GDC 4 that was in effect during the original plant design, which did not account for LBB methodologies, the mechanical design of the PVNGS Units 1 and 3 RCS included postulated breaks in all high energy piping greater than one inch in diameter. With the proposed application of LBB, the dynamic effects of MCL pipe breaks were excluded from the design basis for the RCS piping and components. Following the application of LBB at PVNGS Unit 1 and 3, the limiting pipe breaks considered in the RCS structural integrity analyses are branch line pipe breaks in the following largest tributary pipes: pressurizer surge line, safety injection lines, shutdown cooling lines, charging line, and letdown line. The licensee also analyzed the effects of the BOP pipe breaks, such as feedwater (FW) line breaks and MSLBs. No other changes in the design analyses were requested. For example, no changes were proposed to the definition of a LOCA or its relationship to the regulations addressing the design requirements for the emergency core cooling systems (10 CFR 50.46), containment (GDC 16 and 50), other engineered safety features, and the conditions for environmental qualification of electric equipment important to safety (10 CFR 50.49). The application of LBB methodology to subcompartment loads and compliance with the

appropriate GDC are discussed in Section 4.2.5 of this SE. Using the methodology described above, the licensee evaluated the maximum stresses and CUFs for the RSGs and PUR conditions, as well as the material properties of the RSGs and the new cold leg elbows and field welds.

The licensee stated that, for the crack that will leak at the rate of 10 gpm at normal operating conditions (leakage crack) and crack stability, the areas of concern are the RPV inlet and outlet nozzles, since these are regions of high stress. For the RSGs and PUR conditions, the bending moments for normal operating loads at the nozzles are smaller than the original bending moments used to determine the leakage crack length (defined as the length of crack that will leak 10 gpm at normal operating conditions). Consequently, the leakage crack length for the PUR condition is longer than the leakage crack for the original condition. The leakage crack for the RPV inlet and outlet nozzles is about 11 percent and 8.5 percent of the pipe circumference, respectively. The licensee stated that a critical crack of 50 percent of the pipe circumference remains stable when subjected to both normal operating and SSE loads. Thus, the leakage crack length for the RSGs and PUR configuration is below the stability criterion. Further, the combined normal operating and SSE loads for the RSGs and PUR configuration are less than those used for the original configuration, so a leaking crack in the RPV inlet or outlet nozzle will be detectable well before the crack can grow to an unstable length.

For the material property considerations, the licensee's states the following:

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

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

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

The licensee also states that the replacement RCS piping is at the RSGs' outlet nozzles, which are not at the critical stress locations used in the LBB analysis.

The analysis generally assumed a reduction in metal toughness by a factor of four, but did not apply this reduction to the weld metal. The licensee provided the following response:

> At the time the Palo Verde LBB analysis was performed, there was very limited J-R Curve fracture toughness data available for the piping and weld materials. A representative J-R Curve for the SA-516 Grade 70 piping base material was selected for use. An arbitrary factor of four was applied to this base metal curve to provide margin

for the analysis, including uncertainty in material properties. The reduction was not due to any real or postulated degradation in toughness properties for the piping materials.

In the more recent evaluations performed, some weld metal J-R Curves were shown to be lower than the SA-516 Grade 70 piping base material used in the original LBB analysis. However, the RSGs and PUR LBB assessment demonstrated that the lower bound of the weld metal data was still at least a factor of two higher than the reduced J-R Curve that is the basis for the original analysis. Effectively, the reduced J-R Curve used for the original LBB analysis and the RSGs and PUR LBB assessment is a factor of 4 less than the lower bound of the base metal test data and a factor of two less than the lower bound of the weld metal test data. Therefore, the RSGs and PUR LBB assessment demonstrated that the material property curve used in the original LBB analysis is still conservative for both the base metal and weld metal piping materials and, hence the original LBB analysis remains valid.

For the reasons set forth above, the NRC staff concurs with the licensee's conclusion that the LBB evaluation remains valid for the PVNGS Unit 1 and 3 MCL piping. GDC 4 states that the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The above evaluation demonstrates that the probability of a MCL piping rupture at PVNGS Unit 1 and 3 is extremely low for the RSGs and PUR design basis conditions; therefore, continued application of LBB to the PVNGS Unit 1 and 3 MCL piping is acceptable.

3.4.2 Reactor Coolant Pumps

The RCPs and motors are addressed in Section 5.4.5 of the PURLR. The RCPs were designed and analyzed to meet the pump design specifications and ASME Code criteria. The RCPs are designed to the 1974 Edition of the ASME Code, with no addenda.

In its assessment, the licensee determined that design transients already considered in the AOR would bound any transients applicable to the PUR. For the structural assessment, the licensee evaluated those portions of the RCPs that have relatively low stress margins (less than 10 percent), and compared the dead weight and thermal, seismic, and faulted loads for the RSGs and PUR condition to the loads in the AOR. The only location that needed reassessment of the stresses was the lower section of the motor stand shell/lower window. For this location, the stress was shown to be below the allowable limit. The new loads for the RSGs and PUR conditions were determined to be acceptable for the RCPs, and the pump pressure boundary components were demonstrated to remain within the RCP design specification and the ASME Code. Peak accelerations of RCP motors were evaluated for the RSGs and PUR conditions; calculated values were less than the design limits by significant margins for all cases.

For the reasons set forth above, the NRC staff concurs with the licensee's conclusion that the RCPs remain acceptable for operation at the RSGs and PUR conditions.

3.5 Steam Generators

The RSGs are designed and fabricated to operate at PUR conditions. Generally, the RSGs differ from the OSGs as follows: the number of tubes is increased by 10 percent and tube material is changed from Inconel 600 to Inconel 690; the RSGs are larger, such that primary and secondary water volumes are increased, dry weight is increased, main steam and FW nozzle elevations change, and snubber lugs now project from the shell cone; new recirculation and upper blowdown nozzles are added.

The RSGs were designed and fabricated to the requirements of the ASME Code 1989 Edition (no addenda) for structural properties, thermal-hydraulic characteristics, U-bend fatigue, tube degradation, tube plugging, and repair requirements. The licensee provided the stresses and CUFs for the RSGs, and the stresses and CUFs are below the Code allowables.

With respect to the potential for FIV of the SG tubes, the licensee states that the potential for FIV is minimized due to the design of the RSGs. FIV analyses were performed on selected tubes based on tube span parameters, such as frequency and mode shape of vibration, and fluid flow parameters, including flow velocity and fluid density. Selected tubes were modeled with the ANSYS code to determine natural frequencies and mode shapes. The FIV analysis considers fluid elastic instability and random turbulent excitation mechanisms. The evaluation results showed a maximum stability ratio of 0.38, which is less than the design goal of 0.75, in the fluid exit bend region of the tube bundle. The licensee also stated that the turbulent displacements are within the limit.

The licensee reanalyzed the loads of the sliding base assembly (baseplate and other subcomponents) using the original design methodology. The licensee accounted for the increase in dead weight of the SG, as well as the redesign of the SG skirt. The licensee stated that stresses in the vertical support structures (pads, sliding base, bolting, key ways, etc.) remain within Code limits. For the SG upper supports (i.e., snubber arrangements, upper Z keys, and supporting structures), the loads increased on some components; however, the loads remain less than the design loads.

For the reasons set forth above, the NRC staff concurs with the licensee’s conclusion that the RSGs are acceptable for operation at PUR conditions.

3.5.1 Steam Generator Materials

Steam generator tubes form a part of the reactor coolant pressure boundary, to which GDC 14, "Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 is applicable. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. As set forth in 10 CFR 50.55a, SG tubes are required to meet various specifications of the ASME Code.

The RSGs for PVNGS Units 1 and 3 were manufactured by Ansaldo-Camozzi Energy Special Components of Italy and the tubes were fabricated by Sandvik of Sweden. The design was performed by ABB-Combustion Engineering (which is owned by Westinghouse) based on the Combustion Engineering System 80+ design. The RSGs were designed and analyzed for the PUR conditions in accordance with the requirements of the ASME Code Section III, 1989 edition, no addenda.

The RSGs include several improvements to mitigate potential stress corrosion cracking. For example, the tubing is made of Alloy 690 thermally treated material, which has been shown to have higher resistance to corrosion cracking than the Alloy 600 material used in the original SG tubing. The replacement SG uses eggcrate configuration design to support the vertical runs of the tubes. The eggcrate design will minimize accumulation of contaminants at the tube to tube support intersections and thus minimize corrosion at those locations. Diagonal strips provide out-of-plane support to 90 degree bends and vertical grids support the horizontal run of the tubes in the upper bend region. The tube supports are made from Type 409 stainless steel which was selected for its resistance to erosion and corrosion. To mitigate potential degradation in the tubesheet, the entire length of the tube inside the tubesheet is hydraulically expanded to minimize residual stress and crevices at the top of the tubesheet.

With respect to the structural integrity of the replacement SG tubes, the licensee will assess the integrity of the tubing using the plant's TS requirements for inspection. The evaluation, testing, and analytical processes for condition monitoring and operational assessment will be performed as specified per plant procedures. The licensee stated that it has adopted the techniques and guidance specified in Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines, and various Electric Power Research Institute (EPRI) guidelines, including SG integrity assessment guidelines, in situ pressure test guidelines, and flaw assessment handbook.

Industry operational experience is that thermally treated tubing does not undergo noticeable degradation in service. This experience, therefore, demonstrates the corrosion performance of such tubing. The licensee stated that no new forms of degradation with respect to morphology or physical characteristics are anticipated. Should tube degradation occur, it will be identified, monitored, and assessed via a TS inspection program. The program gives the frequency and sampling requirements for eddy current inspections of SG tubing. The licensee stated that the SG inspection program satisfies the requirements of ASME Section XI, 1992 edition (including Code Cases –356, N401-1 and N402-1) and the guidance of NRC RG 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." The NRC staff concludes that the structural integrity of the replacement SG tubes will not be negatively impacted under the PUR conditions based on the licensee’s adherence to the plant's TSs.

With regard to the leakage integrity in the RSGs, the licensee stated that it uses a conservative primary-to-secondary leakage monitoring criterion that exceeds the TS requirements and EPRI guidance. The licensee uses a conservative primary-to-secondary leakage monitoring program in its plant procedures in which plant shutdown is initiated if the leakage exceeds 50 gallons per day. The leakage limit in the plant's TSs is 150 gallons per day. In addition, the leakage assessment for accident conditions does not change as a result of the RSGs or PUR. The changes resulting from PUR with respect to leakage integrity are bounded by the design basis. The NRC staff concludes that the leakage integrity of the RSGs will not be negatively impacted under PUR conditions because the licensee’s assessment showed that the leakage integrity of the RSGs is within the design basis.

With regard to the tube plugging limit, the licensee stated that the design basis for the 40 percent tube wall thickness is calculated from the margins defined in RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and the requirements in ASME Code,

Section III, NB-3324.1. The licensee calculated a tube structural limit of 0.0121 inch, which is 32 percent of the nominal tube wall thickness of 0.042 inch. The tube nominal diameter is 0.75 inch. The structural limit also satisfies the allowable radius-to-thickness ratio in ASME Section XI IWB-3521.1. The degradation limit based on the structural consideration would be 68 percent of the tube wall thickness. Considering the growth rate of potential degradation and the inspection uncertainty per RG 1.121, the NRC staff concludes that the tube plugging limit of 40 percent in the plant's TSs is conservative and acceptable.

With regard to the corrosion resistance of Alloy 690 thermally treated tubing under the PUR conditions, the licensee stated that there is a significant database of industry literature with regard to its corrosion resistance performance. With the exception of caustic environments containing lead (unanticipated in a SG environment), Alloy 690 thermally treated material has been shown to have superior corrosion resistance over mill-annealed Alloy 600 material. Alloy 690 material has shown improvements in corrosion resistance in both primary and secondary side environments in laboratory tests. The licensee stated that for the environments bounding the PUR conditions, improvement factors of 2 to 10 times have been verified in the laboratory. Alloy 690 has been proven in nuclear plants to have resistance to primary water stress corrosion cracking because thousands of Alloy 690 tube plugs have been installed with up to 10 years of operating experience without cracking. In addition, there has not been any stress corrosion cracking identified with Alloy 690 thermally treated tubing in the last 11 years of operation. Accordingly, the NRC staff concludes that the use of Alloy 690 thermally treated material is acceptable under the PUR conditions on the basis of laboratory tests and operating experience.

With regard to the U-bend fatigue concern, the licensee stated that the RSGs contain design and materials that preclude U-bend fatigue problems identified in NRC Bulletin 88-02, “Rapidly Propagating Fatigue Cracks in Steam Generator Tubes.” Bulletin 88-02 described a tube rupture event at North Anna Unit 1 on July 15, 1987. The cause of the tube rupture at North Anna Unit 1 was high cycle fatigue in the affected tubes. The licensee stated that the tube support material for the PVNGS Units 1 and 3 RSGs is stainless steel SA 240 Type 409. This material is not susceptible to the denting conditions experienced by the carbon steel support plates in the North Anna SGs. As discussed above, the upper tube supports in the RSGs include a combination of eggcrates, vertical strap supports, and batwing supports. The upper tube supports are designed to prevent out-of-plane deflection that would cause fatigue. The licensee stated that the upper bundle support design has about 30 years of operating experience with no evidence of fatigue concerns. As reported in Bulletin 88-02, FIV has a significant effect on tube response in cases where the fluid elastic stability ratio equals or exceeds 1.0. For the RSGs, a maximum stability ratio of 0.7 was imposed. Since the tubes in the RSGs were analyzed for various fluid velocities to ascertain that the stability ratio of 0.7 is achieved, the NRC staff agrees with the licensee’s reasoning and conclusion that the U-bend fatigue identified in Bulletin 88-02 is not applicable to the PVNGS Units 1 and 3 RSGs.

The RSGs have improved tube support design to address the tube wear that occurred in the OSGs. The improvements include: (1) The horizontal grids (eggcrates) provide support to the vertical runs of the tubes, (2) The vertical grids provide vertical and horizontal support to the horizontal run of tubes in the upper bend region, (3) The diagonal strips provide out-of-plane support to 90 degree tube bends, and (4) The vertical grids are welded to the diagonal supports to provide tube stability. These improvements will reduce the stresses in the tubes, and consequently, will reduce the potential for tube wear in the RSGs. On the basis of this

improved design, the NRC staff concludes that tube wear will not be negatively impacted by the PUR conditions.

On the basis of the information the licensee provided, as set forth above, the NRC staff concludes that the proposed PUR is acceptable with respect to RSG tubes because the structural and leakage integrity of the RSG tubes under the PUR conditions satisfy GDC 14 and the ASME Code.

3.6 Pressurizer

The pressurizer is addressed in Section 5.6 of the PURLR. The pressurizer was constructed to the ASME Code, 1971 Edition, with addenda through the Winter 1973 Addenda. The licensee determined that the current design analyses for the pressurizer are bounding for the RSGs and PUR conditions; therefore, the AOR remains bounding for the pressurizer. The NRC staff notes that the proposed RSGs and PUR conditions do not result in a change to the pressurizer operation pressure or temperature, and the temperatures of the RCS hot and cold legs remain within the original design limits. Accordingly, the NRC staff concurs with the licensee's conclusion that the pressurizer remains acceptable for operation at RSGs and PUR conditions.

4.0 **NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSIS**

4.1 Emergency Core Cooling System Performance Analysis

The NRC staff reviewed the ECCS performance to confirm that the system design provides an acceptable margin of safety from conditions which would lead to fuel damage during normal reactor operation including anticipated operational occurrences, and has been accomplished using acceptable analytical methods. Acceptance criteria are based on the provisions of 10 CFR 50.46.

The licensee stated the LOCA analyses results for PVNGS Unit 1 and 3 operating at 4070 MWt (3990 MWt plus 2 percent measurement uncertainty) in its July 9, 2004, PUR submittal. The NRC staff reviewed the analyses to assure that PVNGS Unit 1 and 3, operating at the uprated power of 3990 MWt, would satisfy the ECCS criteria of 10 CFR 50.46(b).

The licensee performed these and previous LOCA analyses using W/CE-approved LB and SBLOCA methodologies described in topical reports CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985, and CENPD-137, Supplement 1-P-A, "Calculative Methods for the CE Small Break LOCA Evaluation Model," January 1977 (S1M), respectively.

The licensee provided results of LBLOCA and SBLOCA analyses, repeating the previous results (above), and results using CENPD-132, Supplement 4-P-A, "Calculative Methods for the C-E Nuclear Power Large Break LOCA Evaluation Model," March 2001, and CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998 (S2M), respectively.

The licensee states that "APS and Westinghouse Electric Company, LLC, have ongoing processes that assure that LOCA analysis input values for peak cladding temperature sensitive parameters bound the as-operated plant values for those parameters" to show that it would

properly model the PVNGS Units 1 and 3 plants in using the above models, such that the reported results could specifically represent the PVNGS Units 1 and 3 ECCS performance within the applicability range of the model.

The NRC SEs of both CENPD-404-P-A and WCAP-12610-P-A discuss LOCA analyses for mixed cores with Zircaloy-clad fuel and Vantage+ (ZIRLO-clad) fuel. These SEs both agree with the statement that: “Because of the close similarity between Vantage+ and Vantage-5 (Zircaloy-clad) fuel assemblies, a mixed core penalty need not be applied to any mixed core combination of Vantage-5 (Zircaloy-clad) and Vantage+ fuel assemblies, if both have the same design features” (WCAP-12610 SE, Appendices F and G). The NRC and industry have understood that the most significant of these “design features” include geometry (e.g., differences in spacer or mixing grids) and surface roughness. These conclusions also apply to the methodologies described in CENPD-132, Supplement 3-P-A, and CENPD-137, Supplement 1-P-A. Therefore, the licensee may use any or all of the above LOCA methodologies to perform LOCA analyses for mixed cores with Zircaloy-clad fuel and ZIRLO-clad fuel.

The licensee provided results for PVNGS Units 1 and 3 LBLOCA analyses Westinghouse performed on its behalf at the PUR using both the CENPD-132, Supplement 3-P-A LBLOCA methodology and the CENPD-132, Supplement 4-P-A LBLOCA methodology. Westinghouse explicitly analyzed ZIRLO-clad fuel only with the CENPD-132, Supplement 4-P-A LBLOCA methodology.

The calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation for both fuels using both models are given in the following table:

Model	CENPD-132-S3	CENPD-132-S4	CENPD-132-S4
Limiting Break Size/Location	0.6 DEG/PD	0.6 DEG/PD	0.8 DEG/PD
Cladding	Zirconium	Zirconium	Vantage+
PCT	2174 ^oF	2110 ^oF	2087 ^oF
Max. Local Oxidation	8.37%	7.6%	12.0%
Max.Total Core-Wide Oxidation (All Fuel)	<0.86%	<0.57%	<0.73%

(DEG= double-ended guillotine; PD= pump discharge.)

The licensee provided results of analyses Westinghouse performed on its behalf for PVNGS Units 1 and 3 SBLOCA at the RSG and PUR conditions using both the CENPD-137, Supplement 1-P-A (S1M) SBLOCA methodology and CENPD-137, Supplement 2-P-A (S2M) SBLOCA methodology. While the licensee states that both Zircaloy and ZIRLO-clad fuels were explicitly treated, it provided bounding values only for the S2M analysis.

The licensee provided the results of a sensitivity study it had performed similar to the sensitivity analyses presented in CENPD-137 Supplement 2-P-A, Appendix E, page 2, Response 1a. The licensee's analyses demonstrate that the S2M SBLOCA methodology applies specifically to PVNGS Units 1 and 3 operating at the proposed PUR. Therefore, the NRC staff concludes that the S2M SBLOCA methodology applies specifically to PVNGS Units 1 and 3 operating at the proposed PUR.

Model	S1M	S2M
Limiting Break Size	0.05ft^2 PD	0.05ft^2 PD
PCT	1907 °F	1618 °F
Max. Local Oxidation	3.57%	1.28%
Max. Total Core-Wide Oxidation (All Fuel)	<0.57%	<0.2%

The licensee states that calculated post-LOCA oxidation for the ZIRLO-clad fuel bounds that calculated for the resident fuel despite having a lower calculated PCT. At the NRC staff's request, the licensee also addressed the concern that the resident fuel may have preexisting oxidation that needs to be considered in estimating the total LOCA oxidation. The licensee provided a response to the concern, including reference to information in the CE topical report CEN-386-P-A. The NRC staff concludes from the results of analyses identified above, and the information contained in the report referred to in the licensee's response to LOCA oxidation concern, that the licensee has substantiated its conclusion that the LOCA analyses for PVNGS Units 1 and 3 operating at PUR conditions take into consideration the total LOCA oxidation and meet the oxidation criteria of 10 CFR 50.46(b)(2). Therefore, the NRC finds that the LOCA analyses for PVNGS Units 1 and 3 operating at PUR conditions have considered the total LOCA oxidation, including preexisting oxidation, and meet the oxidation criteria of 10 CFR 50.46(b)(2).

The NRC staff also notes that the pre-existing oxidation of the fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation. Therefore, the NRC staff concludes that the core-wide hydrogen generation analyses results reported above demonstrate that PVNGS Units 1 and 3 operating at PUR conditions meets the core-wide hydrogen generation criterion of 10 CFR 50.46(b)(3).

As discussed above, the licensee has performed LBLOCA and SBLOCA analyses for PVNGS Units 1 and 3 at an uprated power of 3990 MWt using approved Westinghouse/CE methodologies. The licensee's LBLOCA and SBLOCA calculations demonstrated the following:

- The calculated LBLOCA and SBLOCA values for PCT, oxidation, and core-wide hydrogen generation are less than the limits of 2200 °F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1)-(3), respectively.

- Compliance with 50.46(b)(1)-(3) and (5) assures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). (The NRC staff notes that other matters that

could affect coolable geometry are not involved in the requested amendment.)

In summary, the NRC staff concludes that the licensee's LOCA analyses were performed with LOCA methodologies that apply to PVNGS Units 1 and 3 and demonstrate that PVNGS Units 1and 3 complies with the requirements of 10 CFR 50.46 (b)(1)-(4). Therefore, the NRC staff finds the licensee's LOCA analyses acceptable. Compliance with the long-term cooling requirement of 10 CFR 50.46(b)(5) is discussed in Section 4.1.1 below.

4.1.1 Post Loss-of-Coolant Accident Long-Term Cooling

Regulatory requirements for long-term cooling (LTC) are provided in 10 CFR 50.46(b)(5), which states, "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." In practice, following successful calculated blowdown, refill, and reflood after initiation of a LOCA, the LTC requirement will be met if the fuel cladding remains in contact with water so that the fuel cladding temperature remains essentially at or below the saturation temperature. A potential challenge to long-term cooling is that boric acid (H_3BO_3) could accumulate within the reactor vessel, precipitate, and block water needed to keep the fuel cladding wetted by water. The NRC staff reviewed the licensee's approach to control H_3BO_3 during LTC.

The licensee's July 9, 2004, request stated that the analysis of record for long-term cooling is based on a core power of 4070 MWt (3990 MWt plus 2 percent measurement uncertainty). This is consistent with the licensee's description of LOCA analysis assumptions contained in its January 2003, Revision 11 of the UFSAR.

The licensee stated that the analysis used for the PUR application used the H_3BO_3 precipitation evaluation model described in topical report CENPD-254, "Post-LOCA Long Term Cooling Evaluation Model," dated June 1980. The NRC staff approved this topical report by SE dated July 30, 1979. However, the NRC staff readdressed this model in an April 24, 2002, letter to Entergy Operations titled "Arkansas Nuclear One, Unit No. 2 - Issuance of Amendment Re: Increase in Licensed Power Level." In that letter, the NRC staff identified concerns with respect to the meaning of conservatism in determination of vessel mixing volume, the inconsistency between the Appendix K decay heat generation requirement and that of CENPD-254, and failure to include a 4 weight percent (wt%) H_3BO_3 margin to account for uncertainty. The NRC staff concluded then that these questions would be addressed on a generic basis consistent with the evaluation of CENPD-254, and that the evaluation of the Entergy model would be assessed by alternate means. In the interim, until this generic concern associated with LTC is resolved, the NRC staff will continue to accept the licensee's emergency operating procedures (EOPs) to initiate hot leg injection within 2-3 hours of the LBLOCA.

The NRC staff is using the same alternative approach for evaluating the PVNGS Units 1 and 3 H_3BO_3 assessment pertaining to LTC. The alternative approach addresses the probability of conditions where significant H_3BO_3 accumulation may be encountered, the current NRC staff review of the CENPD-254 assumptions, insights from predictions of significant H_3BO_3 accumulation from other plants, and the licensee's EOPs to reduce the possibility that significant H_3BO_3 accumulation will be encountered. There are a number of additional assumptions (not listed above) within CENPD-254 that the NRC staff has accepted.

The NRC staff compared the PVNGS Units 1 and 3 characteristics with those of other plants. This approach is similar to the NRC staff's approach in evaluating the Arkansas Nuclear One (ANO) PUR referenced above. This comparison is summarized in the Table 3.1. The information shows that the PVNGS Units 1 and 3 PUR characteristics are consistent with those of Byron/Braidwood and ANO-2 with respect to initiation of hot leg injection.

Table 3.1 - Comparison of Characteristics

	Characteristic	Byron/ Braidwood 5% PUR	ANO-2 7.5% PUR	Requested PVNGS Units 1 and 3 PUR
1	Time to reach H_3BO_3 saturation (hours).	8.53 (5/4/01) 6.0 (4/12/02)	~2.4 to 7.3, depending on assumptions	~3.5 (UFSAR)
2	Power (MWt)	3587	3026	4070
3	Decay heat generation rate multiplier (dimensionless)	1 (5/4/01) 1.2 (4/12/02)	1.1	1.1
4	Assumed H_3BO_3 saturation limit (wt%)	23.53	27.6	30
5	Core plus upper plenum volume below hot leg (ft^3)	1072*	940	Multiplying power by mixing volume ratio gives approximately ANO power
6	Time to hot leg injection via emergency operating procedures (hours)**	Consistent with Item 1 prediction	2 to 4	2 to 3

*Value is from NUREG-1269, Loss of Residual Heat Removal System, Diablo Canyon Nuclear Power Plant, Unit 2, April 10, 1987, June 1987.
**EOPs are consistent with the sequence of events described in UFSAR Sections 6.3.2.7 and 6.3.3.

While the NRC staff cannot concur with the licensee's evaluation based on CENPD-254, the NRC staff believes that there is sufficient basis to approve the license amendment, while the questions on assumptions given above are addressed on a generic basis, for the following reasons:

- The low probability of a LBLOCA where conditions leading to significant H_3BO_3 accumulation may be encountered.

- The NRC staff recently reviewed and accepted most of CENPD-254 modeling assumptions. Only a few assumptions are subject to question, and the implications of those assumptions are understood and do not invalidate the NRC staff's finding.

- The licensee's predictions are reasonable when compared to the predictions for other plants.
- The licensee has emergency operating procedures to initiate hot leg injection within 2 to 3 hours to reduce the possibility that significant H_3BO_3 accumulation will be encountered.

Based on the above, the NRC staff concludes that the outstanding issues of certain modeling assumptions in CENPD-254 is not a significant safety concern that would prevent the NRC staff from approving the PUR amendment. The issue exists for the plant's current operating license and the probability for the conditions leading to significant H_3BO_3 accumulation are not significantly increased by the proposed amendment. Based on the four bullets above, the NRC staff concludes that the licensee has met the regulatory requirements for LTC under 10 CFR 50.46(b)(5) with respect to the requested increase in power, and the amendment in this regard is acceptable.

4.2 Containment Response Analysis

The containment building is the final barrier against the release of significant amounts of radioactive fission products. The containment response analyses are performed to demonstrate compliance with 10 CFR Part 50, Appendix A, GDC 16, “Containment Design,” and GDC 50, “Containment Design Basis,” to demonstrate that the design pressure and temperature conditions for the containment structure are not exceeded during design-basis accidents (DBAs). In addition, a long-term pressure response analysis is performed to demonstrate compliance with GDC 38, “Containment Heat Removal.”

The containment structure must be designed to withstand the pressure and temperature conditions resulting from a postulated LOCA and maintain a leaktight barrier. It must also be designed to withstand an MSLB. The analyses performed for containment response also define environmental envelopes for equipment qualification (EQ) and for mechanical and electrical equipment located within the containment. The proposed PUR and larger RSGs will both impact the containment response during DBAs.

4.2.1 Containment Structure

Section 6.2 of the PURLR discusses the potential impact on the containment building by RSGs and PUR conditions. The containment is designed to withstand a pressure of 60 psig and a maximum liner temperature of 300 °F. The RSGs and PUR result in an increase in the containment pressure and temperature during postulated accidents due to the following:

(1) The power increase results in an increase in the RCS average temperature (T_{ave}) and decay heat, which results in more energy being transferred to the containment via the LOCA break flow.

(2) The additional RCS inventory due to the larger SGs increases the mass transferred to the containment during the LOCA blowdown.

(3) The additional SG secondary inventory, larger heat transfer area, and a higher secondary operating pressure result in more energy being transferred to the containment for an MSLB.

(4) The increased power results in more FW, at a higher enthalpy, being delivered to the SG secondary for release into containment during an MSLB.

The licensee's evaluation determined that the peak containment pressure and liner temperature for a LOCA are 58 psig and 270.97 °F, respectively. The peak containment pressure and liner plate temperature for an MSLB are 41.29 psig and 252.16 °F, respectively (for containment design assessment). These remain below the design values for these containment parameters set forth in the AOR.

An additional consideration is whether the increase in Pa, the calculated peak containment internal pressure related to the design-basis LOCA, from 52 psig to 58 psig would require new containment leakage rate tests at the higher pressure before plant restart. After reviewing the applicable regulations and guidance documents, the staff finds that there is no requirement for new tests at the higher pressure before the plant can restart. When the tests next come due, on the normal schedule, they will be performed at the new value of Pa. The staff considers the previous tests, performed at the old value of Pa, to remain valid and constitute an adequate indication of the leak-tightness of the containment, until new tests are performed on the normal schedule.

The licensee stated that principal subcompartments of concern are the reactor cavity, SG subcompartments, and the pressurizer subcompartment. For the pressurizer subcompartment, for which the analyses assume a double-ended break of the pressurizer surge line, the licensee determined that the energy release rates for the RSGs and PUR condition are bounded by the original plant design. For the reactor cavity and SG subcompartments, the application of LBB eliminates the dynamic effects of RCS MCL pipe breaks; therefore, the original design bounds the predicted subcompartment accident pressure. For containment temperature effects, the licensee performed a qualitative assessment that compared the RSG/PUR containment and basemat concrete thermal profiles that were generated using the COPATTA computer code to those values assumed in the original Bechtel Structural Analysis Program (BSAP) for containment internal structures. The results of this comparison determined that the original BSAP analysis remains bounding for RSGs and PUR conditions.

Based on the above reasoning, the NRC staff concurs with the licensee's conclusion that the containment building remains acceptable for operation at RSGs and PUR conditions.

4.2.2 Loss-of-Coolant Accident Containment Analysis

The changes for the proposed PUR and RSGs that have the most impact on the containment response during these DBAs are:

1. The power increase would result in an increase in the RCS average temperature and the decay heat, which results in more energy being transferred to the containment.

2. The additional RCS inventory due to the larger RSGs increases the mass being

transferred to the containment.

3. The additional RSGs mass inventory, larger heat transfer area, and a higher secondary operating pressure result in more energy being transferred to the containment.

Consistent with the current design basis, three break types were investigated:

1. RCS RCP double-ended discharge leg slot break (DEDLSB)

2. RCS RCP double-ended suction leg slot break (DESLSB), and

3. Double-ended hot leg slot break (DEHLSB).

All three-break locations were analyzed assuming both minimum and maximum safety injection (SI) pump flows. The limiting single failure for these analyses is a loss of one train of the CSS.

The analyses of the LOCA events were initiated from 102 percent of PUR power level, 4070 MWt. The LOCA mass and energy releases (M&E) and containment response analyses were performed in accordance with the SRP Sections 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," respectively.

The M&E for the blowdown phase of the LOCA were obtained with the previously accepted CEFLASH-4A code (Reference 1). The M&E for the reflood and post-reflood phases of the LOCA were obtained with the previously accepted methodology (Reference 2) for the cold leg breaks. The reflood and post-reflood phases were not simulated for hot leg breaks. For hot leg breaks, most of the reflood fluid does not pass through a SG before being released to the containment and there are no physical mechanisms to rapidly remove the residual SG secondary energy during or after the reflood period.

Accounting for residual stored heat including decay heat in the primary and secondary systems, the M&E for the long-term phase of the LOCA were obtained with the previously accepted CONTRANS2 containment code (Reference 3).

The containment pressure and temperature profiles were calculated using the previously accepted COPATTA code (Reference 4) based on the M&E obtained for each break location. In reviewing the licensee's initial conditions for these analyses, it was noted that the PUR analyses were based on an initial containment relative humidity of 0 percent, while the AOR, UFSAR Table 6.2.1-6, were based on 50 percent.

The licensee provided justification for the PUR analysis. The limiting relative humidity was reduced to 0-percent to provide a conservative peak pressure calculation. Consistent with SRP guidelines, the most limiting initial condition was selected for the PUR submittal. A higher initial relative humidity within containment results in a lower pressure profile and a lower peak pressure value. The additional mass of water vapor, at 50 percent relative humidity, in the containment atmosphere acts as a heat sink. A comparison of COPATTA computer runs performed by the licensee showed a reduction of 0.37 psi in the peak pressure if a value of 50 percent relative humidity was used. Therefore, assuming 0 percent relative humidity

maximizes the containment pressure response and is conservative.

The NRC staff also noted that in UFSAR Table 6.2.1-6, the RCS expansion multiplier for the LOCA was stated to be 3 percent, while in the PUR submittal, the volumetric expansion multiplier value was stated to be 2 percent. The licensee indicated that UFSAR Table 6.2.1-6 is incorrect and the error would be corrected during the next update. The RCS expansion multiplier for the existing AOR, and for the PUR, is 2 percent.

The DEDLSB LOCA with maximum ECCS was identified as the pipe break with the highest peak pressure for the PUR and with RSGs. The resulting peak pressure is less than 58 psig (73 psia), and remains below the design pressure value of 60 psig. The resulting peak vapor temperature was calculated to be 308.4 ^{o}F.

The impact of PUR and RSG conditions on the peak containment liner temperature was evaluated by the licensee. The peak liner temperature was predicted to be less than 271 ^{o}F. This temperature is below the containment building liner design peak temperature of 300 ^{o}F.

Long-term analyses of the worst case DEDLSB, and of the worst-case pump suction leg break were performed by the licensee to verify the ability of the containment heat removal system (CHRS) to maintain the containment pressure and temperature below the design conditions. These evaluations were based on a conservative performance analysis of the engineered safety features. The CHRS long-term operating mode included one CSS train. The analyses showed that within 24 hours the containment pressure was reduced to less than one half of the peak containment pressure.

Based on the reasoning set forth above, the NRC staff finds there is reasonable assurance that PVNGS Units 1 and 3 will continue to be in compliance with GDC 16, “Containment Design,” GDC 50, “Containment Design Basis,” and GDC 38, “Containment Heat Removal,” at the proposed PUR power level with the RSGs following a DBA LOCA.

4.2.3 Main Steam Line Break Containment Analysis

The changes for the proposed PUR and RSGs that have the most impact on the containment response during MSLB DBAs are:

1. The power increase would result in an increase in the RCS average temperature and the decay heat, which results in more energy being transferred to the SGs, and increases the severity of the MSLB blowdown to the containment.

2. The additional RSGs secondary inventory and a higher secondary operating pressure result in increased mass and additional energy being transferred to the containment.

3. The increased power results in more FW, at a higher enthalpy, being delivered to the SGs for eventual release to the containment.

Consistent with the current design basis, three power levels were investigated to evaluate the overall containment response, the containment pressure and the EQ temperature profile. These analyses were performed by the licensee at 102 percent, 75 percent, and 0 percent of

the proposed PUR power level, and included the RSGs.

The MSLB M&E and containment response analyses were performed in accordance with Sections 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” and 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," respectively, of the SRP. NRC Bulletins and Information Notices were also considered as part of the licensee’s analyses: Bulletin 80-04 (Reference 5) for the treatment of main FW addition, and Information Notice 84-90 (Reference 6) for the effects of main stream line breaks on EQ. The guidance provided in NUREG-0588 (Reference 7) for the treatment of EQ cases was also included in the licensee’s analyses.

The previously accepted SGNIII code (Reference 8 and Reference 9) was used to generate the M&E release data for the blowdown phase of the MSLB. The analyses conservatively assumed the availability of non-emergency power. This allows for continued RCP operation and maximizes the rate of heat transfer to the affected SG maximizing the rate of mass and energy release. With non-emergency power available, an emergency diesel generator failure is not limiting and need not be postulated.

There is a main steam isolation valve (MSIV) in each of the four main steam lines. Each valve has dual solenoid valves to assure closure even with a single failure in the control system. Single failure of the actuation signal will not prevent valve closure since both trains of main steam isolation signal actuation are provided to each MSIV. Any failure would result in the valve going to the closed position. The other MSIV isolates the unaffected SG. For the licensing analyses, the licensee assumed a random failure of an MSIV in the broken steam line. This maximizes the forward and reverse flow to the break and maximizes the consequences of the event.

There are two FW isolation valves (FWIVs) in series in each of the two main FW lines. If one FWIV fails, the second FWIV will provide isolation. All cases were analyzed considering the flashing of fluid in the lines from the FWIVs to the affected SG and there was no need to do a separate analysis assuming FWIV failure.

The containment pressure and temperature responses to an MSLB were obtained with the previously accepted COPATTA code. The primary differences between the LOCA analysis and the MSLB analysis are:

1. For the MSLB, the M&E release calculations terminate when the affected SG dries out, and

2. The Uchida correlation is used for the heat transfer coefficient to the structural heat sinks in the MSLB, while the Tagami correlation is used for the LOCA, consistent with the guidance provided in the SRP.

In addition to these parameters, the time for CSS actuation was re-analyzed by the licensee. The existing analysis, the AOR, conservatively bounded the time to reach the containment high-high-pressure setpoint of 10 psig. For the 102 percent power case with the PUR configuration, the time to reach the pressure setpoint of 10 psig was calculated with the COPATTA code to be approximately 8 seconds. With an instrument and equipment delay time

of 82 seconds, the CSS would start quenching the containment environment at about 90 seconds into the accident, or about 11.5 seconds earlier than in the existing AOR. The licensee concluded that the cumulative effect on the containment response to a DBA MSLB from the increased blowdown due to the PUR was offset by the reduction in the time for CSS actuation.

The MSLB from 102 percent of the proposed PUR rated thermal power is the limiting case for both the containment pressure and for the equipment qualification temperature profile. The peak pressure in the containment building was predicted to be 41.29 psig without the EQ analysis guidance and 41.75 psig with the EQ analysis guidance. This peak pressure is below the containment building design peak pressure of 60 psig.

No long-term analysis was performed for the MSLB since after isolation and blowdown there is no further energy input to containment. The maximum peak containment vapor temperature was calculated to be 405.55 ^{o}F and occurred at 90 seconds for the limiting 102 percent power MSLB containment design case.

The impact of the PUR and RSG conditions on the peak containment liner temperature was evaluated by the licensee. The peak liner temperature was predicted to be about 252.2 ^{o}F. This temperature is below the containment building liner design peak temperature of 300 ^{o}F.

Based on the above, the NRC staff finds that PVNGS Unit 1 and 3 will continue to be in compliance with GDC 16, “Containment Design,” and GDC 50, “Containment Design Basis,” at the proposed PUR power level with RSGs following a DBA MSLB.

4.2.4 Main Steam Line Break Outside Containment Analysis

The proposed PUR and the larger SG volume and larger heat transfer area have the potential to affect the outside containment response to an MSLB for the same reasons discussed in the MSLB Containment section above. An analysis was performed by the licensee to verify the EQ temperature envelope in the main steam support structure. A 1-ft^2 non-mechanistic steam line break was analyzed by the licensee to quantify the effect of the PUR for this limiting break. The analyses were performed in accordance with Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” of the SRP. NRC Bulletins and Information Notices were also considered as part of the licensee’s analyses: Bulletin 80-04 for the treatment of main FW addition, and Information Notice 84-90 for the effects of main stream line breaks on EQ. The guidance provided in NUREG-0588 for the treatment of EQ cases was also included in the licensee’s analyses. The purpose of this analysis was to demonstrate compliance with10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

The break was assumed to be at the first weld outside containment. This assumption minimizes the flow resistance between the break and the affected SG and increases the calculated M&E. The M&E were generated with the EQ analysis guidelines for this event. The analyses focused on M&E releases at 102 percent power and 0 percent power. These analyses included the assumption that the MSIV in the steam line with the least flow resistance fails to close following the isolation signal. This assumption maximizes the M&E release during this event. Superheating within the SG starts when the U-tubes uncover, as specified in NRC

Information Notice 84-90. The turbine stop valves were assumed to close instantaneously at the time of reactor trip. This assumption is conservative for this event because the entire steam inventory at the time of reactor trip is assumed to be forced out of the break. No leakage was assumed through the MSIVs or main FWIVs. The auxiliary feedwater (AFW) logic was assumed to function properly and to isolate all AFW to the affected SG.

There are differences between the methodology used for the new PUR analysis and the AOR methodology. These methodology differences include a reduction in some conservative input values selected in the AOR and a revised reactor trip methodology. Based on detailed analyses performed by the licensee, it was concluded that reactor trip on core protection calculator variable over-power trip and low SG pressure could be credited if the effect of the moderator temperature coefficient (MTC) was considered. The licensee's PUR analysis evaluated all reasonable reactor trips and identified the most conservative trip. The NRC staff finds the licensee's evaluation acceptable, as the resulting analysis used the most conservative reactor trip to develop conservative mass and energy release rates for the MSLB outside containment evaluation.

The AOR zero load case, the limiting case, used an MTC more negative than that allowed by the TS. This conservative MTC led to unrealistically high steam pressures during this event, preventing the actuation of the low SG pressure trip, resulting in temperatures which are higher than those achievable with the MTC allowed by the TS.

The PUR analyses were based on the MTC limit allowed by the TS, and the revised reactor trip methodology. The peak temperature in the main steam support structure based on the M&E releases calculated for the PUR are bounded by the peak temperature values generated in the AOR. The M&E released during the 0 percent PUR case was found to be bounded by the 0 percent power AOR. The NRC staff finds the licensee's use of the TS MTC limit acceptable since the MTC limit allowed by the TS is the most conservative value expected during plant operation, and the resulting analysis used this value to develop conservative M&E release rates for the MSLB outside containment evaluation.

The blowdown phase of the 102 percent power MSLB was simulated with a modified version of the SGNIII code to generate the M&E releases. The revision to the SGNIII code provided a better representation of the secondary side. The revised code provides more detailed modeling for the four main steam lines (as compared to the original analysis which modeled only two main steam lines), for the closing of the MSIVs and for the steam flow through the main steam line cross header path following the closure of MSIVs.

The change to the SGNIII code was conducted under the provisions of the Westinghouse Quality Assurance Program. This change provides the analyst with the ability to use one flow resistance and flow area through the cross header prior to the turbine stop valves closing and a different set of values following the closure of the turbine stop valves. This change provides better modeling of the steam header crossover path by providing a steaming path to the break following the closure of the turbine stop valves and prior to the MSIVs closing. The model change verification and validation are documented in a calculation performed and owned by Westinghouse. The code's response was compared to existing data and found to yield the expected results. The NRC staff finds the change to the SGNIII code to model the four steam lines acceptable since the change only involved a better representation of the physical system

without changes to either the underlying numerical solution schemes or to the conservative selection of the thermal-hydraulic models and input values used for this licensing analysis.

As a result of the change to SGNIII, after 300 seconds into the event, the PUR M&E release rate begins to decrease faster than the M&E release rate in the AOR. The temperature in the AOR continues to increase while the temperature in the PUR analysis is decreasing. The peak temperature for the PUR case occurs at 290 seconds, based on about 44×10^6 Btu of released energy. The temperature for the AOR does not peak until 400 seconds, based on about 49×10^6 Btu of released energy.

The subcompartment pressure and temperature model used for the PUR was identical to the existing AOR. The previously accepted PCFLUD code (Reference 10) was used to evaluate the subcompartment response to the MSLB.

The peak temperature in the main steam support structure occurred for the 0 percent power AOR case. The peak temperature was calculated to be 383 °F, and for comparison the PUR temperature was calculated to be 367 °F. For the 102 percent power case, the AOR peak temperature was calculated to be 373 °F, and for comparison the PUR temperature was calculated to be 357 °F. The licensee has determined that the EQ profile remains limited by the current AOR, the 0 percent power case. The peak pressure in the main steam support structure remains bounded by the AOR, with a calculated value about 5 psi below the design value.

Based on the above, the NRC staff finds that PVNGS Unit 1 and 3 will continue to be in compliance with 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," at the proposed PUR power level with the RSGs following a DBA MSLB outside containment.

4.2.5 Subcompartment Loads

The evaluation of subcompartment loads as described in SRP Section 6.2.1.2, "Subcompartment Analysis," was not specifically addressed in the PURLR. The licensee's UFSAR Section 6.2.1.1.1.1, "Containment Structure Accident Conditions," states, "These analyses were performed at 102% of Licensed Power." In UFSAR Table 6.2.1-6, the reactor power is stated to be 3954 MWt. The PUR power level at 102 percent is 4070 MWt.

The short-term LOCA-related M&E releases are used as input to the subcompartment analyses that are performed to ensure that the walls of a subcompartment can maintain its structural integrity during the short pressure pulse (generally less than three seconds) accompanying a high energy line pipe rupture within that subcompartment. The subcompartments evaluated include the SG compartment, the reactor cavity region, and the pressurizer compartment.

PVNGS Units 1 and 3 are approved to use a LBB methodology, and for the SG compartment and the reactor cavity region, LBB was used to qualitatively demonstrate that any changes associated with operation at the PUR conditions would be offset by the LBB benefit of using smaller RCS nozzle breaks. The licensee stated that the current licensing bases for these compartments remain bounding.

The pressurizer subcompartment analysis assumes a double-ended guillotine break of the pressurizer surge line. The energy release rates were calculated for the proposed PUR condition and compared with the original plant design conditions. The original plant design energy release rates continue to bound the PUR energy release rates by approximately 10 percent. Since the analysis performed for the initial plant licensing found the pressurizer subcompartment adequate, and the energy release rates for the PUR condition remain bounded by the original energy release rates, the licensee stated that the pressurizer subcompartment remains structurally acceptable.

The NRC staff finds there is reasonable assurance that PVNGS Units 1 and 3 will continue to be in compliance with GDC 4, "Environmental and Dynamic Effects Design Bases" and GDC 50, "Containment Design Basis," at the proposed PUR conditions since the AOR energy release rates are approximately 10 percent higher than those expected at the PUR power level and the structural integrity of the limiting subcompartment was previously found acceptable by the NRC staff.

4.2.6 Minimum Containment Pressure Analysis for ECCS Performance Capability

The evaluation of the minimum pressure for ECCS performance as described in SRP 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," was not specifically addressed in the licensee’s submittal. UFSAR Section 6.2.1.5, “Minimum Containment Pressure Analysis for ECCS Performance Capability Studies," states that "a minimum containment pressure analysis was completed in the 1995-1996 time frame, to support an ‘ECCS break spectrum’ analysis for a licensed, rated thermal power of 3876 MWt. This analysis was revised in 2000, when the 'ECCS limiting break reanalysis' utilized more conservative containment heat sink values." The PUR power level at 102 percent is 4070 MWt.

As stated in Section 6.1.1 of the PURLR, the existing ECCS performance analysis was performed for a core power of 4070 MWt (3990 MWt plus 2 percent measurement uncertainty). A rated core power of 3990 MWt, which is a 5 percent increase in the original licensed power of 3800 MWt, was first used in the ECCS performance analysis performed for the 2 percent stretch power license amendment request (Reference 11) and is the rated core power that has been used in all subsequent ECCS performance analyses. The fact that the stretch power analysis was performed for a 5 percent increase in power was not described in the 2 percent stretch power license amendment, nor was the description of the LBLOCA ECCS performance analysis described in the PVNGS Units 1 and 3 UFSAR Section 6.3.3 .

Consistent with the core power used in the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, the minimum containment pressure analysis in UFSAR Section 6.2.1.5 also used a core power of 4070 MWt. This is true for the minimum containment pressure analysis for both the ECCS break spectrum analysis and the ECCS limiting break reanalysis. Like the description of the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, this fact is also not currently described in UFSAR Section 6.2.1.5.

Since the minimum containment pressure analysis described in UFSAR Section 6.2.1.5 was performed at the uprated core power of 4070 MWt, the NRC staff finds that reanalysis of the minimum pressure for ECCS performance to demonstrate continued compliance with 10 CFR 50.46 is not necessary, rather, the current analyses are bounding for the PUR conditions.

4.3 Non-LOCA Transient Analysis

The licensee performed analyses at 4070 MWt (3990 MWt plus 2 percent measurement uncertainty) accounting for installation of the RSGs. The Combustion Engineering Nuclear Transient Simulator (CENTS) code is used for the calculation of plant response to the non-LOCA transients. The NRC staff had previously approved the CENTS code for determination of operating limits in Amendment 137 of PVNGS Unit 1's and Unit 3's operating license. The design-basis events (DBEs) discussed in this section are classified into three categories depending upon the expected frequency of occurrence, i.e., anticipated operational occurrences (AOOs), infrequent events, and limiting faults. The following non-LOCA transient events are discussed:

UFSAR Section	Submittal Section	Transient Event
15.1	6.3.1	Increase in Heat Removal by the Secondary System
15.2	6.3.2	Decrease in Heat Removal by the Secondary System
15.3	6.3.3	Decrease in RCS Flowrate
15.4	6.3.4	Reactivity and Power Distribution Anomalies
15.5	6.3.5	Increase in RCS Inventory
15.6	6.3.6	Decrease in RCS Inventory
	6.3.8	Limiting Infrequent Events

Additional assumptions are as follows: The fission source used for CENTS is ANS/ANSI 5.1-1979, which includes a 2σ uncertainty. The minimum departure from nucleate boiling ratio (DNBR) and departure from nucleate boiling (DNB) were determined using the CETOP-D Code. For transients that include loss of RCS flow, DNBR is determined using a more detailed code called TORC. For control element assembly (CEA) ejection events, cladding and fuel integrity is simulated using the STRIKIN-II Code. The loss of RCS flow and sheared RCP shaft event is simulated using the HERMITE Code, and finally, RCS coastdown following loss of power combined with a sheared RCP shaft and a seized RCP is simulated using the COAST Code. The fission source complies with the requirements of Appendix K to 10 CFR Part 50, and therefore, is acceptable. The codes used for DNBR, core physics, and thermal-hydraulics have been approved by the NRC staff for the ranges of conditions associated with the PUR, and are acceptable.

Initial core conditions and expected instrumentation and engineered safety system response will be identified in the discussion of each transient. It should be noted that PVNGS Units 1 and 3 is one of the CE plants for which the anticipated transients are assumed in the licensing analysis to coincide with a single failure, thus, elevating the severity of the transient. In this manner a degree of conservatism is built into the analysis. PVNGS Units 1 and 3 are not equipped with power-operated relief valves.

4.3.1 Increase in Heat Removal by the Secondary System

Analyses in this section include: decrease in FW temperature, increase in FW flow, increase in steam flow, inadvertent opening of a SG relief or safety valve, and steam bypass control system (SBCS) misoperation. Two of these transients, decrease in FW temperature and increase in FW flow, are classified as AOOs and are bounded by the SBCS misoperation, which is discussed in Section 4.3.1.1, “Increased Main Steam Flow.”

Excessive heat removal causes a decrease in moderator temperature, which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC’s acceptance criteria are based on the following general design criteria: (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) will not be exceeded during normal operations, including AOOs, (3) GDC 20, which requires that the reactor protection system be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs, and (4) GDC 26, which requires the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including during AOOs. Specific review criteria are contained in SRP Section 15.1.1-4.

4.3.1.1 Increased Main Steam Flow

Inadvertent increased opening of the turbine admission valve (TAV) or a malfunction of the SBCS can result in up to an 88 percent increase in nominal steam flow, which bounds an inadvertent opening either of a turbine bypass valve (TBV) or of an atmospheric dump valve (ADV). The increase in main steam flow will decrease RCS temperature and pressure, decrease SG pressure, and increase core power and heat flux.

The reactor protection system and the engineered safety system will initiate a reactor trip on high reactor power or low RCS pressure or low SG pressure. The trip signal should protect against violating any of the SAFDLs i.e., [the minimum departure from nucleate boiling ratio (MDNBR)] or the local power limits.

The acceptance criteria (as defined in the SRP Section 15.1.1) are: (1) RCS pressure should be maintained below 110 percent of the vessel design pressure for an incident of moderate frequency, (2) fuel cladding integrity should be maintained by assuring that Criterion 1 of SRP 4.4 is met (i.e., DNBR is maintained above the MDNBR value), (3) an incident of moderate frequency should not generate a more serious plant condition, and (4) an incident of moderate frequency in combination with any active component (single) failure should not result in loss of function of any fuel barrier other than cladding. A limited number of cladding failures are acceptable.

The analysis was done using the CENTS code. The CETOP-D code, DNBR with the CE-1 critical heat flux (CHF) correlation, was used for the DNBR. The system was initialized at 102 percent the power operating limit. No operator action is assumed for the first 30-minutes after transient initiation.

The increased steam flow leads to a reduction in core inlet temperature and an increase in core power, resulting in reactor trip and closure of the MSIVs. The MDNBR is 1.40, which is above the SAFDL limit of 1.34. Pressures in the RCS and the SG decrease. No other failure is induced by this transient. All of the acceptance criteria are met.

4.3.1.2 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV)

The acceptance criteria and the analysis method and codes are the same as in PURLR Section 6.3.1.3. The analysis models IOSGADV with a loss of power (LOP) following turbine trip. Because of reduced RCS temperature, the resulting power increase will cause a trip via the core protection calculator (CPC). Depending on the core burnup, it is possible for a low SG level trip to be generated earlier than the core power excursion trip. The analysis assumes that the RSGs would reach trip setpoint pressure considerably later compared to the existing setpoint. The MDNBR value is 1.37, which is above the 1.34 acceptance limit. The maximum pressure will remain below 110 percent of the design limit. All of the acceptance criteria are satisfied.

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described in Sections 4.3.1.1 and 4.3.1.2 above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed PUR level and were performed using acceptable analytical models. For the reasons set forth above, the NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 for such events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the events stated above.

4.3.1.3 Steam System Piping Failures Inside and Outside Containment

The steam release from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core

reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed responses of the reactor coolant and auxiliary systems, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, (7) core power excursion due to power demand created by excessive steam flow, (8) variables influencing neutronics, and (9) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27, which requires that the reactivity control systems be designed, with appropriate margin for stuck rods, to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized, and (4) GDC 35, which requires that the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5.

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

The break of a main steam pipe causes a significant increase in steam flow and energy removal, resulting in corresponding decreases in RCS inlet temperature and pressure.

The licensee investigated a spectrum of MSLBs, i.e., with and without LOP and from full and zero power. In addition, a stuck CEA and a high-pressure injection pump failure (single failure) is assumed. One acceptance criterion is that core coolability be maintained. For PVNGS Units 1 and 3, the MSLB is a limiting fault event; maximum RCS pressure should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. Thus, the nil ductility temperature RT_{NDT}, of 10 CFR 50.61 is also an acceptance criterion. The potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above SAFDL limits. If the minimum DNBR falls below SAFDL values, fuel damage would need to be assessed. Some fuel damage is acceptable as long as the core remains in place and maintains a coolable geometry.

The CENTS code was used in these evaluations with three-dimensional reactivity feedback. This capability is based on the HERMITE code, which has been approved by the NRC staff (Reference 9-42 of the PURLR in the licensee's December 21, 2001, PVNGS Unit 2 PUR submittal).

Input parameters, such as the most negative moderator and fuel feedback coefficients were chosen to maximize the possibility of return to power after shutdown.

The results show that the MSLB transient at full power with LOP is the most limiting. The reactor trip signal would be generated by the CPC due to decreasing RCP speed. Other trips that could trip the reactor in this transient are low SG pressure, high SG differential pressure, low RCS flow, and high containment pressure. The rising SG level will lead to FW isolation valve and MSIV closings.

The results also show that the RCS remains below acceptable design limits because the pressurizer safety valves (PSVs) and MSSVs will actuate, and the minimum DNBR remains above the SAFDL limit. Coolability and pressure vessel integrity are assured. Vessel protection against brittle fracture is assured because the end of life reference nil ductility temperature is 78 °F which is below the 270 °F screening criterion of 10 CFR 50.61, and therefore, all of the acceptance criteria are met.

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

MSLB events at hot standby are analyzed to demonstrate adequacy of the shutdown margin. MSLB transients with and without LOP are analyzed assuming one HPSI system has failed. The remaining parameters are chosen as in 4.3.1.3.1 above, i.e., to maximize the possibility for a return to power after trip initiation. The CENTS code is supplemented by the minimum DNBR code HRISE.

The power surge resulting from the cool-down will cause a trip signal from either low SG pressure or a high log power trip. These results indicate that the case with LOP is limiting and that the acceptance criteria are met for vessel pressure and minimum DNBR. Because of enhanced cooling, the RSGs will require more shutdown margin, however, the analysis indicates that there is sufficient shutdown margin and that the plant will not return to power.

The NRC staff has reviewed the licensee’s analyses of steam system piping failure events as set forth in Sections 4.3.1.3.1 and 4.3.1.3.2, and concludes that the licensee’s analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner because the reference temperature for the nil ductility transition is 78 °F (below the 10 CFR 50.61 screening criterion), and abundant core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 31, and 35 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to steam system piping failures.

4.3.1.4 Pre-Trip Main Steam Line Break Power Excursion

During a MSLB, a damaging power spike before a reactor trip is possible. For example, if the reactor is operating at power levels less than the rated power, the power level and power-rate based shutdown signals may be delayed for a short period of time. In this situation MDNBR and fuel damage are greater concerns than exceeding the vessel pressure limits. The NRC’s acceptance criteria are based on (1) GDC 10 which requires that the RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded during normal operation including AOOs, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations including AOOs, and (3) GDC 26, which requires the reliable control of reactivity changes to ensure that the SAFDLs are not exceeded.

The transients examined in cases 4.3.1.3.1 and 4.3.1.3.2 above were designed to maximize the return to power after a trip. This case maximizes the power excursion before reactor trip and the possibility of DNBR challenge to fuel integrity.

The licensee's analysis was based on the CENTS code, with the initial margin and DNBR calculated using CETOP-D, which is based on the CE-1 CHF correlation. At the time of DNBR a more accurate DNBR determination is made using the TORC code.

The CPC VOPT provides an early trip for small SLBs, but for larger SLBs and limiting MTC, the power overshoot may reduce the DNBR below the limit of fuel damage. Full power was selected at 95 percent because the same thermal margin to DNB and SAFDL exists as at 100 percent, but the CPC VOPT setpoint will have room to increase in response to a power surge. The limiting case is from full (95 percent) power with offsite power available. Operator action at 30 minutes after event initiation will initiate cool-down.

The results show that the peak reactor power reaches 118.6 percent of full power and the MDNBR is 1.35. Primary and secondary pressures remain within 110 percent of design values. The screening criteria are met and the power excursion before reactor trip does not cause either the SAFDL or the RCS maximum pressure criteria to be exceeded.

The NRC staff has reviewed the licensee's analyses of the pre-trip MSLB power excursion and, for the reasons set forth above, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, and core cooling will be provided for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the pre-trip MSLB power excursion.

4.3.2 Decrease of Heat Removal by the Secondary System

Loss of external load, turbine trip, inadvertent closure of MSIV, and loss of condenser vacuum will result in decrease of heat removal by the secondary system. These events result in a sudden reduction in steam flow and consequently cause RCS pressurization. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that the SAFDL are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including anticipated operational occurrences, and (3) GDC 26, which requires the reliable control of reactivity changes to ensure that the SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.1-5.

These transients are characterized as AOOs. Loss of external load will generate a turbine trip, which will result in closure of the turbine stop valves. The SBCS and the Reactor Power Cutback System can accommodate the excess steam without reactor trip. However, if these systems are in the manual mode and a turbine trip takes place, the MSSVs will limit secondary overpressure. Loss of condenser vacuum will also result in a turbine trip, which in addition will close the turbine stop valves and will trip the main FW pumps on high backpressure. Primary and secondary pressures will increase very fast, resulting in a reactor trip. The PSVs and the MSSVs will lift and keep primary and secondary pressure within limits (Note: this plant does not have power-operated relief valves). Loss of condenser vacuum is the most limiting transient for decreased heat removal by the secondary system. Loss of external load, turbine trip, inadvertent closure of the steam isolation valve, and loss of condenser vacuum were optimized to result in maximum primary and secondary pressure.

The RCS response was analyzed using the CENTS code, while DNBR was calculated using the CETOP-D code with the CE-1 CHF correlation. The system was initialized at 102 percent power. Loss of condenser vacuum was simulated with turbine trip, TAV closure and main FW flow ramping to zero. TAV closure time and main FW flow ramp period were conservatively selected to bound the actual plant configuration. High pressurizer pressure trip (HPPT) will follow with PSVs and MSSVs opening. Active single failures were considered; however, there are no single failures which would degrade the performance of the PSVs and MSSVs (Spring-loaded PSVs and MSSVs are not assumed to fail). LOP does not increase the peak power, therefore, no single failure is assumed. No operator action is assumed for the first 30 minutes after transient initiation.

The results show that the minimum DNBR remains above the original value and well above the SAFDL limit. The PSVs will open at the setpoints defined in the Technical Specifications, and the maximum primary pressure will peak at 2733 psia, i.e., below the safety limit of 2750 psia.

MSSV will open at setpoints defined in the Technical Specifications. The maximum secondary pressure will be limited to 1399 psia, which is lower than 110 percent of design pressure. Thus, the acceptance criteria have been met for fuel failure, pressure vessel maximum pressure, and SG maximum pressure.

4.3.2.1 Main Steam Isolation Valve Closure

Closure of all MSIVs could take place on a spurious signal. A decrease in heat removal results in an increases in primary and secondary temperature and pressure. Reactor trip will follow and the PSVs and the MSSVs will limit maximum pressures in the primary and the secondary system. The transient evolves as, and is bounded by, a loss of condenser vacuum.

The NRC staff has reviewed the licensee’s analyses of the decrease in heat removal events described above and concludes that the licensee’s analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, and 26 with respect to these events following implementation of the proposed

PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the events stated.

4.3.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

LOP to the auxiliaries can result from either LOP or loss of the onsite distribution system. On loss of Alternating current (AC) power the turbine stop valves close and FW flow to both SGs drops to zero. The RCPs will coast down and the plant will trip on DNBR. Primary and secondary pressure increases are limited from the PSVs and MSSVs. The emergency diesel generators (EDGs) will be activated to provide sufficient power for the operation of the safety equipment, including auxiliary FW. Operator control of the ADVs will regulate pressure in the SGs. This event is classified as an AOO and is bounded by the loss of condenser vacuum.

4.3.2.3 Loss of Normal Feedwater Flow (LOFW)

LOFW from the loss of one or two FW pumps will result in decreasing level and increasing pressure in the SGs. Primary temperature and pressure will rise until a reactor trip occurs from either low SG level or high pressurizer pressure. The trip signal will close the turbine stop valves, the MSSVs will control pressure in the secondary, and the low SG level will activate the AFW. The cool-down is operator controlled via the steam bypass and the condenser. This event is classified as an AOO and is bounded by the loss of condenser vacuum. Therefore, it is not reanalyzed here.

4.3.2.4 Feedwater System Pipe Breaks

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cool-down (by excessive energy discharge through the break) or a RCS heat-up (by reducing FW flow to the affected SG). In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed response of the reactor coolant and auxiliary systems, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27 which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized, and (4) GDC 35, which requires that the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.2.8.

For PVNGS Units 1 and 3, breaks up to 0.2 ft^2 are characterized as small and excess FW capacity is sufficient to supply the break without adverse effect on the SG feed. A large break upstream of the check valve can result in a partial or total loss of FW flow. A break

downstream of the check valve has the potential to allow reverse flow from the SG. Depending on the temperature of the water in the reverse flow, an RCS cool-down or heat-up could result. Such a cool-down is bounded by the MSLB, thus, it is not considered here; therefore, the FW line break is examined as a heat-up event.

The event in its most conservative version can be described as a FW line break downstream of the check valve, loss of all FW flow, and reverse flow through the break. The MSSVs are initially open, but SG dry out and high RCS pressure will result from the transient; a low SG level trip would be generated or a high containment pressure or a high pressurizer pressure (The HPPT is the most conservative). The PSVs will lower pressure in the RCS. It is assumed that offsite power is lost several seconds after the turbine trip. The onsite diesels will be activated and the low SG level will activate AFW to refill the SGs. Operator manual cool-down commences at 30 minutes from transient initiation.

Because this is a limiting low-probability event, vessel pressure should be maintained below 120 percent of the design limit and DNBR should be maintained above the SAFDL limit. Fuel damage should be assessed for the part of the fuel which is calculated to fall below DNBR, and the core should remain coolable.

The CENTS code was used for the analysis, and CETOP-D was used for DNBR (incorporating the CE-1 critical heat flux correlation). Two cases were analyzed for maximum RCS pressure and core integrity and long-term cooling vs AFW capacity.

The transient simulation used limiting values to conservatively maximize RCS pressure and fuel damage. The limiting break size is conservatively determined based on the approved method, i.e., the break size is that which results in a simultaneous HPPT and Low SG Level Trip. A turbine trip will immediately follow and LOP is assumed concurrent with turbine trip. The LOP has a maximum effect on degrading RCS-to-SG heat transfer, thus maximizing RCS pressure. Single failures were considered, and the most limiting would be PSV or MSSV failure to inhibit depressurization, however, there are no credible failures for these valves. No other single failures combined with the LOP would result in a more severe transient. No operator action is assumed for the first 30 minutes following transient initiation.

The sequence of events reveals that the sudden reduction of the primary-to-secondary heat transfer caused by decrease in SG inventory and LOFW is compounded by a turbine trip caused by the rapid heat-up and the coincident LOP. However, the PSVs and the MSSVs open to provide pressure relief and cool-down. Pressure is limited to below 120 percent of the design value and the minimum DNBR is about 1.50, which is well above the limit of 1.34.

The NRC staff has reviewed the licensee’s analyses of FW system pipe breaks and concludes, as described above, that the licensee’s analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue

to meet the requirements of GDCs 27, 28, 31, and 35 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to FW system pipe breaks.

4.3.3 Decrease in Reactor Coolant Flow

4.3.3.1 Total Loss of Reactor Coolant Flow

A decrease in reactor coolant flow or a total loss of coolant flow, occurring while the plant is at power, could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if the SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including anticipated operational occurrences, and (3) GDC 26, which requires that there are reliable controls of reactivity change to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.3.1-2.

Total loss of coolant flow would result from LOP to all RCPs and simultaneous turbine trip and loss of the steam dump and bypass system. Loss of coolant flow results in loss of heat transfer, increase of primary temperature and pressure and low DNBR. Depressurization is accomplished through the MSSVs and the ADVs. The major concern is the MDNBR, which occurs a few seconds after transient initiation. Because of the very short time from transient initiation to MDNBR, no single failure would make the MDNBR value smaller. Therefore, no single failure is assumed. However, initial values were chosen to minimize the DNBR.

Because this is rated as an event of moderate frequency, the maximum RCS pressure should be maintained below 110 percent of its design value. DNBR should be greater than the limiting value.

The analysis is based on the same codes and the same conservative assumptions as Section 6.3.2 of the PURLR. The results indicate that the acceptance criteria are satisfied in that the MDNBR is above the SAFDL limit; thus, cladding integrity is preserved and the maximum RCS pressure remains well below the limit of 110 percent of design value, assuring vessel integrity.

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes, as set forth above, that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has

demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 for such an event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the decrease in reactor coolant flow event.

4.3.3.2 Reactor Coolant Pump Shaft Break With Loss of Offsite Power

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a RCP. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow, while the reactor is at power, results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4.

This event bounds the pump rotor seizure because in the broken shaft case the pump presents higher resistance to flow and accelerates flow decay. Therefore, pump seizure will not be discussed. The sudden stopping of the RCP causes a CPC reactor trip to be followed by a turbine trip and LOP. By including a LOP and stuck open ADV, this event is classified as a limiting fault event for PVNGS Units 1 and 3. The concerns are peak RCS pressure and the MDNBR value. The acceptance criteria are that the maximum RCS pressure should be maintained below acceptable design limits, and the potential for core damage is evaluated on the basis that it is acceptable if the MDNBR remains above SAFDL limits. If the MDNBR falls below SAFDL values, fuel damage must be assessed. Some fuel damage is acceptable as long as the core remains in place and maintains a coolable geometry.

The analysis used the CENTS code and the HERMITE code to generate conditions at the time of the minimum DNBR. Then the TORC code was used to compute the actual value.

The results indicate that the peak pressure stays below the 110 percent design limit, however, the MDNBR falls below the SAFDL limit for a short period of time, thus, some cladding damage is expected. If any cladding damage occurs, however, it is limited to cladding perforation and activity release. The fuel rod retains its structural integrity and fuel coolability.

The NRC staff has reviewed the licensee’s analyses of the sudden decrease in core coolant flow events and concludes, as set forth above, that the licensee’s analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27 and 31 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the sudden decrease in core coolant flow events.

4.3.4 Reactivity and Power Distribution Anomalies

4.3.4.1 Uncontrolled Control Element Assembly Withdrawal From a Subcritical or Low Power Condition

An uncontrolled control element assembly withdrawal (CEAW) from subcritical or low power condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC’s acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1.

The licensee reanalyzed both the uncontrolled CEAW from subcritical and low power (hot zero power) events in support of the proposed PUR. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS Units 1 and 3 TS or documented in the PVNGS Units 1 and 3 UFSAR. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the current NRC-approved PVNGS Units 1 and 3 reload design methodology.

The licensee performed these analyses using the CENTS code (Reference 12) and the CETOP-D code (Reference 13). The NSSS response is simulated using CENTS, while the

transient DNBR values are calculated using CETOP-D. The licensee analyzed and provided results for both the existing PVNGS Units 1 and 3 operating conditions and the proposed PUR conditions. The acceptance criteria for these transients are that DNBR remain above the acceptance limit and that fuel temperature remains below the fuel melt temperature. For the CEAW from subcritical event, the licensee calculated a MDNBR value of 1.60. For the CEAW from low power event, the licensee calculated a MDNBR value of 1.45. These results demonstrate that the DNBR acceptance limit of 1.34 is satisfied. The fuel temperature acceptance criterion is commonly evaluated using a peak linear heat rate (PLHR) criterion. During the PUR review the NRC staff identified that the TS PLHR safety limit was being violated for these events. However, the licensee performed an adiabatic deposited energy calculation which demonstrated that the peak fuel temperature remains well below the limiting fuel centerline temperature for melting fuel. As a result of this review, the licensee submitted a license amendment request to change the TS safety limit from a PLHR to a fuel centerline melt temperature safety limit. This issue was resolved as part of Amendment No. 145, dated December 2, 2002 (Reference 14).

The NRC staff has reviewed the licensee's analyses of the uncontrolled CEA withdrawal from a subcritical or low power startup condition and concludes, as set forth above, that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

4.3.4.2 Uncontrolled Control Element Assembly Withdrawal at Power

An uncontrolled CEAW at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2.

The licensee reanalyzed the uncontrolled CEAW from power event in support of the proposed PUR. The licensee ensured that limiting initial conditions and input parameters were used in

the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS Units 1 and 3 TS or documented in the UFSAR. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with current NRC-approved PVNGS Units 1 and 3 reload design methodology.

The licensee performed the analysis using the CENTS code and the CPC Fortran code. The licensee analyzed and provided results for both the existing PVNGS Units 1 and 3 operating conditions and the proposed PUR conditions. The termination of this event relies upon a reactor trip signal from the CPC system to ensure that the SAFDLs are not exceeded. The acceptance criteria for this transient is that the minimum DNBR remain greater than or equal to the limit, and that the fuel temperature remain less than the melting temperature. The licensee's analysis demonstrated that the acceptance criteria are not violated during this transient.

The licensee's analysis assumed an initial core inlet temperature of 548 °F, which is the lower limit of the core inlet temperature range for PVNGS Units 1 and 3. As part of its review, the NRC staff questioned the licensee's use of an initial core inlet temperature at the lower limit of the range (rather than the upper limit), and the impact on DNBR results. The licensee stated that this analysis credits the CPC VOPT, and also credits the initial conditions of the event as being a known amount of thermal margin away from the DNBR SAFDL condition during normal operation. This known amount of thermal margin is maintained by PVNGS Units 1 and 3 TS 3.2.4. The initial parameter values used in the analysis are chosen such that the initial thermal margin is minimized. Thermal margin degradation during any given transient is calculated in terms of core power and is called the required over power margin (ROPM). The licensee states that the effect of a lower initial core inlet temperature is to delay the reactor trip, allowing the core power to increase further prior to the trip. The later reactor trip occurs due to the effect of temperature shadowing on the excore neutron detectors, which provide inputs for the CPC VOPT. The licensee stated that starting this transient from a higher initial core inlet temperature would result in an earlier CPC VOPT, which in turn would result in lower thermal degradation during the transient. The licensee demonstrated through ROPM analyses that the actual value of initial core inlet temperature has a rather insignificant effect on the reactor trip system as a result of the transient, and the ROPM is essentially unaffected by the selection of an initial core inlet temperature of 548 °F or 560 °F. The NRC staff has determined that the licensee's explanation is reasonable, and, therefore, is acceptable.

The NRC staff has reviewed the licensee's analyses of the uncontrolled CEAW at power event and concludes, as set forth above, that the licensee's analyses have adequately accounted for the changes in core design needed for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the uncontrolled control rod assembly withdrawal at power event.

4.3.4.3 Single Full-Length Control Element Assembly Drop

The NRC staff's review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations, (2) the sequence of events, (3) the analytical model used for analyses, (4) important inputs to the calculations, and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3.

As described in the UFSAR, a single full-length CEA drop results from an interruption in the electrical power to the CEDM holding coil of a single full-length CEA. The limiting case is the CEA drop, which does not cause a reactor trip to occur and results in an approach to the DNBR SAFDL.

The licensee reanalyzed the control rod misoperation event in support of the proposed PUR. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS Units 1 and 3 TSs or documented in the UFSAR. The licensee stated that the initial thermal margin preserved by the TS is a function of core power, and is the same between 95 percent and 100 percent of RTP. Below 95 percent RTP, additional amounts of thermal margin are preserved. The licensee determined that the limiting case for both the current and proposed power levels is the 95 percent RTP case. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the PVNGS Units 1 and 3 reload design methodology.

The licensee performed the reanalyses using the CENTS code and the CETOP-D code. The licensee analyzed and provided results for both the existing PVNGS Units 1 and 3 operating conditions and the proposed PUR conditions. The acceptance criteria for this transient are that DNBR remain above the acceptance limit, and fuel temperature remains below the fuel-melt temperature. The licensee provided quantitative results which demonstrated that the MDNBR value remains above the acceptance limit of 1.34 throughout the event. The licensee also provided quantitative results which demonstrate that the PLHR remains below the acceptance throughout the event.

The NRC staff has reviewed the licensee's analyses of the single full-length CEA drop event and concludes, as set forth above, that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the single full-length CEA drop event.

4.3.4.4 Startup of an Inactive Reactor Coolant Pump

A startup of an inactive RCP transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including AOOs, (3) GDC 20, which requires that the reactor protection system be designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs, (4) GDC 26, which requires that reliable control of reactivity changes are designed to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs, and (5) GDC 28,which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.4-5.

The licensee reevaluated the startup of an inactive RCP transient considering the PUR conditions. The licensee concluded that reanalysis is not necessary and that the event remains bounded by the UFSAR AOR. PVNGS Units 1 and 3 TS preclude operation with less than all RCPs in each loop operating during Mode 1 and 2 operation. Additionally, the analysis of record examines this event for operating Modes 3 through 6 only because plant operation with less than all RCPs running is only permitted in these modes. Therefore, the PUR will not impact the analysis of record for this event.

The NRC staff has reviewed the licensee's analyses of the startup of an inactive RCP event and, based on the foregoing, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet

the requirements of GDCs 10, 15, 20, 26, and 28 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the startup of an inactive RCP event.

4.3.4.5 Inadvertent Deboration

Unborated water can be added to the RCS via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. Operator action is needed to stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covered (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences, and (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to assure that specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.4.6.

The licensee did not reanalyze this event, but reassessed it with respect to the PUR. The PVNGS Units 1 and 3 current AOR demonstrates that Mode 5 (cold shutdown) with the RCS drained results in the least time available for detection and termination of an inadvertent deboration (ID) event. Because the limiting mode is Mode 5, the PUR will not affect the current AOR conclusions. The licensee's evaluation considered the larger RSGs, which increase the RCS volume. This increase in RCS volume results in an increased time for dilution (or a decreased dilution rate), thus, the analysis of record remains bounding. Additionally, the licensee verified that at the proposed PUR level, for Mode 1 and Mode 2 operation, the operator action time available remains bounded by the current Mode 5 AOR and that the PUR has no impact on the ID event. The NRC staff agrees that this event remains bounded by the current AOR.

The NRC staff has reviewed the licensee's analyses of the ID of the reactor coolant due to a chemical and volume control system malfunction and concludes, for the reasons set forth above, that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the decrease in boron concentration in the reactor coolant due to a chemical and volume control system malfunction.

4.3.4.6 Control Element Assembly Ejection

CEA ejection accidents cause a rapid positive reactivity insertion and an adverse core power distribution, which together could lead to localized fuel rod damage. The NRC staff evaluated the consequences of a CEA ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covered initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria contained in SRP Section 15.4.8 and used to evaluate this accident include:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Level C" as defined in the ASME Boiler and Pressure Vessel Code.

As described in the UFSAR, a CEA ejection event results from a circumferential rupture of a CEDM housing or of the CEDM nozzle. The CEA ejection may lead to a rapid positive reactivity addition resulting in a rapid power excursion and RCS pressurization.

The licensee reanalyzed the CEA ejection event in support of the proposed PUR. To ensure that the specific acceptance criteria are satisfied at the PUR level, the licensee analyzed two cases, a fuel performance case and a peak RCS pressure case. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS Units 1 and 3 TS or documented in the UFSAR. Only the high pressurizer pressure trip is credited in the analysis. Although a reactor trip on CPC or plant protection system (PPS) VOPT may occur much earlier after the initiation of the event, no credit is taken for this trip. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the PVNGS Units 1 and 3 reload design methodology.

The licensee analyzed the fuel performance case using the STRIKIN-II code (Reference 15) to simulate response of the fuel during the transient and to determine the energy deposition in the fuel. The CE-1 model in STRIKIN-II was utilized to calculate the minimum DNBR during the transient. The licensee analyzed the peak RCS pressure case using the CENTS code.

The licensee analyzed and provided results for both the existing PVNGS Units 1 and 3 operating conditions and the proposed PUR conditions. The licensee provided quantitative

results which demonstrate that all acceptance criteria are satisfied for both cases. For the fuel performance case, the radially averaged fuel specific enthalpy is less than the 280 cal/gm at the hottest axial location of the hot fuel pin and the fuel centerline enthalpy is less than 250 cal/gm. The licensee calculated the maximum radially averaged fuel enthalpy in the hot node to be 141 cal/gm. For the peak RCS pressure case the licensee calculated a peak RCS pressure of 2702 psia, which is below the Service Level C acceptance criterion limit.

The NRC staff has reviewed the licensee's analyses of the CEA ejection accident and, as set forth above, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage to significantly impair the capability to cool the core. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDC 28 for such events, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the CEA ejection accident.

4.3.5 Increase in Reactor Coolant Inventory

Two potential transients involve increase of reactor coolant inventory: inadvertent activation of the core cooling system and of the pressurizer level control system concurrent with LOP, and inadvertent malfunction of the CVCS concurrent with LOP. The licensee states that these transients are bounded by the AOR. The NRC staff concurs with this assessment and deems the AOR as bounding for increase of reactor coolant inventory transients.

4.3.6 Decrease in Reactor System Inventory

Inadvertent opening of a pressurizer safety valve is discussed in Section 4.1.

4.3.6.1 Double-Ended Break of a Letdown Line Outside Containment (Upstream of the Letdown Control Valve)

RCS fluid release outside containment can result from a break in a letdown line, an instrument line, or a sample line. The letdown line is the largest and a double-ended break would yield bounding results. The flow from the letdown line is small, and although this is a LOCA, it is not treated as a LOCA because none of the phenomena associated with a LOCA are present, therefore, the focus of this analysis is on radioactive releases.

The acceptance criteria are based on GDC 55, which requires isolation valves for lines connected to RCS lines that penetrate primary containment, unless it can be demonstrated that the isolation provisions for a class of lines, such as certain small instrumentation lines, are acceptable on some other defined basis. The allowable radioactive release from such lines is limited by the provisions of 10 CFR Part 100 exposure guidelines.

No single active failure of a containment isolation valve was considered because there are two isolation valves in Series. The analysis was based on the CENTS code, supplemented by CETOP-D for the calculation of DNBR. This being a heat-up transient, initial conditions were

chosen to maximize fuel degradation. Because of several alarms, operator action was credited at 10 minutes from transient initiation by isolation of the letdown line via closing isolation valves per EOPs.

The licensee states that the results indicate that the acceptance criteria for offsite doses are met. Radioactive releases and resulting doses are evaluated in Section 4.4 of this SE. Due to the fact that this scenario is not evaluated as a LOCA, only dose calculation and radioactive release will be evaluated for a double-ended break of a letdown line outside containment, and this evaluation is contained in Section 4.4.

4.3.6.2 Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and SG safety or atmospheric relief valves. Reactor protection and engineered safety features are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 100. The NRC staff's review covered (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events (with and without offsite power available), (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions consistent with the plant's emergency operating procedures (EOPs), and (7) the results of the accident analysis. A single failure of a mitigating system is assumed for this event. The NRC staff's review for SGTR discussed in this section is focused on the thermal and hydraulic analysis for the SGTR in order to (1) support the review related to 10 CFR Part 100 for radiological consequences, which is discussed elsewhere in this SE, and (2) confirm that SGs do not experience an overfill. Preventing a SG overfill is necessary in order to prevent failure of the main steamlines. Specific review criteria are contained in SRP Section 15.6.3.

An SGTR is analyzed in the UFSAR with and without concurrent LOP. The results with LOP are bounding, thus, only this case is analyzed here. LOP is assumed to occur due to grid instability about 3 seconds after reactor trip.

Upon tube rupture, RCS water mixes with shell side water and radioactivity is transferred to the condenser via the turbine. The condenser air removal pumps will transport this activity to the atmosphere. However, because there is a concurrent LOP, there will be a turbine trip, loss of normal FW flow, loss of RCS forced circulation, and loss of condenser vacuum. With the steam bypass control system unavailable, cool-down is accomplished with AFW and through the ADVs. This, however, is a direct route for radioactive emissions to the atmosphere.

To maximize emissions, a single failure is assumed with the ADV stuck fully open. Reactor initial conditions are also chosen to maximize steam release and, therefore, atmospheric emissions.

Radiation monitors, low pressurizer level, and high SG level let the operator diagnose SGTR and trip the plant manually before reaching the reactor trip point. This will keep the ADV open for a longer period of time and maximize emissions.

The analysis of an SGTR is based on the CENTS code. The analysis covers two events,

SGTR and a stuck-open ADV creating an excess steam demand. This transient is a limiting event. The plant EOPs provide operator instructions for plant recovery.

The results indicate that the behavior of the PUR plant configuration with the RSGs is similar to the existing plant configuration. EOPs are designed to preclude pressurization and challenge to the MSSVs, aid diagnosis and plant stabilization, accomplish functional recovery, provide post-tube-rupture tube coverage, maintain adequate RCS inventory, and accomplish shutdown and depressurization.

From a reactor protection point of view, the results are acceptable because the plant does not over pressurize nor does it sustain any fuel damage during the transient. As mentioned above, the radiation consequences of this event are discussed in Section 4.4 of this SE.

4.3.6.2.1 Steam Generator Tube Rupture With Concurrent Loss of Offsite Power (No Stuck-Open ADV)

As in the previous case, upon tube rupture, primary water will mix on the shell side and reach the atmosphere via the turbine, the condenser, and the condenser air removal pumps. However, SGTR will be followed by reactor and turbine trip and loss of normal FW flow, forced RCS flow, and condenser vacuum. Cool-down is maintained by using the AFW and releasing steam through the ADVs.

Radiation monitors will initiate alarms to notify the operator and aid in event diagnosis. The EOPs include explicit instructions to guide the operator to a reactor cool-down. The objectives of the EOP guidance and corresponding operator actions are: use the ADVs to control pressure and avoid challenge to the MSSVs, diagnose the event and stabilize the plant, cool-down the plant using the ADVs on both SGs before SG isolation, do a manual main steam isolation, isolate the affected SG, and cooldown and maintain adequate RCS inventory.

The results indicate that the plant will not over pressurize and the MDNBR will remain well above the SAFDL limits. (The licensee states that the atmospheric radioactivity release will be within the 10 CFR Part 100 limits.)

The NRC staff has reviewed the licensee’s analysis of the SGTR accident and concludes that the licensee’s analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the SGTR event.

4.3.7 Limiting Infrequent Events

4.3.7.1 AOOs in Combination With a Single Active Failure

The limiting infrequent event is designed to test the plant’s capability to respond to extreme transient conditions. The acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that

the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

The licensee created a composite limiting transient to bound the MDNBR of infrequent events (including AOOs in combination with a single failure). It is assumed that the unspecified event degrades the DNBR to the SAFDL level. The most limiting single failure is LOP, resulting in the coast-down of all RCPs. This is combined with the maximum linear heat rate produced by a CEAW. No single failures or operator errors can degrade DNBR more than the above circumstances; therefore, no other failures are assumed. Initial conditions conservatively assume a 116 percent power level due to a preexisting condition from the undefined AOO and a turbine trip coincident with reactor trip, although a 3 second delay exists. No operator action is assumed for 30 minutes after transient initiation.

The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

The NRC staff has reviewed the licensee’s analyses of the limiting hypothetical AOO transient with LOP. The NRC staff concludes that the licensee’s analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system will continue to ensure that the MDNBR and the peak RCS pressure will remain within the acceptance limits for this hypothetical event. In addition, core geometry and long-term cooling will remain within acceptable limits for such an event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 26, 27, and 31 for this hypothetical event. Therefore, the NRC staff finds the proposed bounding transient acceptable.

4.3.7.2 Anticipated Transient Without Scram (ATWS)

The pressure vessel has a certain depressurization capability and a maximum pressure design value. Certain vessels have been judged to have designed pressures marginally below the maximum pressure expected to develop in an ATWS. Pressure vessels in CE plants are in this category. The ATWS rule requires an independent and diverse shutdown system. With such a system, the probability of an ATWS event is thought to be acceptably low. Therefore, the NRC staff review for an ATWS event in a CE plant is to assure that a diverse and independent shutdown system is available.

According to 10 CFR 50.62, PWRs manufactured by CE must be equipped with systems diverse and independent from the reactor trip system to scram the reactor, trip the turbine, and initiate AFW under conditions of an ATWS. The licensee indicates that the existing system does not need any modification or resetting due to the PUR and installation of the RSGs. Inasmuch as the effects of the ATWS transient depend only on the vessel peak pressure design value, the vessel venting capability, and the presence of the diverse and independent shutdown system, the PUR and the RSGs, which do not affect these items, are irrelevant to the reactor's response to an ATWS.

4.3.7.3 Station Blackout

The requirements in Section 10 CFR 50.63 provide that each light-water cooled nuclear power plant must be able to withstand a loss of all AC power. Such a loss of power is a particular concern when offsite power is interrupted. In addition, the rule provides that each plant must cope with a loss of AC power for a certain time interval depending on plant specific factors. The rule also provides that the plant may have provisions for alternate AC power supply meeting specified criteria to satisfy the requirements of 10 CFR 50.63.

In response to the station blackout (SBO) rule, PVNGS Units 1 and 3 provided two gas turbine-generators as an alternate AC power source. Implementation adhered to the guidance in RG 1.155, "Station Blackout." The AC source would be available within 1 hour from the initiation of the transient and be able to provide adequate power for 3 hours for a total requirement of 4 hours. The PUR was examined with respect to potential increases in needed AC power. The result indicated that all of the systems needed for decay heat removal in the 4-hour post-SBO period will be adequately powered.

The NRC staff has reviewed the submitted information and concluded that the PVNGS Units 1 and 3 is equipped with one additional source of power in compliance with the SBO rule and adheres to the guidance in RG 1.155, which is adequate to remove decay heat. In addition, the NRC staff established that neither the PUR nor the RSGs affect the existing power source with respect to the SBO rule. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the plant's compliance with the SBO rule.

Section 6.11.4 of this SE discusses additional aspects of compliance with the SBO rule.

4.4 Radiological Accident Evaluation

This section of the safety evaluation addresses the impact of the proposed changes on

previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements which the NRC staff used in reviewing the requested amendment are the accident dose guidelines in 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19, “Control Room.” The NRC staff also used the review guidance of the accident-specific criteria in Section 15 and Section 6.4 of the SRP in its review. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in the following documents in performing this review:

- Safety Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors.”
- Safety Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.”
- RG 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors.”
- SRP Section 6.4, “Control Room Habitability Systems.”
- SRP Section 15.1.4, “Inadvertent Opening of a Steam Generator Relief or Safety Valve.”
- SRP Section 15.1.5, “Steam System Piping Failures Inside and Outside Containment (PWR),” Appendix A.
- SRP Section 15.2.8, “Feedwater System Pipe Breaks Inside and Outside Containment (PWR).”
- SRP Section 15.3.3, “Reactor Coolant Pump Rotor Seizure.”
- SRP Section 15.4.8, “Spectrum of Rod Ejection Accidents (PWR),” Appendix A.
- SRP Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment.”
- SRP Section 15.6.3, “Radiological Consequences of Steam Generator Tube Failure.”
- SRP Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,” Appendix A and Appendix B.
- SRP Section 15.7.3, “Postulated Radioactive Releases Due to Liquid-Containing Tank Failures.”
- SRP Section 15.7.4, “Radiological Consequences of Fuel Handling Accidents.”

4.4.1 Accident Dose Calculations

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were performed by APS in support of its proposed PUR. Information regarding these analyses was provided in Sections 6.4 and 9.9 of the licensee's PURLR. The NRC staff reviewed the assumptions, inputs, and methods used by APS to assess these impacts, and performed independent calculations to confirm the conservatism of the APS analyses.

APS performed an assessment of all significant non-LOCA and LOCA events currently analyzed in the PVNGS Units 1 and 3 UFSAR. These events and the corresponding SRP Sections that address them are listed below:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a LOP (SRP 15.1.4)
- MSLB Outside Containment with LOP (SRP 15.1.5)
- Feedwater Line Break (FWLB) Outside Containment with LOP (SRP 15.2.8)
- Single RCP Sheared Shaft with LOP (SRP 15.3.4)
- Control Element Assembly Ejection (SRP 15.4.8)
- Double-Ended Break of a Letdown Line Outside Containment (DBLLOCUS) (SRP 15.6.2.3.2)
- SGTR with LOP and Single Failure of ADV (SRP 15.6.3)
- SGTR with LOP (SRP 15.6.3)
- LBLOCA and SBLOCA (SRP 15.6.5)
- Radioactivity Release Due to Liquid Containing Tank Failure (SRP 15.7.3)
- Fuel Handling Accidents (FHA) (SRP 15.7.4)
- Limiting Infrequent Events

APS considered the impact of the proposed changes on each of the above listed DBAs. Where the impact could not be dispositioned qualitatively (e.g., demonstrating that previous analysis bounds the proposed plant configuration), APS performed reanalysis of the impacted DBAs. For these reanalyses, APS determined the 0-2 hour exclusion area boundary (EAB) whole body and thyroid doses and the 0-30 days low population zone (LPZ) whole body and thyroid doses. APS assessed the potential doses to control room personnel from the LBLOCA, CEA ejection, RCP sheared shaft, and SGTR with LOP and single failure of an ADV. APS had previously shown these to be the limiting events with regard to control room exposures. It should be noted that the flashing fractions used for the affected generator in calculating control room doses were based on predicted primary temperature profiles.

The NRC staff's independent evaluations considered the control room dose from all of the DBAs listed above. The accident-specific sections that follow briefly describe the accident, the APS evaluation of the impact of the proposed changes including reanalyses that were performed, and the NRC staff's evaluation. Table 1 attached to this SE tabulates the analysis assumptions used by the NRC staff in its confirmatory analyses.

The magnitude of the radiological consequences of a DBA is proportional to the quantity of fission products released to the environment. This release is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point. In general, the inventory of fission products in the fuel rods, the irradiation of materials outside of the reactor and the concentration of radioactive material in the RCS are directly proportional to the rated thermal power. Thus, an increase in the rated thermal power can be expected to increase the inventory of radioactive materials available for release. The previously analyzed transport mechanisms could be affected by plant modifications associated with the PUR and by installation of the RSGs, potentially resulting in a larger release rate. Table 2.1-1 of the PURLR in the licensee's December 21, 2001, PVNGS Unit 2 PUR submittal tabulated the significant reactor core and coolant system parameters impacted by the proposed changes. The RCS flow rate, the mass of RCS liquid, steam line pressure, and FW and steam mass flow rates are parameters for which changes could impact the transport of radioactive material. APS assessed the impact of these changes. The licensee states that this information is the dame for PVNGS Units 1 and 3.

4.4.2 IOSGADV with LOP

This DBA postulates an inadvertent opening of a SG ADV during power operations, resulting in the release of steam from the affected SG to the environment. The affected SG will rapidly depressurize and release its inventory of dissolved radionuclides through the fully open ADV to the environment. It is assumed that the operators isolate the failed ADV 30 minutes after it opens. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS specific activity and RCS primary-to-secondary leakage are at the maximum values allowed by TS. Although the thermodynamic transient analyses project no fuel damage for the current fuel cycle, APS assumed 5.5 percent fuel failure to bound future fuel cycles. APS made one change in the method of evaluation of this event. Previous calculations assumed a SG iodine decontamination factor (DF) of 10 for releases from the unaffected SG. Since the SG inventory, i.e., level, is maintained, a DF of 100 for the affected SG, as used for other DBAs, is appropriate. Since this is consistent with NRC staff practice, the NRC staff finds this change in method acceptable.

The NRC staff performed a confirmatory analysis. The assumptions used by the NRC staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the IOSGADV with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The NRC staff's independent calculations confirmed that the control room doses from this event would be acceptable.

4.4.3 MSLB Outside Containment with LOP

This DBA postulates an unisolable failure in one of the main steam lines at a location outside of containment, resulting in the release of steam from the affected steam line. Both SGs will rapidly depressurize through the MSLB releasing dissolved radionuclides through the faulted steam line to the environment. The MSIVs close to isolate the unaffected SG from the break. The faulted SG dries out in 30 minutes. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage and the initial SG specific activity are at the maximum values allowed by TS. APS conservatively assumes that the initial RCS specific activity is 3.6 μCi/gm dose equivalent I-131, which is greater than that allowed by TS. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the depressurization, resulting in the release of radioiodine from the fuel at a rate 500 times the normal appearance rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is at 60 μCi/gm dose equivalent I-131. A SG iodine DF of 100 is assumed in the unaffected SG.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the MSLB with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The staff's independent calculations confirmed that the control room doses from this event would be acceptable.

4.4.4 FWLB Outside Containment with LOP

This DBA postulates an unisolable failure in one of the main FW lines at a location outside of containment, resulting in the release of high energy water and steam from the affected FW line. Although there are two check valves on this line within the containment, APS conservatively assumes that both SGs will rapidly depressurize through the FWLB, releasing dissolved radionuclides to the environment, until the FW and MSIVs close to isolate the unaffected SG from the break. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TS. A SG iodine DF of 100 is assumed in the unaffected SG after 30 minutes.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the FWLB with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The NRC staff's independent calculations confirmed that the control

room doses from this event would be acceptable.

4.4.5 Single RCP Sheared Shaft with LOP

This DBA postulates that a RCP shaft has suddenly sheared causing a reduction in the flow through the affected RCS loop. A reactor trip occurs, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core leading to localized boiling and fuel damage. APS determines the limiting product of multiplying the failed fuel fraction and the radial peaking factor to meet acceptable EAB, LPZ, and control room dose limits. APS assumes that 10 percent of the core inventory of noble gases and iodine are located in the fuel rod gap. Following the reactor trip, an operator opens a single ADV and it is assumed to stick fully open, creating a pathway for the noble gases and iodine released from the fuel to be released to the environment. The affected SG empties and all RCS leakage into that SG is released directly to the environment. At 30 minutes after the RCP shaft shears, the operators initiate a controlled cool down using the unaffected SG. Since the main condenser is assumed to be unavailable, the plant is cooled down by releases of steam from the ADVs on the unaffected SG to the environment. Also at 30 minutes, the operators begin refilling the affected SG, covering the tubes at 90 minutes. The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TS. A SG iodine DF of 100 is assumed in the unaffected SG and, after 90 minutes, also in the affected SG. The assumption of operator actions in this event represents a change in analysis method over that used in previous calculations. The NRC staff finds that APS proposed actions and timings are reasonable in that they can all be accomplished within the control room and are addressed in plant procedures.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB, LPZ, and control room doses estimated by APS for the RCP sheared shaft with LOP event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

4.4.6 CEA Ejection

This DBA postulates the mechanical failure of a CEDM pressure housing that results in the ejection of a CEA. This failure breeches the reactor pressure vessel head and results in a LOCA to the containment. As a bounding value, APS assumes that 17 percent of the fuel rods in the core fail, instantaneously releasing all of the noble gases and iodine in the fuel rod gap. APS assumes that 10 percent of the gases are located in the fuel rod gap. A radial peaking factor of 1.77 is applied.

Two release cases are postulated:

- Release of fission products to the containment atmosphere, from where it will enter the environment via (1) the containment purge pathway, (2) containment design leakage, and (3) leakage of containment sump water outside of the containment.

- Release of fission products to the RCS with subsequent release to the environment via primary-to-secondary leakage through SGs.

APS assumes that a containment purge is in progress at the start of the event. The release is based on the release of noble gases and iodine from the RCS, assuming 1 percent failed fuel. For the containment leakage case, the containment is assumed to leak at the maximum rate allowed by TS for the first 24 hours and 50 percent of this rate for 30 days. Analysis credit was not taken for any iodine removal mechanisms. For the containment sump water leakage case, the analysis assumes that 50 percent of the iodine released from the core is dissolved in the sump water and is transported outside of the containment via the ECCSs. Leakage from these systems is assumed to occur at a rate of 3000 ml/hr. Of the iodine that leaks from these systems only 10 percent is assumed to become airborne and available for release. For the second release case, the APS analysis assumes that the RCS primary-to-secondary leakage is at the maximum value allowed by TS. A SG iodine DF of 100 is assumed in the SGs.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB, LPZ, and control room doses estimated by APS for the CEA ejection event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

4.4.7 DBLLOCUS

This DBA postulates a failure of a piping system outside of containment that provides a path for RCS dissolved noble gases and iodine to be released to the environment. The current PVNGS Units 1 and 3 UFSAR considered the double-ended break of a letdown line outside of containment. APS states that it reviewed all analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. APS states that the magnitude of the releases to the environment for this DBA is primarily a function of the leak rate and the assumed RCS specific activity. The NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary. The NRC staff bases this decision on the fact that the RCS specific activity assumed in the analysis is established by TSs, which are not affected by the PUR or RSGs. Since the operating RCS pressure range per TSs remain the same and unaffected by this amendment, the leak rate will not change significantly either.

4.4.8 SGTR with LOP and Single Failure of ADV

This DBA postulates a complete severance of a single tube in one of the SGs resulting in the transfer of RCS water to the ruptured SG. The primary-to-secondary flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. The SG pressure increases rapidly resulting in a release of contaminated steam to the environment via the MSSVs. It is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs. At that time, the ADV on the ruptured SG goes fully open. When this occurs, operators divert FW to the affected SG so as to maintain the SG tubes covered and thereby reduce radioactivity releases. The continuing steam releases via the fully open ADV cools down the plant. It is assumed that the releases continue for eight hours, at which time the RCS pressure reaches equilibrium with the SG pressure, stopping the release.

The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the event, resulting in the release of radioiodine from the fuel at 500 times the normal appearance

rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is 60 µCi/gm dose equivalent I-131. The assumptions used by the NRC staff in their confirmatory analyses are presented in Table 1. The APS analysis was performed using dynamic thermo-hydraulic inputs obtained from the proprietary CENTS computer code. Since the NRC staff does not have access to the CENTS code, the NRC staff's confirming calculation was based on a low resolution extrapolation of graphs of thermo-hydraulic data provided in the transient analysis section of the PURLR. The flow versus time data in Table 1 are the values extrapolated by the NRC staff from the graphs identified by APS in its letter of September 4, 2002. The EAB, LPZ, and control room doses estimated by APS for the SGTR with LOP and single failure event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

4.4.9 SGTR with LOP

This DBA postulates a complete severance of a single tube in one of the SGs resulting in the transfer of reactor coolant water to the ruptured SG. The primary-to-secondary flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. The SG pressure increases rapidly resulting in a release of contaminated steam to the environment via the MSSVs. It is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs. Since the main condenser is assumed to be unavailable, the plant is cooled down by releases of steam from the SGs to the environment. The steaming through the ADVs on both SGs continues until the ruptured SG is isolated at 90 minutes. After the ruptured SG is isolated, steaming of the unaffected SG continues. It is assumed that the releases continue for eight hours, at which time the RCS pressure reaches equilibrium with the SG pressure, stopping the primary-to-secondary leakage.

The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the event, resulting in the release of radioiodine from the fuel at a rate 500 times the normal appearance rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is 60 µCi/gm dose equivalent I-131.

The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The APS analysis was performed using dynamic thermo-hydraulic inputs obtained from the proprietary CENTS computer code. Since the NRC staff does not have access to the CENTS code, the NRC staff's confirming calculation was based on a low resolution extrapolation of graphs of thermo-hydraulic data provided in the transient analysis section of the PURLR. The flow versus time data in Table 1 are the values extrapolated by the NRC staff from the graphs identified by APS in its application dated July 9, 2004. APS determined that the predicted EAB and LPZ doses remain bounded by those estimated for the SGTR with LOP and Single Failure of ADV, and that the control room doses due to this were bounded by those estimated for other DBAs. Accordingly, the EAB and LPZ doses estimated by APS for the SGTR with LOP event were found to be acceptable. In addition, the NRC staff's independent calculations confirmed that the control room doses were within the acceptance criteria of the regulatory requirements and therefore acceptable.

4.4.10 LBLOCA and SBLOCA

A LOCA is a failure of the RCS that results in the loss of reactor coolant and, if not mitigated, fuel damage, possibly including a core melt, leading to substantial releases of fission products. The containment building holds up the majority of the radioactivity released from the core. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Fission products from the damaged fuel are released into the RCS and then into the containment. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. A LBLOCA is postulated as the failure of the largest pipe in the RCS. A SBLOCA involves a spectrum of smaller breaks. APS assessed several small break sizes and determined, that with regard to radiological consequences, the 0.02 ft^2 break size was more limiting. The APS analyses of these events were performed in a similar manner. The significant differences in the analyses are listed below:

- The timing of many events and actions in the accident scenario are a function of the break size and are derived from the transient analyses discussed in the PURLR.

- For a LBLOCA, APS assumes a core melt that releases a substantial portion of the core fission product inventory consistent with the guidance in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." APS assumes that 100 percent of the core inventory of noble gases and 25 percent of the core inventory of iodines are available for release from the containment. For the sump case, APS assumes that 50 percent of the core inventory of iodine is in the sump. The iodine specification is 91 percent elemental, 5 percent particulate, and 4 percent organic.

- For the SBLOCA, APS assumes that fuel damage occurs, releasing the fission products in the fuel rod gaps. No core melt is projected. APS assumes that 100 percent of the gap inventory of noble gases and 25 percent of the gap inventory of iodines are available for release from the containment. For the sump case, APS assumes that 50 percent of the gap inventory of iodine is in the sump. The iodine specification is 91 percent elemental, 5 percent particulate, and 4 percent organic.

For the LBLOCA and SBLOCA, APS considered several release pathways:

- Release from unisolated containment purge line

- Release from containment leakage

- Sump water leakage from ECCS systems outside of the containment

- (LBLOCA only) Containment atmosphere bypass via depressurized SGs

- (SBLOCA only) Primary-to Secondary leakage via SGs.

For the containment purge path, APS assumes that a containment purge is in progress at the

start of the event. The release is based on the release of noble gases and iodines from the RCS, assuming a pre-incident iodine spike and an RCS specific activity of 60 µCi/gm dose equivalent I-131.

For the containment leakage case, the containment is assumed to leak at the maximum rate allowed by TS for the first 24 hours and 50 percent of this rate for 30 days. APS credited iodine removal by containment sprays and by mechanistic plateout. The NRC staff believes that this assumption is inconsistent with the assumption that 25 percent of the core inventory of iodine is available for release. The NRC staff is of the opinion that this represents double crediting of plateout, since the 25 percent iodine available for release already reflects a reduction of 50 percent for plateout, per Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” This double crediting would reduce the level of conservatism of the analysis.

In response to a NRC staff request for justification, APS identified that the assumption had been submitted to the NRC staff in an amendment request dated June 25, 1991, and had been part of analyses performed in support of two other amendment requests. The NRC approved the requested amendments. Although the NRC staff was able to demonstrate acceptable doses without assuming mechanistic plateout in its independent analyses, it did not explicitly reject the licensee’s use of the same. APS stated its position that the mechanistic plateout assumption is part of its licensing basis. Although the licensee is voluntarily requesting a PUR, the application of mechanistic plateout is independent of power and is neither ameliorated nor acerbated by the power increase. Based on the above, the NRC staff accepts the plateout assumption for the PUR request as it is bounded by previous analysis.

For the containment atmosphere bypass via the depressurized secondary, APS had previously determined that the leakage would be equivalent to a containment leak rate of 0.9 standard cubic feet per minute (scfm). The containment source term and duration of release are as modeled for the containment leakage case above.

For the containment sump water leakage case, the analysis assumes that 50 percent of the iodine released is dissolved in the sump water and is transported outside of the containment via the ECCSs. Leakage from these systems is assumed to occur at a rate of 3000 ml/hr. Of the iodine that leaks from these systems only 10 percent is assumed to become airborne and available for release.

For the primary-to-secondary release pathway, the APS analysis assumes that the RCS primary-to-secondary leakage is at the maximum value allowed by the TS. All of the radionuclides released from the fuel gap are assumed to be dissolved in the RCS. A SG iodine DF of 100 is assumed in the SGs. The release duration is assumed to be three hours.

The assumptions used by the NRC staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ and control room doses were estimated by APS for the LBLOCA event. The EAB and LPZ doses were estimated by APS for the SBLOCA event. The control room doses for the SBLOCA were not evaluated, since it would be bounded by the LBLOCA doses. The NRC staff performed independent calculations of the EAB, LPZ, and control room doses for the release paths that contribute the majority of the dose. The NRC staff obtained results similar to those obtained by the licensee, and the NRC staff analyses confirmed the APS

conclusions.

4.4.11 Waste Gas System Failure

For this DBA event, APS evaluated the radiological consequences of an accidental release of the maximum inventory of a waste gas decay tank as addressed in the PVNGS Units 1 and 3 UFSAR. APS stated that it reviewed all of the analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. Since the maximum inventory of a waste gas decay tank is independent of reactor power and other parameter changes associated with the RSGs, the NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary.

4.4.12 Radioactivity Release Due to Liquid Containing Tank Failure

For this DBA event, APS evaluated the radiological consequences of an accidental release of maximum inventory of the refueling water tank as addressed in the PVNGS Units 1 and 3 UFSAR. APS stated that it reviewed all of the analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. The maximum inventory of a refueling water tank is independent of reactor power and other parameter changes associated with the RSGs. Accordingly, the NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary.

4.4.13 Fuel-Handling Accident

For this DBA event, APS evaluated the radiological consequences of the release of fission products due to damage of a fuel assembly during movement of irradiated fuel. APS stated that the current PVNGS Units 1 and 3 UFSAR analysis had been performed assuming a power level of 4070 MWt to support a license amendment request for relief from certain TSs associated with the movement of irradiated fuel. Those supporting analyses had been conservatively performed assuming a power level of 4070 MWt. The NRC approved Amendment 143 by letter dated July 25, 2002. The NRC staff concurs that the current analyses remain bounding and that further reanalysis is not necessary.

4.4.14 Limiting Infrequent Events

This DBA is a composite of infrequent events including AOOs with a single failure. The initial conditions and sequence of events were chosen to bound the infrequent events and AOOs with single failure. This event is a loss of reactor coolant flow that occurs in conjunction with a high transient reactor power level. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS assumes that the event causes 10 percent of the fuel rods to fail, releasing their gap activity to the RCS. A SG iodine DF of 100 is assumed in both SGs.

The NRC staff performed a confirmatory analysis. The assumptions used by the NRC staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by

APS for the AOO with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The NRC staff's independent calculations confirmed that the control room doses from this event are acceptable.

5.0 NUCLEAR FUEL

5.1 Core Thermal and Hydraulic Design

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on GDC 10, which requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs.

The licensee performed core thermal-hydraulic analyses in support of the proposed PUR. The licensee performed these analyses at the proposed uprated core power level of 3,990 MWt using NRC-approved methods, consistent with the licensee's current reload analysis methodologies. In its October 11, 2002 letter, the licensee verified that all thermal-hydraulic parameters remain within the code restrictions and limitations and, in accordance with the reload methodology, will continue to do so for each future operating cycle. Steady state DNB analyses were performed using the TORC code (Reference 16), CETOP-D code, and the CE-1 CHF correlation (References 17 and 18). The licensee determined that maintaining the current TS DNBR limit of 1.34 will provide assurance at the 95/95 probability/confidence level that the hot rod will not experience DNB under PUR conditions. This DNBR limit includes appropriate penalties associated with implementing the statistical combination of uncertainties (SCU) methodology (Reference 19).

The selection of the fuel rod gap conductance values can impact key transient parameter results such as pressure, temperature, and DNBR. The licensee provided results of sensitivity studies performed to evaluate the effects of gap conductance values. The sensitivity study results demonstrated that gap conductance has a more significant impact on fuel DNBR than on system pressure results. The licensee stated that the calculation, selection, and use of gap conductance values are consistent with the PVNGS Units 1 and 3 design basis methodology currently described in the UFSAR. The licensee applies NRC-approved methods and ensures that the gap conductance values used in the analyses are conservative for the safety analyses. Additionally, as discussed in Section 4.3 of this SE, the licensee reanalyzed or evaluated the UFSAR Chapter 15 transients at the PUR conditions to verify that the DNBR limit is satisfied during normal operation and AOOs.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed PUR on the thermal and hydraulic design of the core and the RCS. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the thermal and hydraulic design and demonstrated that the design has been accomplished using acceptable analytical methods, is equivalent to proven designs, provides

acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and is not susceptible to thermal-hydraulic instability. On this basis, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC 10, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to thermal and hydraulic design.

5.2 Core Design

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational occurrences, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burn-up, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (2) GDC 11, which requires that the core and associate coolant system be designed so as to assure that the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity, (3) GDC 12, which requires that the reactor core and associated coolant system be designed so as to preclude power oscillations which could result in conditions exceeding specified acceptable fuel design limits, (4) GDC 13, which requires that instrumentation be designed to monitor variables and systems affecting the fission process over their anticipated ranges for normal operation, for anticipated operational occurrences and for accident conditions, and that controls be provided for maintaining the variables and systems within prescribed operating ranges, (5) GDC 20, which requires that the reactor protection system be designed to initiate automatically operation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions, (6) GDC 25, which requires that the protection system be designed so that a single malfunction of the reactivity control system does not cause a violation of the specified acceptable fuel design limits, (7) GDC 26, which requires two independent reactivity control systems of different design, with one system using control rods and being capable of reliably controlling reactivity changes to assure that SAFDLs are not exceeded during normal operation including AOOs, and the second system having the capability to control the rate of reactivity changes resulting from planned, normal power changes, (8) GDC 27, which requires the reactivity control systems to be designed to have the capability in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, and (9) GDC 28, which requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding, and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3, "Nuclear Design."

In support of the PUR, the licensee provided information which demonstrates that the typical uprate core design will be similar to existing core designs for PVNGS Units 1 and 3. The licensee provided a comparison of several key core design parameters for a PUR core design.

The licensee utilizes NRC-approved core design methodology (Reference 16) for current core designs and will continue to apply this same methodology for future uprated power core designs. In accordance with this approved methodology, the licensee verifies that all key parameters remain within code restrictions and limitations for each reload cycle. The licensee will continue to perform cycle-specific core design analyses and to verify the applicability of the assumptions and parameters to future reload cycles, including the uprated unit reloads, in accordance with the approved reload process and methods. Additionally, as discussed in Section 4.3 of this SE, the licensee reanalyzed or evaluated the UFSAR Chapter 15 transients at the PUR conditions to verify that the acceptance criteria limits are satisfied during normal operation and AOOs.

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed PUR on the nuclear design. Because the licensee demonstrated that the typical uprate core design using currently approved methodology will be similar to existing core designs, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients. Similarly, the licensee has demonstrated that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the nuclear design.

5.3 Fuel System Design

The fuel system consists of fuel rods, spacer grids, guide thimbles, top and bottom end plates, and reactivity control rods, including burnable poison rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is needed, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, failure mechanisms, and safety of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling, (2) GDC 10, which requires that the reactor core and associated coolant and control systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (3) GDC 27, which requires that the reactivity control system be designed to have the capability, in conjunction with the ECCS, of controlling reactivity changes to assure that the capability to cool the core is maintained under accident conditions with appropriate margin for stuck rods, and (4) GDC 35, which requires that an ECCS be provided to transfer heat from the reactor core following any loss of reactor coolant above certain rates. Specific review criteria are contained in SRP Section 4.2.

5.3.1 Cladding Collapse

If axial gaps in the fuel column occur, the cladding has the potential to be flattened by the large

external coolant system pressure, a phenomenon called cladding collapse. A flattened cladding is considered a failed rod because of the large local strain. The fuel rod is designed to preclude cladding collapse.

The licensee analyzed the cladding collapse condition using the approved CEPAN code to calculate the collapse time as a function of residence time. The result showed that the rod collapse time is greater than the planned residence time for PUR conditions. This indicates that cladding collapse will not occur for the fuel rod design. Since the licensee used the approved methodology within its approved range of applicability with appropriate PUR typical inputs, the NRC staff concurs with the licensee's cladding collapse analysis for PVNGS Units 1 and 3 under PUR conditions.

5.3.2 Strain Fatigue

The fuel rod strain fatigue capability could be impacted by PUR conditions of higher operating temperatures and cyclic strains. The approved analysis of strain fatigue is based on the O'Donnell and Langer curve as described in the SRP Section 4.2.

The licensee reanalyzed the strain fatigue capability under PUR conditions using the O'Donnell and Langer curve. The result showed that the fuel system design maintained its strain fatigue capability. Since the licensee used the approved analysis within its approved range of applicability with appropriate PUR typical inputs, the NRC staff concludes that the strain fatigue capability is acceptable for PVNGS Units 1 and 3 under PUR conditions.

5.3.3 Clad Stress and Strain

SRP Section 4.2 states that the stress and strain limits in fuel designs should not be exceeded for normal operations and AOOs. During PUR conditions, the fuel system could experience high-power duty loading for certain AOOs, thereby exceeding the stress and strain limits.

The licensee reexamined the fuel system loading using the approved methodologies for calculating stress and strain limits. The results showed that the stress and strain limits were not exceeded for PUR conditions. Since the licensee used the approved methodologies within their approved ranges of applicability and with appropriate PUR typical values of input parameters, the NRC staff concludes that the fuel system design meets the stress and strain limits for PVNGS Units 1 and 3 under PUR conditions.

5.3.4 Rod Internal Pressure

Rod internal pressure is considered a driving force for fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. The NRC staff has approved topical report for Westinghouse (formerly CE) fuel designs CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," in which a rod pressure limit can exceed the system pressure provided that the fuel to cladding gap remains closed, i.e., there is no clad-liftoff.

The rod internal pressure will increase during PUR conditions. The licensee performed a bounding analysis using the approved fuel performance code FATES3B. The result showed that the maximum predicted rod pressure was below the critical pressure limit of no clad liftoff. Since the licensee used the approved methodologies within their approved ranges of

applicability and with appropriate PUR typical values of input parameters, the NRC staff considers that the rod internal pressure analysis is acceptable for PVNGS Units 1 and 3 under PUR conditions. However, because the methodology has been validated only up to a peak rod average burnup of 60,000 MWD/MTU, this conclusion is subject to that burnup limit.

5.3.5 Cladding Oxidation

SRP Section 4.2 identifies cladding oxidation buildup as a potential damage mechanism for fuel designs. The SRP further states that the effect of cladding oxidation needs to be addressed in safety and design analyses such as in the thermal and mechanical analyses. Recently the NRC staff realized that, in order to maintain adequate cladding ductility at high burnups, the total amount of oxidation or corrosion should be limited during normal operations, including AOOs. The licensee has adopted a corrosion limit of 100 microns for ZIRLO-clad fuel in its UFSAR.

The cladding corrosion will conceivably increase during PUR conditions. The licensee performed a bounding corrosion analysis that showed the maximum corrosion was within the 100 microns limit under PUR conditions. Based on the acceptable corrosion result, the NRC staff concludes that the impact of corrosion on the thermal and mechanical performance will be minimal for PVNGS Units 1 and 3 under PUR conditions.

5.3.6 Fuel System Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed PUR on the fuel system design. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. On this basis, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC 10, GDC 27, and GDC 35 following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the fuel system design for peak rod average burnup of up to 60,000 MWD/MTU.

5.4 Neutron Fluence

The NRC staff reviewed the licensee's fluence calculation methodology, including input parameters and approximations used. Fluence values determine reactor pressure boundary material properties which are then used to calculate pressure-temperature limits for cold overpressure protection and pressurized thermal shock. The acceptance criteria are based on GDCs 14, 30 and 31. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," also describes acceptable methods for vessel fluence calculations; however, the AOR for PVNGS Units 1 and 3 dates back to its initial safety analysis, before the issuance of RG 1.190, and is the basis for the design of the reactor pressure vessel.

The licensee performed fluence calculations in the existing analysis of record at the 4200 MWt power level. In addition, the out-in type of fuel loading was assumed, however, the proposed PUR is for 3990 MWt and the loading pattern has been for low leakage since cycle 2 for PVNGS Units 1 and 3. Both conditions are conservative and the AOR bounds the values calculated for the PUR. PVNGS Units 1 and 3 has a calculated vessel fluence value which has been reviewed and approved with the approval of the 32 EFPY pressure temperature curves of amendment 117 dated May 20, 1998. The NRC staff concludes that the fluence values of record are conservative and therefore, are acceptable.

6.0 BALANCE-OF-PLANT

6.1 Balance of Plant Components

The licensee described the BOP systems and components evaluation in Section 8 of the PURLR. The licensee evaluated the safety-related BOP piping, components, and supports affected by the RSGs and PUR operation in accordance with the provisions of the ASME Code, 1974 Edition, with addenda through the Winter 1975 Addenda, which is the code of record. The evaluation considered the normal, upset, emergency, and faulted conditions. The licensee provided the maximum stresses in the affected BOP piping. The stresses are below the Code allowables.

The BOP components (e.g., heat exchangers, pumps, and valves) that are most affected by the RSGs and PUR are those of the main steam system. The affected components include the MSSVs, MSIVs, turbine stop valves, turbine throttle valves, low pressure FW heaters, high pressure FW heaters, condensate pumps, FW pumps, and heater drain pumps. The RSGs and PUR conditions do not raise the main steam operating pressure above the original design pressure. The licensee found that the original component design analyses bound the predicted flow rates, temperatures, and pressures for the RSGs and PUR conditions.

The PUR results in an increase in main steam flow and FW flow. In relation to the potential for FIV in the affected BOP components, the licensee stated that all secondary side component steam and water velocities are predicted to remain below the original component design values when operating at PUR conditions. The licensee also stated that the post-PUR startup testing will be conducted in accordance with ASME OM-S/G-2000, “Standards and Guides for Operation and Maintenance of Nuclear Power Plants,” including Part 3, “Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems.”

The NRC staff noted that the acceptability of several secondary system components (i.e., steam traps) relies on an improvement in the steam quality (from 0.25 percent to 0.1 percent) to offset the increase in steam flow under the PUR conditions. The licensee stated that the improved steam quality is a performance criterion for the RSGs, and the steam quality will be measured during the post-PUR startup test program. The licensee also stated that the measured steam quality for the OSGs is currently higher than 0.25 percent (which is the design value for the OSGs), such that the anticipated 0.1 percent for the RSGs will result in a more than four-fold reduction in total moisture carryover. Since the secondary system capacities are not challenged at the current operating conditions, the licensee concludes that they will be adequate for the RSGs and PUR conditions.

The licensee's programs for valves are discussed in Section 9.5 of the PURLR. The licensee reviewed the design basis of the safety-related power-operated valves, including motor, air, and solenoid operated valves considering GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The design parameters included pressure, temperature, and differential pressure. The licensee concludes that:

(1) There are no changes to the design basis pressure or temperature of any safety-related power-operated valves.

(2) The existing valve actuators will adequately perform their intended design function after PUR, with the exception of the main steam isolation valve bypass valves (MSIVBVs), which require a modification to operate with increased differential pressure.

The licensee stated that the MSIVBVs have a design function to close on a main steam isolation signal and are normally closed during full power operation. The actuators are being modified such that the valves are capable of closing under maximum, worst-case differential pressure loads. The modifications consist of increasing the pre-load of the actuator stanchion springs, increasing the pressure of the air that opens the valves (by removing the regulator), and increasing the size of the actuator air exhaust piping and solenoid to improve the MSIVBV stroke time.

The NRC staff noted that the PUR and installation of RSGs results in an increase in the post-accident containment temperature and pressure, and requested additional information related to the overpressurization of isolated piping, referencing GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The licensee stated that the slight increase in temperature has a minimal effect on equipment in containment, and the existing analyses performed to address GL 96-06 remain bounding for the RSGs and PUR conditions. The licensee concluded that the systems and components within containment subject to post-accident environmental heating and pressurization remain capable of withstanding the predicted peak pressures and temperatures.

The MSIVBVs are discussed further in Section 6.7.

6.2 Auxiliary Feedwater System

The AFW system is designed to supply an independent source of water to the SGs during plant startup, hot standby, normal shutdown, and in the event of loss of the main FW supply. The system consists of two redundant, safety-related essential trains and one nonessential train, all of which supply both SGs. AFWS supply is provided by the condensate storage tank (CST) with a backup supply available from the reactor makeup water tank.

The essential AFW trains, which are not routinely used during startup and shutdown, can function automatically as required in the event of a LOP, SBO or an accident. Under these conditions, the decay heat is removed from the SGs to the atmosphere via the ADVs. The major components of the essential AFW trains are two redundant 100 percent capacity safety grade pumps which are powered from redundant and diverse sources (one steam turbine driven pump and one motor driven pump). These two essential AFW trains are cross-connected through redundant normally closed motor-operated valves in series such that either

essential AFW pump can supply FW to either or both SGs in the event of a single active failure of the AFW components. A third 100 percent capacity nonessential/ nonsafety grade motor driven pump is provided for use during startup, hot standby, and normal shutdown conditions. This pump may also be manually loaded to an essential emergency bus if necessary.

The licensee re-analyzed the minimum flow requirements of the AFW and the minimum CST inventory dictated by accident analyses of the limiting transients. Results of the re-analyses demonstrate that the AFW performance remains acceptable for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

The CST, which has a capacity of 550,000 gallons, provides a reserve capacity of 300,000 gallons to support the AFW for secondary side cool-down. The increased decay heat resulting from plant operations at the proposed 2.94 percent PUR will cause a slight increase in the condensate inventory needed for the secondary side cool-down. However, results of the re-analyses also confirm that the existing CST inventory bounds the condensate inventory necessary for PVNGS Units 1 and 3 operations at the 2.94 percent PUR.

Based on our review of the licensee's rationale and evaluation, as set forth above, the NRC staff finds that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the AFW and CST. Also, based on the experience gained from the NRC staff review of PUR applications for similar PWR plants and the results of the licensee's reanalyses that show the AFWS and CST performance remaining acceptable for PVNGS Units 1 and 3 operation at the PUR, the NRC staff concludes that the AFW and the CST are acceptable for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

6.3 Condensate and Feedwater System

The condensate and feedwater system (CFWS) provides FW from the condenser to the SGs. The CFWS includes all components and equipment from the condenser outlet through the containment isolation valves to the SGs and to the heater drain system. The system, with the exception of the portions from the main steam support structure walls for both the downcomer and economizer lines up to the SGs, serves no safety function and is, therefore, designed as nonsafety-related, Quality Group C, and non-seismic Category I equipment. The portions of the CFWS from the main steam support structure walls for both the downcomer and economizer lines (including containment isolation valves) up to the SGs, are safety related and designed to Quality Group B, seismic Category I requirements in order to assure FW system isolation in accident situations. Adequate isolation is provided at connections between seismic Category I and nonseismic systems, therefore, failure of the nonsafety-related portions of the CFWS will not affect plant safe shutdown.

The licensee evaluated the CFWS for plant operations at the proposed 2.94 percent PUR. The licensee stated that the condensate and FW pump flow rates will increase, resulting from plant operations at the proposed PUR. Condensate flow rates will increase by approximately 1 percent while FW flow rates will increase by approximately 6 percent. However, the CFWS will perform its intended function for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

Since the CFWS does not perform any safety related function with the exception of containment and FW system isolation in accident situations, and since plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the CFWS, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of the CFWS, except with respect to FW isolation.

The FWIVs, which also serve as containment isolation valves, are designed to isolate the FW system from the SGs during a MSLB, FWLB, or LOCA. Operation of the FWIVs may cause potentially large dynamic pressure changes and must be considered in the design of the valves and associated piping. The worst case loads occur following a MSLB from no load conditions with both FW pumps in service providing maximum flow following the break. The FWIVs are designed to close against a pressure differential of 1875 psi. The licensee stated that the FW pumps' maximum discharge pressure (deadhead pressure) is 1636 psia, therefore, the existing FWIV design bounds any potential fluid dynamics change associated with plant operations at the proposed 2.94 percent PUR. In addition, the FWIV closure time is not affected by PUR since the resulting PUR differential pressure is bounded by the original system design.

The NRC staff has reviewed the licensee's rationale and evaluation, and determined that the existing FWIV design bounds operation at PUR conditions. On this basis, and in light of the experience gained from the NRC staff review of PUR applications for similar PWR plants, the NRC staff concurs with the licensee that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR will have no impact on the FWIVs.

6.4 Circulating Water System

The circulating water system (CWS) is designed to remove the heat from the condenser to the atmosphere via the mechanical draft cooling tower, thereby maintaining adequately low condenser pressure. The CWS is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents. The licensee stated that the performance of this system was evaluated for PUR conditions and determined that the system is adequate for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

Since the CWS does not perform any safety function, and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of the CWS.

6.5 Main Turbine

The turbine-generator, which converts steam power into electrical power has a turbine control and overspeed protection system. Overspeed protection is accomplished by three independent systems (normal speed governor, mechanical overspeed, and electrical backup overspeed control systems). These overspeed protection systems are designed to satisfy the requirements of GDC 4 with respect to SSCs important to safety being appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure.

The licensee assessed (with the help of the turbine manufacturer) turbine operations with respect to the design acceptance criteria to verify the mechanical integrity of the turbine under the conditions of plant operations at the proposed 2.94 percent PUR. The assessment included a structural evaluation of the turbine components as well as system performance. The existing minimum overspeed setting is 110 percent and maximum overspeed setting is 111 percent. Since plant operations at the proposed power level do not raise the main steam operating pressure above the original design pressure, the turbine manufacturer concluded that the existing overspeed settings remain acceptable. Also, results of the evaluations showed that there would be no increase in the probability of turbine overspeed. Therefore, the turbine can continue to be operated safely at the proposed 2.94 percent PUR.

Because mainsteam pressure does not increase with the PUR, the NRC staff finds that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR do not change the design aspects and operation of the turbine-generator overspeed protection system. Therefore, the turbine-generator overspeed protection system continues to meet the requirements of the GDC 4. For this reason, and based on the experience gained from the NRC staff review of PUR applications for similar PWR plants, the NRC staff agrees with the licensee that operation of the turbine at the proposed 2.94 percent PUR is acceptable.

6.6 Main Turbine Auxiliaries

The turbine auxiliary system, together with the CWS and main condensers, are designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. The licensee assessed the performance of these systems for the PUR, and determined that these systems are adequate for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

Since these systems do not perform any safety function and their failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of these systems.

6.7 Main Steam Supply System

The function of the main steam supply system (MSSS) is to deliver steam from the SGs to the high-pressure turbine over a range of flows and pressures covering the entire operating range from system warmup to maximum operating conditions. The system also provides steam to the moisture separator/reheaters, the FW pump turbines, the auxiliary steam system, and the turbine gland seal.

Steam produced in the two SGs is routed by four main steam lines up to the common header at the high pressure turbine. The portions of the MSSS from the SGs up to and including MSIVs, MSIVBVs, MSSVs, and ADVs, are located in the flood and tornado protected main steam support structures, and designed as safety-related, Quality Group B and seismic Category I equipment to satisfy the requirements of GDC 2 and GDC 4.

The safety functions of the MSSS are: provision of steam for safety-related auxiliaries and engineered safety feature pumps; provision of a heat sink during certain transients and

accidents; limiting RCS pressure during certain transients; SG and MSSS overpressure protection; and termination of MSLB events. Also, the MSSS allows cool-down of the SG via the ADVs when the condenser is not available, as in during an SBO event. The licensee has stated that its evaluation shows systems associated with the MSSS will operate within the same manner as in the current configuration. The PUR will result in an increase in steam/feedwater flow rate; however, these increase values are within the original design specifications for the MSSS. The licensee stated that the increased mass flow rates have been evaluate and found acceptable.

Because the PUR does not create conditions beyond the original design specifications for the MSSS systems, the NRC staff finds that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR does not change the design aspects and operations of the MSSS. Therefore, the MSSS satisfies the requirements of GDC 2 and GDC 4.

The main steam lines from the MSIVs to the turbine stop valves and all branch lines are nonsafety-related, and are not required to support safe shutdown of the reactor. These portions of the MSSS are designed to the requirements of American National Standards Institute (ANSI) B31.1, “Power Piping.” Since these portions of the MSSS do not perform any safety-related function, and their failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on their design and performance.

The licensee performed an evaluation of the effects resulting from plant operations at the proposed 2.94 percent PUR on the MSSS including the MSIVs, MSIVBVs, MSSVs and ADVs. The licensee stated that the steam flow resulting from plant operations at the proposed 2.94 percent PUR will be 17.9×10^6 lbm/hr which is approximately 5.9 percent above the steam flow of 16.9×10^6 lbm/hr at the current full power operation. The main steam design conditions of 1270 psig and 575EF remain unchanged and bound all predicted operating conditions for the system and components. The licensee concluded that with the exception of MSIVBVs, plant operations at the proposed 2.94 percent PUR will have an insignificant or no impact on the MSSS and its associated components.

The MSIVBVs allow small amounts of steam to bypass the MSIVs during plant startups. When plant startup conditions require, using the MSIVBVs allows the piping downstream of the MSIVs to heat-up before opening the MSIVs. The MSIVBVs are automatically closed upon receipt of a main steam isolation signal (MSIS). As a result of plant operations at the proposed 2.94 percent PUR, the MSIS setpoint will be raised from >890 psia to >955 psia. The licensee stated that the MSIVBVs will be modified before implementation of this license amendment so that the valves are capable of closing against the increased differential pressure.

Based on our review of the licensee’s rationale and evaluation, and the experience gained from our review of PUR applications for similar PWR plants, and because the PUR only affects the MSIVBVs in the MSSS, and as they will be modified to close against the increased differential pressure, the NRC staff concurs with the licensee that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR will have an insignificant or no impact on the MSSS.

6.8 Miscellaneous Cooling Water Systems

6.8.1 Plant Cooling Water System

The plant cooling water system (PCWS) is designed to remove heat from the nonsafety-related, normally operating, closed cooling water systems over the full range of normal plant operation. The PCWS uses a portion of the CWS flow from the plant cooling towers to remove heat from the nuclear cooling water system (NCWS), the turbine cooling water system (TCWS), and the condenser vacuum pump seal coolers. Cooled circulating water returned from the cooling towers is pumped in parallel through the TCWS heat exchangers, NCWS heat exchangers and the condenser vacuum pump seal coolers, and is discharged back to the CWS. The licensee performed an assessment and stated that the PCWS heat loads considered in the original design remain bounding under PUR conditions.

Since the PCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of these systems.

6.8.2 Turbine Cooling Water System

The TCWS provides cooling for the nonnuclear-related components in the various turbine plant auxiliary systems. The licensee assessed the effects of the increase in heat loads on the TCWS and stated that the design heat loads for the TCWS components remain bounding under PUR conditions.

Since the TCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of this system.

6.8.3 Nuclear Cooling Water System

The NCWS is designed to supply cooling water to various nonsafety-related components and heat exchangers during plant normal operation including start-up, normal shutdown and hot standby. The NCWS acts as an intermediate barrier between systems that contain or may contain radioactive or potentially radioactive fluids and systems that should not contain such fluids. It serves no safety-related function. The licensee performed evaluations of the effects of the increase in heat loads on the NCWS and stated that the NCWS has the capacity to accommodate the additional heat loads.

Since the NCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the 2.94 percent PUR on the design and performance of this system.

6.8.4 Essential Cooling Water System

The essential cooling water system (ECWS) is a closed loop system which serves as an intermediate barrier between the essential spray pond system and systems which contain

radioactive or potentially radioactive fluids in order to eliminate the possibility of an uncontrolled release of radioactivity. It provides cooling water to various safety and non-safety systems during all phases of normal plant operation, including startup through cold shutdown and refueling, as well as following an SBO event, LOCA, MSLB or FWLB. The ECWS heat loads resulting from plant operations at the proposed 2.94 percent PUR will increase slightly. The licensee performed assessments to determine the effects of the increases in heat loads on the ECWS.

Based on our review and the experience gained from our review of PUR applications for similar PWR plants, and because the additional heat loads only results in minor temperature increases in the ECWS for normal and accident scenarios, and the design heat loads for the ECWS components still remain bounding under the PUR conditions, the NRC staff finds that plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the ECWS and have an insignificant or no impact on the ECWS. Therefore, the NRC staff concludes that the ECWS is acceptable for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

6.8.5 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool (SFP) cooling and cleanup system (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the pool, and to clarify and purify the water in the pool. The SFPCCS consists of two independent full-capacity essential SFP cooling trains each with one pump and one heat exchanger, and two separate non-essential cleanup and purification trains each with a fuel pool cleanup pump, filter, strainer, and ion exchanger. Electrical power to each of the SFP cooling pumps is supplied from Class IE emergency power buses. Heat is removed from the SFP heat exchangers by the NCWS. Supplemental SFP cooling can be provided by interconnections to both loops of the shutdown cooling system.

The SFP cooling trains of the SFPCCS are housed in the seismic Category I flood-and-tornado-protected fuel building. The SFPCCS itself, with the exception of the cleanup and purification trains, is designed to Quality Group C, seismic Category I requirements. The SFP cleanup and purification trains can be isolated by manual valves from the SFPCCS. Failure of these nonseismic, Quality Group D cleanup components during a seismic event will not affect operation of the cooling trains.

The SFP cooling system is designed to maintain the SFP water temperature at or below 125EF during normal operation. As stated above, supplemental fuel pool cooling is provided by the shutdown cooling system to maintain the SFP water temperature at or below 145EF during planned full core offload outages or in the event that an unplanned full core offload is performed.

As a result of plant operations at the proposed 2.94 percent PUR, the decay heat load for any specific fuel discharge scenario will increase slightly. The licensee stated that the maximum allowable SFP decay heat load is administratively controlled so that the heat load in the SFP is less than the available SFP cooling train heat removal capability, considering the worst single failure.

The licensee stated that the SFP thermal analyses do not determine a peak calculated SFP

temperature; instead, the thermal analyses utilize the peak allowable SFP temperature as an end point to calculate the maximum allowable heat load. The predicted actual heat load is compared to the maximum allowable heat load to verify that adequate heat removal capability remains. This approach ensures that the SFP cooling train heat removal capacity is greater than the predicted heat load and that the peak SFP temperature will remain lower than the maximum allowable temperature.

The licensee performed an assessment which indicates that the maximum heat load for a full core off load increases from 45.8 x 10^6 Btu/hr to 47.0 x 10^6 Btu/hr. The lowest heat removal capability (with one SFP cooling train cooled by ECWS and augmented by one shutdown cooling train) during a full core offload is 49.0 x 10^6 Btu/hr.

Also, the SFP has a water temperature monitor system which alarms in the control room when the SFP water temperature reaches 125EF. In the event that the alarm goes off due to high SFP temperature, plant procedures provide direction for the operator to take corrective actions (i.e., place the standby SFP cooling train in operation, place the shutdown cooling system in the SFP cooling assist mode, stop fuel movement, etc.). This will provide additional measures to prevent the SFP water temperature limits from being exceeded.

Based on the review of the licensee's rationale, experience gained from our review of other similar PUR applications, the capacity of SFP and shutdown cooling trains, and the fact that the plant has administrative controls and operating procedures in place to ensure that backup cooling capability is provided for all SFP cooling scenarios, the NRC staff finds that the design and operation of the SFP cooling system for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR is acceptable.

Since the SFP cleanup and purification system does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of plant operations at 2.94 percent PUR on the design and performance of this system.

6.9 Miscellaneous Mechanical Reviews

6.9.1 Heating, Ventilation, and Air Conditioning Systems

Heating, ventilation, and air conditioning (HVAC) systems are provided for PVNGS Units 1 and 3 for safety protection, personnel comfort, and equipment protection functions. The HVAC systems are described in UFSAR Sections 6.4 and 9.4.

To demonstrate that, based on a review of documented design basis calculations, the total heat load increases are within the design margin at PUR conditions, the licensee stated:

> The impact of PUR on the various HVAC systems is documented in an engineering study prepared in support of the PUR licensing submittal. The auxiliary building ventilation system (ABVS) and control building heating, ventilation, and air conditioning system (CBHVACS) are the only HVAC systems credited in safety analysis that are impacted by operation at PUR conditions. The worst-case event for heat loads on ABVS remains LOCA and MSLB. The

worst-case event for heat loads on CBHVACS remains any event resulting in a LOP. The containment heating, ventilation, and air conditioning systems (CHVACS) and turbine building heating, ventilation, and air conditioning system (TBHVACS) systems are not credited in any safety analysis. New individual design calculations were prepared and/or revised as required to evaluate the heat load impact of the RSGs and PUR on the normal operation of the plant HVAC systems.

6.9.1.1 Containment Heating, Ventilation, and Air Conditioning System

The CHVACS include those systems that function during normal plant operation, the containment pre-access period, or during extended shutdown. These systems are not required to operate during any DBA. The HVAC equipment and ducting within the containment and the main steam support structure are designed to retain structural integrity, but is not required to function during and after a SSE. Those portions of CHVACS that penetrate the containment boundary are designed as seismic Category I insofar as they are required to function to maintain containment isolation capability.

The CHVACS normal mode of operation is designed to maintain the temperature and reduce the humidity below 90 percent. The normal cleanup system, together with the normal purge system is designed to control the airborne radioactivity below the level needed for personnel access for inspection, maintenance, and refueling operations. The refueling purge is designed to maintain the airborne radioactivity at a level that permits sustained personnel occupancy during refueling. The containment is maintained at a negative pressure relative to the atmosphere during the purge cycle.

In the event that the concentration of either airborne particulate or iodine activity in the containment is higher than desired levels, air cleaning is accomplished by activating the recirculating filtration unit. This unit is equipped with charcoal and high efficiency particulate air (HEPA) filters to reduce containment airborne radioactivity to acceptable levels. The main steam support structure normal ventilation system provides once through ventilation with 100 percent filtered outside air for the structure.

The licensee states that the design basis for the containment heat load calculations considered a RCS with hot leg reactor coolant temperature (T_{hot}) of 621EF. This T_{hot} of 621EF bounds the PUR T_{hot}, and therefore, the total heat load resulting from PUR will be less than the original design.

In regard to how, based on a review of design basis calculations, the total heat load increases were determined to be within the design margin at PUR conditions, the licensee stated the following:

- The CHVACS is not credited in any safety analysis.

C The design calculation was performed to compare the normal heat load contribution of the RSGs to that of the OSGs. This calculation considered the increased RSGs' surface area and the improved reflective metal insulation system. Based on the results of this calculation, the RSGs's contribution to the total containment heat load is

predicted to be less than that contributed by the OSGs.

- The heat load from insulated piping and components in the containment building is analyzed in an existing design basis calculation for the bulk containment volume. The piping and equipment heat loads considered in this calculation are based on the original plant RCS T_{hot} of 621°F, which bounds the PUR predicted T_{hot} (618.9°F with 10 percent tubes plugged) temperatures. Equipment motor heat is based on the motor rated brake horsepower, and remains bounding for operation at PUR conditions. As a result, the heat loads considered in the existing CHVACS design basis calculation for containment remain conservative with respect to the predicted PUR heat loads.

- The RCS T_{hot} PUR conditions do not affect the nonsafety-related reactor cavity cooling subsystem and the CEDM cooling units as its original design is based on the original licensed T_{hot} of 621°F. In summary, the RSGs and PUR conditions are bounded by the existing CHVACS design.

Since the CHVACS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR on the design and performance of the CHVACS.

6.9.1.2 Auxiliary Building Ventilation System

The normal ABVS is a once-through air system and serves the equipment rooms, access control areas, the mechanical and electrical penetration areas, areas below the 100-ft 0-inch elevation of the main steam support structure, and the remainder of the auxiliary building to maintain an environmental condition suitable for personnel access and limit potential radioactivity release to the atmosphere during normal operation. The ductwork at all levels of the auxiliary building below level 140-ft 0-inch is designed to retain structural integrity, but is not required to function during and after a SSE. During normal plant operation, treated outside air is distributed through the building on a once-through basis.

The Engineered Safety Features (ESF) equipment rooms and the safety-related AFW pump rooms are served by an essential ABVS during emergency operation. The essential system provides individual ESF equipment room cooling and filtered exhaust to the atmosphere during emergency operation. The essential ABVS includes those systems that function post-LOCA within the ESF pump room, and the exhaust system, which maintain the auxiliary building below elevation 100-ft 0-inch at a negative pressure post-LOCA to prevent unfiltered release of possible airborne radioactivity to the surroundings. The ESF pump room and safety-related AFW pump room coolers are designed to maintain the required room temperatures to ensure the operability of the ESF pumps and motors during accident conditions. The ESF equipment room essential air coolers consist of a recirculating air handling unit, including a cooling coil, in each pump room. The essential exhaust filtration system consists of two essential exhaust filtration units shared with the fuel building, and a connecting tunnel and plenum. Following a LOCA, the ESF equipment and safety-related auxiliary FW pump rooms are automatically isolated (at approximately the 100-ft elevation) from the auxiliary building normal HVAC system on receipt of a safety injection actuation signal (SIAS). The building pressure is reduced to a measurable negative pressure relative to below ambient by the fuel building essential exhaust

fans for the space below elevation 100-ft 0-inch in the auxiliary building. Air exhausted from the ESF equipment rooms is filtered by the fuel building essential filter units. The essential exhaust filter units are automatically actuated by starting the fans and opening the dampers to the units in response to an ESF pump start signal.

The licensee stated the following:

C The increased reactor power level will affect the ABVS. The ABVS system piping design temperatures, pump motor maximum operating horsepower, electrical equipment, lighting heat loads are, with one exception, not affected by the PUR. The increased reactor power does result in an increased post-accident (LOCA and MSLB) containment temperature as discussed in Section 6.2 of the PUR submittal. This affects the transmission of heat loads through the containment wall into the adjacent rooms. However, this increase in heat loads remains bounded by the original ABVS design.

C The heat loads or other input parameters considered in the original SBO design remains bounding for PUR. The essential equipment rooms' original design environmental condition remains bounding for PUR. Heat transfer due to fluid transport through ECCS piping was evaluated. The ABVS heat loads have been revised to account for the new heat loads predicted to occur post-LOCA and MSLB. The heat loads remain within the individual room cooling coil capacities, and within the total ABVS and essential chilled water system capacities.

In regard to how total heat load increases are within the design margin at PUR conditions, the licensee stated the following:

C The worst-case event for heat loads on ABVS remains LOCA and MSLB.

C The ABVS system is minimally impacted by the PUR as documented in revisions to the design basis calculations. The increases in post-LOCA and MSLB temperatures have nominally increased the transmission heat loads through the containment wall to the AFW pump and electrical penetration rooms. The increase in temperature remains below the 104EF maximum design temperature.

C The increase in a transmission heat load was evaluated in the design basis for the ABVS and essential chilled water system calculations and remained well within the design capabilities of these systems. The existing ABVS system design basis calculations bound operation at PUR conditions for the piping and equipment heat loads.

The auxiliary building essential ventilation system's and safety-related portions of the auxiliary building normal ventilation system's conformance with the requirements of GDC 2 and GDC 4 to Appendix A of 10 CFR Part 50 is unchanged. Based on the licensee's rationale and evaluation, as set forth above, the NRC staff concurs with the licensee that plant operation at the proposed 2.94 percent PUR will have no impact on the ABVS. In addition, based on the NRC staff experience with reviews of PUR applications at other PWR plants, the NRC staff concludes that plant operation at the proposed 2.94 percent PUR will have no impact on the ABVS.

6.9.1.3 Turbine Building Heating, Ventilation, and Air Conditioning System

TBHVACS is not a safety-related system. The system operates during normal plant operation and during the shutdown period, depending upon heat removal needs. The TBHVACS includes the turbine building general area ventilation subsystem, switchgear room and battery room ventilation subsystem, and a lube oil room ventilation subsystem. The TBHVACS consists of supply air handling units and associated equipment to supply air at all levels; approximately 80 percent of this air is exhausted through the roof exhaust fans, and the remaining 20 percent is filtered through wall openings and other exhaust fans.

The licensee stated that the TBHVACS heat loads are based upon the piping design temperatures, pump motor rated horsepower, mechanical equipment design temperatures, electrical, control equipment and lighting loads, and transmission loads from adjacent rooms/structures. The heat loads used for the original plant design remains bounding for the PUR heat loads.

In regard to how the total heat load increases are within the design margin at PUR conditions, the licensee stated the following:

- The TBHVACS is not credited in any safety analysis.

- The turbine building heat loads evaluated in the design basis calculations remain bounding for operation at PUR conditions. These calculations are based on the original plant piping design temperatures and considered the OSG's higher secondary operating pressure and temperature.

The TBHVACS has no safety function. The system is separated from safety-related plant systems and areas; therefore, failure of the system will not compromise the operation of any essential plant systems or result in an unacceptable release of radioactivity and, therefore, PVNGS Units 1 and 3 continues to conform with the requirements of GDC 2.

Since the TBHVACS is not credited in any safety analysis, and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of plant operations at the proposed 2.94 percent PUR on the designs and performances of the TBHVACS.

6.9.1.4 Control Building Heating, Ventilation, and Air Conditioning System

The CBHVACS includes an essential HVAC subsystem and a normal HVAC subsystem. Both HVAC subsystems are provided for the following two areas:

- Control room, computer room, and associated rooms at elevation 140-ft. The essential HVAC system, as well as the habitability systems for the control room, are discussed in Section 6.4.

- ESF switchgear, ESF equipment rooms, and battery rooms without ventilation. Temperatures in the ESF air handling units' room will be less than 96EF. Essential

equipment in these areas is qualified to this temperature.

The HVAC system for the upper and lower cable spreading rooms operates in the normal mode only, and is included as a part of the normal HVAC system of the ESF switchgear, ESF equipment rooms, and battery rooms. Without ventilation, temperatures in the upper and lower cable spreading rooms will be less than 105EF. Essential equipment in these areas is qualified to this temperature.

The licensee stated:

C The CBHVACS heat loads are based on the piping design temperatures, pump rated motor horsepower, mechanical equipment design temperatures, electrical and control room lighting loads, transmission loads from adjacent structures, and personnel loads. The heat loads used for the original plant design remain bounding for the PUR heat loads.

C Modifications to instruments installed in the control room do not affect the control room heat loads.

The licensee also stated that the control building heat loads evaluated in the CBHVACS design basis calculations are not impacted and remain bounding for operation at PUR conditions. The CBHVACS is credited in safety analyses that are impacted by operation at PUR conditions. The worst-case event for heat loads on CBHVACS remains any event resulting in a LOP.

PVNGS Units 1 and 3's redundancy in emergency systems is unchanged and its compliance with the guidance of RG 1.52, "Design, Inspections, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and SRP Section 6.4 is continued. The essential system and portions of the normal system are seismic Category I, Quality Group C and thus PVNGS Units 1 and 3 continues to conform with GDC 2. Since the outside air intakes are located in a concrete tornado missile protected plenum, these systems in PVNGS Units 1 and 3 continue to conform with GDC 4.

Based on the licensee’s rationale and evaluation, the NRC staff concurs with the licensee that plant operation at the proposed 2.94 percent PUR will have no impact on the CBHVACS. In addition, PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the CBHVACS, and based on our experience with reviews of PUR applications at other PWR plants, the NRC staff concludes that plant operation at the proposed 2.94 percent PUR will have no impact on the CBHVACS.

6.10 Low Temperature Overpressure Protection

Low temperature overpressure protection (LTOP) is a set of actions prescribed to ensure that the vessel will not be subjected to conditions of brittle fracture. This is accomplished using the shutdown cooling system relief valves, whenever the cold leg temperature is below a predetermined “LTOP enable” temperature and there is the possibility of vessel pressurization.

The NRC staff reviewed the low temperature overpressure protection limits to confirm (1) that

the vessel fluence calculations were carried out using acceptable analytical methods, and (2) that the results are conservative.

PVNGS Units 1 and 3 has approved 32 effective full-power year (EFPY) pressure temperature curves, therefore, the fluence methodology and the systems calculations have been reviewed and approved. As stated in Section 5.4, the fluence calculation of record is conservative and bounds the value calculated for the proposed PUR. Similarly, the NRC staff concludes that an approved method was used for the transient analysis (i.e., heat and mass input transients).

Regarding the impact of the proposed PUR, the parameters affected are: decay heat, greater RCS and SG secondary volume, greater SG metal mass, and changed SG hydraulic characteristics. The mass and energy addition transients are impacted. The licensee performed calculations which account for the new parameter values and established that the existing pressure temperature curves bound the limits resulting from the parameter changes due to the PUR. The NRC staff finds this conclusion reasonable because there exits a large margin in the estimated fluence and consequently in the material properties at the uprated level. Therefore, the current pressure temperature and LTOP limits are acceptable for operation at the proposed PUR level.

The NRC staff review of the information submitted regarding pressure temperature curves and the LTOP, as set forth above, indicates that the current pressure temperature curves and LTOP limits are applicable for operation at the PUR level.

6.11 Miscellaneous Electrical Reviews

6.11.1 Offsite Power System

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the information, analyses and documents for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic requirement that loss of the nuclear unit, the largest operating unit on the grid or the loss of the most critical transmission line will not result in the LOOP to the plant. Branch Technical Position (BTP) Instrumentation and Control System Branch (ICSB)-11, "Stability of Offsite Power Systems," and GDC 17 outline an acceptable approach to addressing the issue of stability of offsite power systems. Acceptance criteria are based on GDC 17 of Appendix A to 10 CFR Part 50. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to 8.2 and BTPs Power System Branch (PSB) -1 and ICSB-11.

6.11.1.1 Grid Stability

The licensee evaluated all single-contingency disturbances and determined that the grid stability is maintained. The NRC staff requested in RAI #1 that the licensee address and discuss the following items:

a. Identify the nature and quantity of MVAR support necessary by each PVNGS maintain post-trip loads and minimum voltage levels.

b. Identify what MVAR contributions each PVNGS Unit is credited in its support of

the offsite power system or grid.

c. After the PUR, identify any changes in MVAR quantities associated with Items a. and b. above.

d. Discuss any compensatory measures necessary to adjust for any shortfalls in Item c. above.

e. Evaluate the impact of any MVAR shortfall listed in Item (d). above on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges with the transmission system operator.

In its response to question 1a:, the licensee stated that the change in switchyard voltage resulting from a PV unit trip is negligible for most operating conditions due to the large number of generators connected to the transmission hub (9300 MW of connected generation with more than 3000 MVAR net capability), and the automatic voltage regulation capability of these generators.

PVNGS has taken a conservative approach when only one unit is in operation to ensure adequate switchyard voltage if a design basis event should occur. When only one PVNGS unitis is operating, TS (LCO) 3.8.1, Acti on G specifies the action to be taken if the required offsite circuit(s) do not meet required capability. The Bases for TS 3.8.1, Action G.1 and G.2 provide methods to restore required capability of the offsite circuit(s). These methods do not involve actions that require the unit to be operated near its maximum MVAR loading capability.

In its response to question 1b:, the licensee stated that there are no specific levels of MVAR contribution that are credited to support the offsite power system or grid. Even under transmission grid heavy load conditions, the PVNGS generators are usually within only about 50 percent of the maximum MVAR capability.

Transmission grid power flow models consider the generator maximum MVAR output level only for steady state analysis. This value serves as a constraint on maximum steady state switchyard voltage and has no effect on the results of dynamic calculations, such as stability runs. The parameter that affects stability is the generator transient MVAR output which can reach several times the steady state limit during grid disturbances. This transient capability is unaffected by PUR.

In its response to question 1c:, the licensee stated that the maximum gross generator MVAR is usually set at 600 MVAR in the transmission grid power flow models for pre-uprate models, 541 MVAR for the post-uprate summer models, and 510 MVAR for the post-uprate winter models. Winter and summer models are provided for post-uprate, because generator output, in summer, is limited due to increased condenser back-pressure caused by higher circulating water temperatures. The minimum MVAR level is usually set at !310 MVAR based on generator minimum terminal voltage considerations, and is unaffected by PUR.

The grid power flow models are used to verify the capability of the transmission system to

operate properly following various contingencies that are simulated. PVNGS does not rely on these models to determine switchyard voltage following tripping of a PVNGS unit.
The methods that PVNGS uses to assure adequate post-trip switchyard voltage are discussed in the answer to Question 1e. below.

In its response to question 1d:, the licensee stated that no compensatory measures are necessary for the reduction in generator maximum MVAR output due to PUR based on the following:

a. PVNGS does not operate in the 500 to 600 MVAR range, but almost always below 300 MVAR. Recent transmission grid changes, such as the addition of numerous non-nuclear generating plants in the PVNGS area and the addition of the PVNGS to Rudd 525 kV transmission line, have reduced the need for reactive power support from PVNGS. Unusual grid conditions involving MVAR demand significantly higher than historic levels would only be caused by excessively high customer loading and high transmission line flows. In this case, to provide for the customer loads, many of the non-nuclear generators near PVNGS would have to be operating, so they, too, would be sharing in transmission grid MVAR support. Furthermore, a high PVNGS switchyard voltage would be needed to support sagging distribution system voltages in Phoenix in this scenario.

b. For transmission grid studies, the only way for the analyst to force the generator MVAR output to reach its maximum limit is to artificially lower bus voltages to unrealistic levels at other switchyards. Whether pre or post-uprate, the results of the heavy boosting conditions demonstrate higher stability margins than for heavy bucking conditions. Since the heavy boosting condition is not the limiting case with regard to stability, the change in maximum MVAR capability is inconsequential for these studies. For the bucking case, the MVAR absorption level of the PVNGS generator is forced down to a level that causes generator terminal voltage to reach its minimum limit of 22.8 kV while the generator is still well within its MVAR capability band. Therefore, there is no impact on stability results due to PUR.

In its response to question 1e:, the licensee stated that adequacy of post-trip voltage when two or three units are initially operating is assured as discussed in a letter from PVNGS to the NRC dated July 16, 1999, “Response to NRC RAI regarding proposed amendment to TS 3.8.1, AC Sources - Operating and 3.3.7, Diesel Generator (DG) - Loss of Voltage Start (LOVS).”
As discussed above, since 1999 a number of transmission grid changes have occurred that have reduced the need for reactive power support from PVNGS. As discussed in the 1999 letter, there is no credible scenario where tripping of one PVNGS unit with one or more of the other units still on line would result in inadequate switchyard voltage.

When only one PVNGS unit is initially operating, adequacy of post-trip voltage is assured by the conservative measures governed by TS LCO 3.8.1, Action G discussed above. Reduction of generator maximum MVAR capability has no effect on the allowed plant operating parameters during this condition.

LCO 3.8.1, Action G does not rely on any information exchanges with the transmission system operator. Switchyard voltage is monitored by a meter in the PVNGS Unit 1 control room.

Generator gross MVAR output is monitored by a meter in the affected unit's control room.

Additionally, the NRC staff asked the licensee in RAI #2 to provide major assumptions, results, and conclusions of the current grid reliability analysis. Furthermore, the NRC staff asked the licensee to provide contingency management details necessary for maintaining grid stability of the control area surrounding the PVNGS site including the contingencies analyzed. Also, does the contingency analysis include the tripping of all three PVNGS Units? If not, provide a discussion. In response to above question, the licensee, on July 19, 2005, stated that PUR has no effect on the contingencies that are analyzed to demonstrate offsite power stability. Analyzed contingencies, as discussed in the UFSAR Section 8.2.2, are tripping of one PVNGS unit, faulting and tripping of the most significant transmission line, and loss of the largest major customer load - each at maximum boosting and maximum bucking conditions and with 7 percent PVNGS generation margin added for conservatism. These studies conclude that such contingencies would not result in instability providing that the transmission system is operated in accordance with the PVNGS transmission system operating procedure. Construction of numerous non-nuclear generating stations in the PVNGS area created the possibility that heavy generating levels could compromise stability margins under certain operating conditions involving heavy bucking of switchyard voltage (absorption of MVARS by the generators) and transmission lines out of service prior to the disturbance. To ensure an adequate stability margin, the grid operator implemented an operating procedure that limits the generation levels of the non-nuclear plants when such conditions occur.

Simultaneous tripping of multiple PVNGS units is not included in the stability analysis discussed in UFSAR Section 8.2.2. A design basis event in one unit, such as a LOCA, would not cause tripping of the other units. Although a major transmission system disturbance could cause tripping of multiple units due to loss of the PVNGS transmission system, such tripping would be a consequence, rather than a cause, of the event.

Analysis of the effects of major transmission system disturbances is under the purview of transmission grid organizations, the Western Electric Coordinating Council (WECC), and North American Electric Reliability Council (NERC), rather than PVNGS. It is the responsibility of these organizations to establish reliability criteria for transmission system design and operation and to verify that those criteria are met. Some of the studies that are performed for this purpose consider the effect of the simultaneous loss of multiple elements, such as two transmission lines or two generating units, including simultaneous tripping of two PVNGS generators. These studies are not part of the PVNGS design bases.

The transmission lines that comprise the California to Oregon intertie (COI) can be affected by loss of significant generation resources in the Southwest. This limitation has been recognized for many years, and operating constraints and remedial action scheme have been implemented to protect against instability. The remedial action scheme is designed to mitigate the effects of the tripping of two PVNGS units, considering the effects of PUR.

The NERC reliability criteria include consideration for "Category D" contingencies. This level of contingency is defined as an "Extreme event resulting in two or more (multiple) elements removed or cascading out of service." This could be caused by various initiators, such as "Loss of all generating units at a station." However, it is recognized that such an event "May involve substantial loss of customer demand and generation in a widespread area or areas," and that

"Portions or all of the interconnected systems may or may not achieve a new, stable operating point." Simultaneous tripping of three PVNGS units is in this category. NERC does not require that transmission systems be designed or operated to ensure stability or continuity of offsite power to nuclear generating plants during such events. If such a requirement were implemented, it would require substantial changes in the amount of spinning reserve needed, as well as additional operating margin for transmission lines involving either curtailment of flow during normal operation or installation of additional lines. The probability of grid instability during an event that involves tripping of three PVNGS units is not significantly increased as a result of PUR, since the remedial action scheme sheds at least as much load as the uprate levels of two units. Therefore, the NRC staff concludes that the impact of the PUR on grid stability is acceptable to meet GDC 17 for grid stability.

6.11.1.2 Main Power Transformer

The licensee evaluated the impact of PUR on the main power transformers (one transformer per phase) and determined that the PUR will raise the main transformer oil temperature. This oil temperature increase remains below the transformer's rated temperature capacity. Although not required by the PUR analysis, the main transformer cooling will be modified to increase the reliability of the system. Therefore, the NRC staff finds this acceptable.

6.11.1.3 Isolated Phase Bus

The isolated phase bus (or isophase bus) is the electrical connection from the main generator output terminals to the low voltage terminals of the main transformer and to the high voltage terminals of the unit auxiliary transformer. The isophase bus rating is 1600 MVA (forced cooling). This rating is greater than the main generator and main transformer ratings. Therefore, the PUR has no impact on the isophase bus operability.

6.11.1.4 Auxiliary Power System

The licensee evaluated the impact of an increased load from the RCPs, condensate pumps, and heater drain pumps on the startup transformers/unit auxiliary transformers. The total electrical load increase due to PUR is within the rated capacities of the startup transformers/unit auxiliary transformers. The effect of the horsepower load increase on the non-Class 1E 13.8 and 4.16 kV auxiliary electrical distribution system was also evaluated. The higher current due to increased non-class 1E pump break horse power (BHP) decreases the voltage at the 4.16 kV ESF bus and downstream equipment when house loads are fed from startup transformers. However, analysis of this effect demonstrates that the voltage decrease will not result in spurious operation of the loss of voltage or degraded voltage relay. In addition, the decreased voltage will not adversely affect the function of any class 1E equipment downstream of the breakers and relays. In RAI # 6, the NRC staff asked the licensee to provide the basis of the above conclusion. In its response, on July 19, 2005, the licensee provided the BHP and terminal voltage changes of the affected components. The licensee stated that the voltage on the Class 1E 4160 V buses needs to recover to at least 3805 V following automatic load sequencing (resulting from a design basis event such as a LOCA) to ensure that the degraded voltage relays reset, thus avoiding their actuation. At minimum allowable switchyard voltage (taking into account metering uncertainty) the voltage will recover to at least 3842 V. The increased loading on the non-Class 1E buses lowers this value about

4 V, so the Class 1E buses will still recover to at least 3838 V which provides margin above the 3805 V limit. On the basis of its review, the staff finds the licensee's conclusion acceptable.

The affected 13.8 kV switchgear, circuit breakers and cables and 4.16 kV transformers, switchgear, circuit breakers and cables were compared to their rated capacities. The values increased but were below each component's rated capacity. Therefore, the PUR has no impact on the startup transformers/ unit auxiliary transformers and other equipment.

6.11.1.5 Conclusion

The NRC staff has reviewed the offsite power system and concludes that it meets the requirements of GDC 17 for the PUR. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the licensee's proposed PUR acceptable.

6.11.2 AC Power Systems (Onsite)

The AC onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the AC onsite power system. Acceptance criteria are based on GDC 17 as it relates to the capability of the AC onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

The two EDGs provide an independent source of Class 1E onsite power (4160 volts) for each of the two trains of ESF bus. Evaluation of the loads on each of the safety buses demonstrated that those loads do not increase due to PUR. Therefore, the EDGs remain capable of supplying ESF equipment during all operating and accident conditions.

The NRC staff has reviewed the AC onsite power system which includes the standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment, and found it to be consistent with GDC 17. Therefore, the NRC staff finds the licensee's proposed PUR acceptable with respect to the onsite AC power system.

6.11.3 DC Power Systems (Onsite)

The DC power systems include those DC power sources, distribution systems, and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the DC onsite power system. Acceptance criteria are based on GDC 17 and 10 CFR 50.63 as they relate to the capability of the onsite electrical power to facilitate the functioning of structures, systems, and components important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

In response to the NRC staff's RAI #9 regarding the effect of PUR on onsite DC power system, the licensee on July 19, 2005, stated that PUR has no effect on the design or operation of the DC power system or DC load devices. Therefore, loading, voltage, and short circuit values are

unaffected. On the basis of its review, the NRC staff agrees with the licensee that there are no changes to the DC loads, voltage drops, or short circuit current values.

The NRC staff has reviewed the DC onsite power system and concludes that it meets the requirements of GDC 17. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the licensee's proposed PUR acceptable with respect to the design of the DC onsite power system.

6.11.4 Station Blackout

The term "station blackout" refers to the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant. Station blackout (SBO) therefore involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or the loss of power from alternate AC (AAC) source. The staff's review covers the station blackout event for an established period of time, and to recover therefrom. Acceptance criteria are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and 8.2 Appendix B.

The licensee stated that the SBO coping duration for the PVNGS is 4 hours and two gas turbine generators and their associated equipment act as the AAC power source. The AAC power source was designed to be available within 1 hour of the onset of an SBO and would power the equipment necessary to cope with an SBO for the remaining 3 hours of the coping duration.

In response to staff's RAI # 4 regarding offsite ac power design characteristic group (P group) change from P1 to P3 due to loss of offsite power to all three Palo Verde units on June 14, 2004, the licensee, on July 19, 2005, stated that PUR has no effect on the frequency or duration of SBO events. However, the licensee agreed to change the offsite power design characteristic group for PVNGS from P1 to P3 because of June 14, 2004, event. As a result, SBO coping duration is changed from 4 hours to 16 hours with EDG classification of "C" and EDG reliability of 0.95.

The 4 hour coping strategy (original study) assumed that the unit would achieve and maintain hot standby using the atmospheric dump valves (ADVs) for heat removal and that charging pumps would be used for RCS inventory control. The 16 hour coping strategy (revised study) assumes minimal operator action in the first hour, and at the end of four hours the operators would start a cooldown to shutdown cooling entry conditions during the remaining 12 hours of the coping period. The ADVs will be used for heat removal, the pressurizer vent will be used for RCS pressure control, and RCS inventory will be controlled using a high pressure safety injection pump.

The licensee stated that the decay heat used for the 16 hour coping analyses is based on the ANSI/ANS-5.1 - 1979 decay heat curve, plus a 2 sigma uncertainty. The time dependent decay heat is developed using the following parameters:

- Fuel enrichment = 5%

- Fuel burnup up to 70,000 MWD/MTU
- Three operating cycles, each cycle consists of 505 days plus a 25 day outage
- Power level = 3990 MWt

The licensee stated that the resultant decay heat curve is conservative, bounding and consistent with industry practices. Areas of evaluation for the 16 hour coping period included the following:

- RCS Inventory
- Condensate Inventory
- Class 1E Battery Capacity
- Compressed Air Capacity
- Loss of Ventilation in Areas Containing Equipment Needed During an SBO
- Containment Isolation
- Communication

Additionally, the licensee indicated that some procedures will be revised and compressed air system will be supplemented. The licensee has provided a list of potential procedures changes to change from a 4 hour coping strategy to a 16 hour coping strategy. The licensee has provided evaluations and analyses for coping with an SBO for 16 hours in a letter dated October 28, 2005. On October 21, 2005, the licensee informed the NRC that APS will implement the changes needed to revise from four hour SBO coping duration to a 16 hour coping duration within six months following the NRC approval of the proposed coping changes.

The staff has determined that the licensee has provided sufficient information with respect to SBO to proceed with the issuance of PUR. The staff has accepted the licensee's 16 hour coping implementation date which will be approximately October 28, 2006 (six months for the staff's review plus six months for the licensee's implementation). The staff finds that the licensee's existing SBO coping strategy is acceptable until the 16 hour SBO coping implementation date . The staff's finding is based on the following: (1) the PVNGS grid is considered to be reliable after fixing the problem associated with the loss of offsite power event of June 14, 2004, at PVNGS; (2) the EDGs are reliable; (3) the AAC power source has sufficient capacity and capability for more than 4 hours; (4) probability of having an SBO during this time period is very low. Therefore, the staff finds the licensee's proposed PUR acceptable based on the licensee's commitment to complete its evaluations and analyses for coping with an SBO for 16 hours. In accordance with the licensee's proposal contained in its October 21, 2005, letter, a license condition has been added to the amendment pages for all three PVNGS units to incorporate this commitment.

The staff has evaluated the effect of PUR on the necessary electrical systems and environmental qualification of electrical components. Results of these evaluations show that the increase in a core thermal power would have negligible impact on the grid stability, SBO, or the environmental qualification of electrical components. This is consistent with the requirements of GDC 17, 10 CFR 50.63, and 10 CFR 50.49.

6.11.5 Environmental Qualification of Electrical Equipment

The term "environmental qualification" applies to equipment important to safety to assure this

equipment remains functional during and following design basis events. The NRC staff's review covers the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The NRC staff's review is to ensure compliance with the acceptance criteria which ensures that the equipment continues to be capable of performing its design safety functions under all normal environmental conditions, AOOs, and accident and post-accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important to safety that is located in a harsh environment. Specific Review criteria are contained in SRP Section 3.11.

In accordance with 10 CFR 50.49, safety related electrical equipment must be qualified to survive the temperature, pressure, and radiation environment at their specific location during normal and accident operating conditions. The LOCA and MSLB analyses were revised for PUR. The revised LOCA and MSLB analyses resulted in a change to the environmental parameters for the equipment required by 10 CFR 50.49. The licensee determined that the total integrated gamma dose inside and outside the containment building has been reduced due to increased sump water volume as a result of increase in RCS volume. The beta dose in the containment has increased as a result of the increase power level. The revised gamma doses inside and outside containment and beta doses inside containment for each component was compared with EQ test doses. This comparison demonstrated that the electrical equipment remains qualified as required by 10 CFR 50.49 and in accordance with Institute of Electrical and Electronics Engineers, Inc. (IEEE) 323-1974. A comparison of the revised LOCA and MSLB (inside and outside containment) temperature profiles with the existing profiles indicated that the PUR peak accident temperature is decreased for MSLB and increased for LOCA. The containment pressure profile during LOCA also changed. Conservative MSLB and LOCA long-term pressure and temperature profiles were developed. These 180-day profiles were used to qualify the containment and main steam support structures equipment. The assessment of EQ data files and test reports reveals that the equipment required remains qualified with the exception of non-standard Raychem splices and ICI connectors. ICI connectors and non-standard Raychem splices have been qualified for PUR conditions.

The NRC staff has reviewed the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The NRC staff concludes that the environmental qualification of the electrical equipment is acceptable and meets the relevant requirements of 10 CFR 50.49. Therefore, the NRC staff finds the licensee's proposed PUR acceptable with respect to environmental qualification of electrical equipment.

6.11.6 Miscellaneous Electrical Reviews Conclusion

The NRC staff has evaluated the effect of PUR on the necessary electrical systems and environmental qualification of electrical components. Results of these evaluations show that the increase in a core thermal power would have negligible impact on the grid stability, SBO, or the environmental qualification of electrical components. This is consistent with the requirements of GDC 17, 10 CFR 50.63, and 10 CFR 50.49. However, it should be noted that approval of the licensee's proposed PUR is contingent upon the NRC staff's approval of the licensee's evaluations and analyses for coping with an SBO for 16 hours that will be submitted by October 31, 2005

6.12 Instrumentation and Controls

Nuclear power plants are licensed to operate at a specified core thermal power. The instrument measurement uncertainty should be considered to avoid exceeding the power level assumed in the design basis transient and accident analysis. The safety-related instrument trip setpoints are calculated to ensure that sufficient allowance exists between the trip setpoint and the safety limit to account for instrument uncertainties. The NRC's regulatory requirements related to this review can be found in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(c)(1)(ii)(A) which requires that, where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the most severe abnormal situation anticipated without exceeding a safety limit. LSSS are settings for automatic protective devices related to variables having significant safety functions. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated systems. Regulatory Guide (RG) 1.105, Revision 3, "Setpoint for Safety-Related Instrumentation," is used to evaluate the conformance with 10 CFR 50.36.

6.12.1 Suitability of Existing Instruments

PVNGS Units 1 and 3 RPS initiates a reactor shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and the RCS pressure boundary during anticipated operational occurrences (AOOs) and to initiate the engineered safety feature (ESF) systems in mitigating accidents.

The RPS is designed to trip the reactor by de-energizing the control element drive mechanism (CEDM) coils whenever any monitored condition reaches a trip setpoint. To meet the design requirements for redundancy and reliability for each measured variable, more than one, and often as many as four channels are used. In many cases, field sensors that input to the RPS are shared with the ESFAS.

The licensee stated that the PVNGS existing instrumentation and control systems will continue to perform its intended safety functions under the SPU operations and that no modification on the protection system is required except for nominal trip setpoints and TS allowable value (AV) changes in some of the reactor trip and ESFAS functions to support steam generators replacement and SPU power level conditions. Except for the allowable value associated with the low SG pressure trip and main steam isolation system actuation setpoint changes, the TS allowable values associated with the RPS and the ESFAS setpoints will remain unchanged.

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, the licensee is referencing the power uprate licensing report (PULR) for Unit 2 as the basis for the analyses and evaluations for Units 1 and 3. Based on the information provided in the submittal, the staff finds the existing instruments are suitable for Units 1 and 3 at power uprated conditions.

6.12.2 Instrument Setpoint Methodology

The TS defines LSSS as an allowable value. During reviews of recent proposed license amendments that contain changes to LSSS setpoints, the NRC staff has identified concerns regarding the method used by the licensee to determine the allowable values. Allowable values are used in the TS as LSSS to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The allowable value is an operability limit in the TS, but the Bases must state that the limiting trip setpoint preserves the safety limit and therefore the LSSS setpoints required by 10 CFR 50.36 are met. In order for the NRC staff to assess the acceptability of this amendment request, by letter dated March 31, 2005, from Mr. J. Lyons, USNRC, to Mr. A. Marion, Nuclear Energy Institute (NEI), "Instrumentation, Systems, and Automation Society S67.04 Methods for Determing Trip Setpoints and Allowable Values for Safety-Related Instrumentation," the NRC staff requested the licensee to provide additional information related to:

(1) The setpoint methodology used to establish allowable values associated with LSSS setpoints.
(2) Discuss how the methodology and controls at the plant in place ensure that the analytical limit (AL) associated with an LSSS will not be exceeded.
(3) How the TS surveillances ensure the operability of the instrument channel.

Additional information was provided by the NRC staff in a letter dated August 23, 2005, from Mr. B. Boger, NRC, to Mr. A. Marion, NEI, "Instrumentation, Systems, and Automation Society (ISA) S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-related Instrumentation." In this letter, the NRC staff discussed the concepts that it felt would satisfactorily address both the NRC staff's and industry's concerns with instrument settings, and ensure compliance with 10 CFR 50.36, "Technical Specifications."

The NRC staff requested additional information (both March 31 and August 23, 2005, letters) associated with operability of instrument settings related to this amendment request. The NRC staff believes that demonstration of the operability of instruments is required to ensure compliance with the requirements of 10 CFR 50.36(c)(3) which requires that TS surveillances demonstrate that the plant is operating within its safety limits. Verification that the instrument is functioning as required is an integral part of this periodic testing. In addition, 10 CFR 50.36(c)(1)(ii)(A), which discusses the requirements for LSSS, states that "If, during operation, it is determined that the automatic safety system equipment does not *function as required* (emphasis added), the licensee shall take appropriate action, which may include shutting down the reactor."

The NRC staff's position is that simply resetting an instrument whose setpoint is found outside the predefined test acceptance criteria band back to its nominal setpoint and entering the data into a corrective action program, without a prompt evaluation of the condition, is not sufficient to determine the operability of the instrument that is being placed back into service. This is because an instrument may be degraded or fail due to conditions other than statistical variations in uncertainties, including drift. The NRC staff and the NEI Setpoint Methods Task Force (SMTF) reached agreement on this issue during the June 2, 2005 meeting as part of Concept 2 (as-found trip setpoint (TSP)) of the NEI May 18, 2005, letter. This letter states that, if the as-found TSP exceeds a predefined test acceptance criteria band during periodic

surveillance, additional evaluation and potential corrective action *"is"* (emphasis added) warranted as necessary to ensure continued performance of the specified safety function. Incorporating this requirement into theTS provides reasonable assurance that the next surveillance as-found value of the TSP will continue to protect plant safety limits.

The NEI May 18, 2005, letter Concept 7 (Operability) discusses factors that could be considered in this SE. It should be noted that, although the TS would contain a note to verify that the as-found TSP was within the predefined test acceptance criteria band and that exceeding the limits would warrant additional evaluation, the detailed discussion of the evaluation process and the factors to be considered would not be required in either the TS or the Bases, and that the process for evaluation is consistent with the guidance that has recently been developed by the NRC staff and the NEI Operability Determination Process Task Force as part of the effort to revise the operability guidance in Generic Letter (GL) 91-18.

In response to the NRC staff's positions on the instrument setpoint methodology, the licensee provided information and clarifications by supplemental letters dated July 9, September 29, and November 1, 2005.

The NRC staff finds the licensee responses to the RAI that include TS requirements which implement the concepts described in the NEI May 18, 2005, letter to be acceptable. The NRC staff requested the licensee provide a brief description of the methodology used to determine its setpoints. The purpose of this request was to solicit information from the licensee to determine whether TSPs were calculated in a manner that accounted for credible uncertainties associated with the instrument channel. This could be accomplished by referring to RG 1.105, "Setpoints for Safety-Related Instrumentation," or an NRC approved plant-specific setpoint methodology. In addition, a predefined test acceptance criteria band should be developed consistent with the assumptions and uncertainties associated with the tested portion of the instrument channel and the determination of the TSP calculated to protect the safety limits. This information is necessary for the NRC staff to conclude that the TSP provides reasonable assurance that the safety limits will be protected, a finding necessary to support issuance of the amendment request.

The PVNGS TSs define LSSS as an allowable value. PVNGS is committed to RG 1.105, Revision 1 as basis for meeting the requirements of 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 13 and 20. PVNGS used the principles of ANSI/ISA S67.04-1988, "Setpoints for Nuclear Safety-Related Instrumentation" and RP67.04 (then draft 9), "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" as the design guide for instrument uncertainty and setpoint determination. The total loop uncertainty (TLU), which is used to calculate a limiting setpoint, is determined by the square root sum of the squares combination of the COT (potential instrument uncertainties expected during Channel Operability Testing) and nCOT (composite of all other potential instrument uncertainties not addressed in the COT). The PVNGS methodology and resulting calculation can demonstrate that the allowance between the AL and the allowable value exceeds the magnitude of the nCOT and that of the entire TLU for the LSSS value being changed under this amendment request. The AL and the associated safety limit will be preserved if a trip setpoint is found to be within the allowable value during surveillance testing.

In its September 29, 2005 letter, the licensee proposes to add the following note to each of the

protective function that has allowable value changed due to the RSG generator program:

Note 1. If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value the channel shall be declared inoperable.

Note 2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the UFSAR [or a document incorporated into the UFSAR such as the technical requirements manual].

These notes are being added to the LCOs associated with the calibration requirements for these instruments. The September 29 and November 1, 2005, letters also contained the TS Bases changes associated with these proposed footnotes. The NRC staff reviewed the proposed Bases changes to ensure consistency with the TS and design bases for the PVNGS units. Based on the information submitted on July 9, September 29, and November 1, 2005, the NRC staff finds acceptable the licensee responses to the RAIs that include TS requirements which implement the concepts described in the NEI May 18, 2005, letter.

6.12.3 Instruments and Control-related TSs Changes Related to the PUR

The licensee stated that the TS allowable values, setpoints, and response times are not being changed except for the low SG pressure trip allowable value and the main steam isolation system actuation allowable value. These allowable value changes are due to the new operating conditions.

LCO 3.3.1, RPS 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 SG Pressure - Low (LSGP) trip function (functions 6 and 7 in Table 3.3.1-1) provides protection against an excessive rate of heat extraction from the SGs and a resulting rapid, uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the 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.

LCO 3.3.2, RPS 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 LSGP trip function (functions 2 and 3 in Table 3.3.2-1) provides shutdown margin to prevent or minimize a return to power following a large MSLB in Mode 3.

LCO 3.3.5, ESFAS Instrumentation, and Table 3.3.5-1, which it references specify the required number of channels operable for each EFS 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 LSGP 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.

The licensee proposed changing the LSGP allowable value from $890 psia to $955 psia in the above three LCOs after power uprated to 3990 MWt RTP. The licensee stated that the larger SGs and greater plant power output will result in a higher SG operating pressure. To ensure that the revised NSSS control systems would provide an acceptable plant response at the uprated power conditions, the licensee analyzed the standard NSSS control systems design-basis transients using the existing control system evaluation code. The licensee stated that the analysis results demonstrated acceptable plant responses to the analyzed transients, and confirmed that there is adequate margins between the design setpoints, limiting setpoints, and ALs. The NRC staff reviewed the results presented by the licensee and concluded that the requested increase in the LSGP setpoint from $890 psia to $955 psia is acceptable.

6.12.4 Instruments and Controls Conclusion

Based on the review of the PVNGS Units 1 and 3 PUR amendment request, the NRC staff finds that the PVNGS instrumentation and control systems will continue to perform their intended functions as required by plant license which complies with the NRC's acceptance criteria related to the quality of design of protection and control systems.

The licensee has properly entered the footnotes that apply to the allowable value in TS Tables 3.3.1-1, 3.3.2-1, 3.3.5-1, and 3.7.1-1 that address the instrument channel operability. The NRC staff finds this is in conformance with 10 CFR 50.36, and therefore is acceptable.

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6.13 Essential Spray Pond System

The ultimate heat sink (UHS) provides heat dissipation capability for the reactor and its essential equipment through the essential spray pond system (ESPS) during normal shutdown, refueling, and accident conditions. The UHS consists of two independent seismic Category I essential spray ponds. Each pond serves one train of the ESPS. Redundant manually operated seismically qualified butterfly valves are provided between ponds, such that the total inventory from both ponds is available to either ESPS train. Warm water returned to the ponds is pumped through ESPS spray nozzles. Heat dissipation is by evaporation to the atmosphere. The spray nozzles are designed to provide an optimum spray spectrum and are arranged to minimize interference between sprays. Makeup to the ponds is provided by the nonessential domestic water system with a backup source available from the nonessential station makeup water reservoir.

In accordance with the Improved Technical Specifications, the UHS is only required to have sufficient water inventory without makeup for a duration of 26 days following a LOCA. During the operating license application review for PVNGS Units 1 and 3, the NRC staff, in Supplement No. 3 to its SE dated September 1982, concluded that the guidance in RG 1.27 regarding sufficient UHS water capacity was satisfied.

Based on our review of the licensee's rationale and the experience gained from our review of PUR applications for similar PWR plants, we find that plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the UHS, and that the PUR has an insignificant or no impact on the UHS and ESPS. Therefore, the NRC staff concludes that the UHS and ESPS remain acceptable for PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR.

7.0 MISCELLANEOUS TOPICS

7.1 Post-Loss-of-Coolant Accident Hydrogen Generation

APS's PURLR discusses the licensee's review of the effect of the PUR on hydrogen generation and distribution in the containment. The licensee used the analysis methods discussed in the UFSAR to analyze the effects of post-accident hydrogen generation.

The licensee determined the core wide oxidation rate following a LOCA at PUR conditions to be 0.86 percent. This is less than the limit specified in 10 CFR 50.46 and is therefore acceptable. The existing inventories of aluminum and zinc are not changed by the PUR. The hydrogen recombiners are assumed to start at 100 hours. This is consistent with the PVNGS Units 1 and 3 design basis (UFSAR Section 6.2.5.1). The licensee has determined that the peak predicted hydrogen concentration in the containment remains below 3.99% by volume and is therefore acceptable.

Under postulated LOCA conditions, the Reactor Drain Tank room may become essentially a closed room with the venting only from an annular opening in the ceiling. This situation is discussed in UFSAR Section 6.2.5. The licensee's treatment of this situation was previously found acceptable in an NRC SE dated May 23, 1996. The licensee used the same methodology as used for the original design to show that the gas plume leaving the room remains below the combustible limit. This conclusion is based on conservative post-LOCA conditions which bound those predicted for the PUR. The NRC staff finds the licensee's analysis acceptable since: (1) the core oxidation following a LOCA is less than 1 percent, (2) the hydrogen concentration in the main containment volume remains less than 4 percent and the gas plume from the Reactor Drain Tank room is below the combustible limit when analyzed at the higher power, and (3) acceptable methods were used to perform these analyses.

7.2 Fire Protection Program

GDC 3, "Fire protection," addresses generic issues for nuclear power plants regarding the design of SSCs to minimize the probability and effects of fires and explosions. In addition, 10 CFR 50.48 defines requirements for licensees' fire protection programs. PVNGS Units 1 and 3 were licensed to operate on June 1, 1985 and November 25, 1987, respectively. Therefore, PVNGS Units 1 and 3 are not subject to the requirements of Appendix R to 10 CFR Part 50, which is only applicable to plants licensed to operate prior to January 1, 1979. The requirements have been reflected in SRP Section 9.5.1, and the approval and requirements for implementing the fire protection program have also been incorporated into an operating license condition for each operating unit. The NRC staff reviewed the licensee's PUR submittals to

determine the effects of increasing the plant power rating on the approved fire protection plan.

In Section 9.7 of the PURLR, "Fire Protection Program," the licencee states that operation of PVNGS Units 1 and 3 at the proposed 2.94 percent PUR will not affect the design or operation of the plant's fire detection systems, fire suppression systems, or fire barrier assemblies installed to satisfy NRC fire protection requirements, or result in an increase in the potential for a radiological release resulting from a fire. Any changes to the plant configuration or combustible loading as a result of modifications necessary to implement the PUR will be evaluated by the licensee under the plant's existing NRC-approved fire protection program.

The licensee performed a thermal-hydraulic analysis of the important plant process parameters following a fire assuming PUR conditions. This analysis indicates that only the operator time constraints related to the time needed to deplete the CST and the reactor makeup water tank volumes during plant cool down are affected by the PUR. The licensee has concluded that the safe shutdown methodology and results identified in the UFSAR are maintained considering the modified operator response times for the PUR. All other important plant process parameters and time constraints remain unchanged. The licensee has made no other changes to the plant's hot standby SSCs, components or procedures necessary to achieve and maintain cold shutdown conditions within 72 hours.

The NRC staff has reviewed the licensee's rationale and assessment, and based on that rationale, we conclude that the licensee has adequately considered the effect of the PUR on the fire protection program.

7.3 High Energy Line Breaks Outside Containment

With regard to protection for SSCs important to safety against pipe breaks outside containment, PVNGS Units 1 and 3 is designed in accordance with the guidelines of SRP BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." The plant design accommodates the effects of postulated pipe breaks and cracks, including pipe whip, jet impingement and environmental effects. The means used to protect essential/safety-related systems and components include physical separation, enclosures, pipe whip restrainers, and equipment shields.

The licensee assessed the systems evaluated in the UFSAR to determine the effects of plant operations at the proposed 2.94 percent PUR on high energy line breaks (HELB) outside containment. System operating parameters for the PUR were assessed against the system pressure and/or temperature parameters used in the existing plant bases to demonstrate the acceptability for HELB effects. The assessment shows that plant operations at the proposed 2.94 percent PUR have an insignificant or no impact on the consequences (e.g. environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment. In addition, there is no impact on the methods of protection of safety-related systems from HELBs.

We reviewed the licencee's rationale, and based on that rationale, as set forth above, and the experience gained from our review of PUR applications for similar PWR plants, the NRC staff concurs with the licensee's conclusion that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR have an insignificant or no impact on the consequences (e.g. environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment.

Therefore the NRC staff concludes that the PUR is acceptable in this regard.

7.4 Erosion/Corrosion Program

An increase in power will affect fluid velocities, temperatures, and moisture content within many systems of the plant. These parameters directly influence the erosion/corrosion characteristics of these systems. The licensee has established an erosion/corrosion program to monitor pipe wall thinning in single and two-phase flow in high energy carbon steel piping and for identifying corrosion damaged components that should be repaired or replaced. This erosion/corrosion program identifies the piping components and locations that should be monitored for flow accelerated corrosion (FAC). NRC GL 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” provides guidance for inspecting pipes and components subject to FAC and requests that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems.

The licensee stated that its FAC program fully conforms to NRC GL 89-08. The plant components which are susceptible to FAC are modeled in CHECWORKS, EPRI’s predictive model for FAC. The program and procedures are in place to monitor and maintain the structural integrity of high-energy carbon steel piping. Changes to the model due to the PUR will be made before the installation of the RSGs and implementation of the PUR. The changes will include all parameters affecting FAC and at that time the component wear rates before and after the PUR can be compared. The licensee stated that the model changes are expected to be minor and will result in insignificant changes to plant component susceptibility to FAC.

On the basis of the information the licensee provided, the NRC staff concludes that the proposed PUR is acceptable with respect to the erosion/corrosion program because (1) the PUR will result in negligible effects on the parameters that influence FAC and on the licensee’s erosion/corrosion program, and (2) the erosion/corrosion program has adequate provisions to manage erosion/corrosion in high energy piping under the PUR conditions.

7.5 Flooding

To assure conformance with the requirements of 10 CFR Part 50, Appendix A, GDC 2, “Design bases for protection against natural phenomena,” the review of the plant flood protection include all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

7.5.1 Containment Sump pH and Containment Flooding

The licensee re-examined the effect of increased steam generator primary water inventory on the post-LOCA containment flood level. The licensee concluded that the increase in water was small compared to the post-LOCA total volume of water in the containment and does not change the conclusion of the existing analyses that the containment flood level remains below the current acceptable level. Since the current containment flood limit is not exceeded, the NRC staff finds the PUR to be acceptable with respect to containment flooding.

7.5.2 Outside Containment Flooding

The source of flooding at the site is a probable maximum flood (PMF) for various rivers and washes in site vicinity. These desert water courses are normally dry with flow occurring in them only as a result of rainfall runoff. The PVNGS Units 1 and 3 site is located at an elevation above the PMF level occurring in the desert streams in the area. Therefore, all safety-related systems and components are located above the PMF level.

During the PVNGS Units 1 and 3 operating license application review, the NRC staff, in an SE report dated November 1981, concluded that the design of the facility for flood protection conforms to the requirements of GDC 2 with respect to protection against natural phenomena, and conforms to the guidelines of NRC RG 1.102 concerning flood protection. Also, the plant has adequate protection for safety-related equipment from the effects of postulated piping failure outside containment in accordance with the guidelines of SRP BTP ASB 3-1, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.”

Since events of natural phenomena are not power dependent, we conclude that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR will not result in an increase in the probability of flooding caused by PMF.

With respect to building flooding in areas outside the containment, the worst-case flow from high or moderate energy piping systems including the main steam support structure and main FW system are used in the existing flooding analyses. The licensee performed evaluations on the effect of the PUR on building flooding in areas outside the containment and concluded that the effect of the PUR on building flooding in areas outside the containment is bounded by the existing analysis as discussed in the UFSAR Section 3.6.1.

We have reviewed the licensee’s rationale, and based on that rationale, as set forth above, and the experience gained from our review of PUR applications for similar PWR plants, the NRC staff concurs with the licensee's conclusion that PVNGS Units 1 and 3 operations at the proposed 2.94 percent PUR will have an insignificant or no impact on building flooding resulting from high or moderate energy pipe break outside containment.

7.6 Human Factors Considerations

The following is the guidance with respect to information on human factors considerations that is needed by the NRC staff as part of its review of proposed power uprates.

16. Changes in Emergency and Abnormal Operating Procedures

    Describe how the proposed power uprate will change the plant emergency and abnormal operating procedures.

17. Changes to Operator Actions Sensitive to PUR

    Describe any new operator actions needed as a result of the proposed PUR. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the PUR.

18. Changes to Control Room Controls, Displays and Alarms

Describe any changes the proposed PUR will have on the operator interfaces for control room controls, displays, and alarms. Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed PUR and how operators were tested to determine that they could use the instruments reliably.

19. Changes on the Safety Parameter Display System

Describe any changes the proposed PUR will have on the Safety Parameter Display System. How will the operators know of the changes?

20. Changes to the Operator Training Program and the Control Room Simulator

Describe any changes the proposed PUR will have on the operator training program and the plant referenced control room simulator, and provide the implementation schedule for making the changes.

For Item 1 above, in PURLR Section 9.12.2, the licensee states that an assessment of the expected plant response indicated that minor EOP/Abnormal Operating Procedures (AOP) changes are expected which would not affect operator actions or mitigation strategies that are taken credit for in accident analyses. The licensee explained that the PUR results in changes in operating procedures, such as surveillance tests, normal operating, general operating, and/or alarm response procedures. The procedure changes will be incorporated before operation of Unit 2 at the higher power levels of the PUR.

For Item 2, in PURLR Section 9.12.2, the licensee states that there are no changes to operator credited actions or mitigation strategies.

For Items 3 and 4, in PURLR Section 9.12.1, the licensee states that the PUR will have a limited impact on the operator interfaces for control room displays, controls, and alarms. There will be a few alarm setpoints and indicators changed in the control room because of the PUR. The Qualified Safety Parameter Display System will be modified for the larger SGs that will be installed in the Fall 2003 refueling outage along with the PUR. In all cases, operators are to be trained on the changes before operation at the higher power levels of the PUR. Administrative control procedures provide for such training.

For Item 5, in PURLR Sections 9.12.3 and 9.12.4, the licensee states that because the simulator is modeled after Unit 1, a separate software model will be developed for the Unit 2 changes to plant responses to transient and accident scenarios because of the PUR and larger SGs to support the licensed operator training for Unit 2. The licensee states that this new model will not replace the Unit 1 model for normal examination and evaluation, and that the simulator fidelity will not be affected. The operators for Unit 2 are to be trained on the modifications, Technical Specification changes, procedural changes, and the changes in the Unit 2 response to transients and accident scenarios because of the PUR.

In previous sections of this SE, which are listed below, there have been references to operator actions:

- 4.3.1, "Increase in Heat Removal by Secondary System," states that the NRC review covers operator actions.

- 4.3.1.3, "Steam System Piping Failures Inside and Outside Containment," states that the NRC review covers operator actions.

- 4.3.2.4, "Feedwater System Pipe Breaks," states that the NRC review covers operator actions.

- 4.3.6.2, "Steam Generator Tube Rupture," refers to the operator being able to diagnose the tube rupture and trip the plant manually before reaching the reactor trip point.

- 4.3.6.2.1, "Steam Generator Tube Rupture with Concurrent Loss of Offsite Power (No Stuck Open PORV)," states that the EOPs include explicit instructions to guide the operators to a reactor cool-down.

- 4.4.8, "SGTR with LOP and Single Failure of ADV," states that it is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs and divert FW to the affected SG so as to maintain the SG tubes covered and thereby reduce radioactivity releases.

- 4.4.9, "SGTR with LOP," states that it is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs.

In PURLR Section 9.12.2, the licensee stated that an assessment of the expected plant response indicated that minor EOP/AOP changes are expected which would not affect operator actions or mitigation strategies that are taken credit for in accident analyses. Based on this, the staff concludes that the licensee has properly addressed human factors considerations for the PUR for the proposed amendment.

8.0 **STATE CONSULTATION**

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendment. The State official had no comments.

9.0 **CONCLUSION**

In summary, the licensee has proposed changes to its operating license and technical specifications to support the replacement of SGs and subsequent operation at an increased maximum power level of 3990 MWt for PVNGS Units 1 and 3. These proposed PURs will be implemented to Unit 1 after operating cycle 12, scheduled for Fall 2005 and to Unit 3 after operating cycle 13, scheduled for Fall 2007. In this SE, the staff evaluated the analyses submitted by the licensee to verify that results are acceptable and demonstrate compliance to applicable design basis acceptance criteria during the PUR and RSGs conditions.

The NRC staff has concluded, based on the considerations discussed throughout this SE, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in

compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

10.0 REFERENCES

1. CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

   CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

   CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

   CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

2. FLOOD3, Updated version of the NRC approved FLOOD MOD2 computer code.

3. Combustion Engineering Nuclear Power LLC. Topical Report CENPD-140-A, dated June 1976, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis."

4. Bechtel Power Corporation, "Containment Pressure and Temperature Transient Analysis (COPATTA)," as described in Bechtel Topical Report BN-TOP-3, Revision 4, "Performance and Sizing of Dry Pressure Containments."

5. USNRC Bulletin Notice No. 80-04, dated, 2/8/1980, "Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition."

6. USNRC Information Notice No. 84-90, "Main Steam Line Break Effect on Environmental Qualification of Equipment," dated December 7, 1984.

7. NUREG-0588, Interim Staff Position on Environmental Qualification of Safety- Related Electrical Equipment, dated December 1979

8. Combustion Engineering letter DP-456, F. M. Stern to E. Case, dated August 19, 1974, Chapter 6, Appendix 6B to CESSAR System 80 PSAR.

9. ABB-CE Software Verification and Validation Report #00000-AS95-CC-010, Revision 0, Computer code SGNIII, dated December 7, 1995.

10. PCFLUD Computer Program Version 5.0 SQA Classification B, Bechtel Corporation.

11. Letter 102-03578-WLS/AKK/GAM, W.L. Stewart (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN-50-528/529/530, Proposed Amendments to Facility Operating Licenses and to Technical Specifications and Various Bases, Related to Power Uprate," January 5,1996.

12. CENPD-282-P-A, “Technical Manual for the CENTS Code,” Volumes 1-3, approved in a letter from M.J. Virgilio, USNRC, to S.A. Toelle, ABB Combustion Engineering, dated March 17, 1994.

13. Letter from C. M. Trammell, USNRC, to W. F. Conway, APS, “Approval of Reload Analysis Methodology Report - Palo Verde Nuclear Generating Station (TAC Nos. M85153, M85154, and M85155),” dated June 14, 1993.

14. Letter from J. Donohew, USNRC, to G. R. Overbeck, 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.

15. CENPD-135P, “STRIKIN-II - A Cylindrical Geometry Fuel Rod Fuel Rod Heat Transfer Program.”

16. CENPD-161-P-A, “TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core,” April 1986

17. CENPD-162-P-A, “ Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution,” September 1976.

18. CENPD-207-P-A, “ Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distribution,” December 1984.

19. CENPD-356(V)-P-A, “Modified Statistical Combination of Uncertainties,” Rev. 01-P-A, May 1988.

Attachment: Table 1
List of Abbreviations

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Date:

TABLE 1

NRC STAFF'S ANALYSIS ASSUMPTIONS FOR ACCIDENT DOSE CALCULATIONS

Assumptions Common to One or More Analyses

Reactor power, 4070 MWt (includes 2 percent uncertainty)

Source Term	Core 4070 MWt *Ci*	RCS 1% F.F *uCi/gm*
Kr-83m	1.69E7	0.013
Kr85m	5.28E7	1.3
Kr-85	1.79E6	6.1
Kr-87	8.77E7	1.0
Kr-88	1.30E8	2.8
Kr-89	1.69E8	0.076
Xe-131m	1.06E6	5.9
Xe-133m	5.63E6	0.34
Xe-133	2.29E8	360.0
Xe-135m	7.39E7	0.74
Xe-135	2.18E8	7.7
Xe-137	2.17E8	0.17
Xe-138	2.02E8	0.63
I-131	1.02E8	3.0
I-132	1.55E8	0.83
I-133	2.29E8	4.4
I-134	2.68E8	0.52
I-135	2.08E8	2.5

RCS mass, lbm	560,000
Initial RCS specific activity, μCi/gm dose equivalent I-131	
CEA Ejection, LBLOCA, SBLOCA, DBLLOCUS, FSAR Chapter 15.7	3.6
Other	1.0
Initial secondary specific activity, μCi/gm dose equivalent I-131	0.1
RCS to secondary leak rate @SG, gal/min	0.5
Dose conversion factors	ICRP30 / RG1.109
Offsite breathing rate, m^3/sec	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
24-720 hours	2.32E-4
Iodine spike appearance rate parameters	
Filtration efficiency fraction	1.0
Letdown flow, gpm	150
RCS initial activity, uCi/gm d.e. I-131	1.0

RCS leakage, gpm	1.0
Iodine spike duration, hrs	8
Control room volume, ft^3	1.61E5
Normal ventilation makeup flow, cfm	1200
Essential HVAC system	
Filtered air makeup, cfm	1000
Filtered recirculation, cfm	25740
Unfiltered inleakage, cfm	63
Filter efficiency, elemental, %	95
Filter efficiency, organic, %	95
Filter efficiency, particulate, %	95
Control room breathing rate, m^3/sec 3.47E-4	
Control room occupancy factors	
0-24 hours 1.0	
1-4 days 0.6	
4-30 days 0.4	
Limiting control room χ/Q (includes occupancy factors), sec/m^3	
0-8 hrs	1.56E-3
8-24 hrs	1.08E-3
1-4 days	4.15E-4
4-30 days	1.03E-4
Offsite χ/Q, sec/m^3	
EAB: 0-2 hr	2.3E-4
LPZ: 0-8 hr	6.4E-5
8-24 hr	4.8E-5
24-96 hr	2.6E-5
96-720 hr	1.1E-5
Essential ESF filtration units efficiency, %	
Elemental	95
Organic	95
Particulate	95

Assumptions for LBLOCA Analyses

Core release fractions	
Noble gases	1.0
Iodines	0.25

Iodine species fraction

	Atmosphere	Sump
Particulate/aerosol	0.05	0.0
Elemental	0.91	1.0
Organic	0.04	0.0

Time to CMNT isolation signal (CIAS), sec	12
Control room switchover from normal to emergency mode after CIAS, seconds	50

Containment Purge Pathway

Containment purge duration, sec	12
RCS activity released during purge period, %	100
Initial RCS specific activity	
Iodine, µCi/gm dose equivalent I-131	60.0
Noble gases	PUR Section 7.6.2 Source term
Containment purge flow rate, cfm	
0.00-0.01	39,000
0.01-1.00	35,990
1.00-4:00	37,490
4.00-5.00	33,830
5.00-6.00	26,660
6.00-7.00	19,900
7:00-8:00	13,830
8:00-9:00	8,652
9:00-10.00	4,585
10:00-11:00	1785
11:00-12:00	359

Containment Leakage Pathway

Containment volume, ft^3	
Main sprayed region	2.27E6
Auxiliary sprayed region	2.0E5
Unsprayed region	1.5E5
Total	2.62E6
Containment release, %/day	
0-24 hours	0.1
24-720 hours	0.05
Containment release via depressurized secondary, scfm	0.9
Duration of release, days	30
Containment mixing flow, unsprayed volume change per hour	3.3 (8250 cfm)

Containment air transfer rates, cfm		
Main sprayed to unsprayed		7582
Auxiliary sprayed to sprayed		668
Containment spray lambda, hr^{-1}	Main	Aux
Elemental	19.6	6.05
Organic	0.0	0.0
Particulate	0.32	0.09
Containment spray DF		
Elemental iodine		6.51
Plateout		93.4
Containment elemental iodine plateout lambda, hr^{-1}		
Main sprayed region		2.14
Auxiliary sprayed region		14.4
Unsprayed region		14.4
Containment spray timings		
Injection spray initiation, sec		92
Injection spray duration, sec		386
Particulate spray reduction, sec		478
Elemental spray cutoff, sec		478

ECCS Leakage Pathway

ECCS leak rate, ml/hr	3000
Start of ECCS leakage, minutes	20
Duration of release, days	30
Containment sump volume, ft^3	7.0E4
Fraction of core iodine inventory in sump	0.5
Iodine flash fraction	0.1

Refueling Water Tank Backleakage Pathway

RWT volume, scf	1.15E5
Fuel building volume, scf	7.45E5
Maximum backleakage from SI to RWT, gpm	43
Partition coefficient of iodine in sump water backleakage	1000

Assumptions for Small Break LOCA Analyses
(* depends on break size, 0.02 ft^2 break data shown))

Fraction core inventory in gap	0.1
Fraction of gap released	1.0

Iodine species fraction

	Atmosphere	Sump
Particulate/aerosol	0.05	0.0
Elemental	0.91	1.0
Organic	0.04	0.0

Time to CIAS signal, sec	129*
Control room switchover from normal to emergency mode after CIAS, seconds	50

Containment Purge Pathway

Containment purge duration, sec	138*
Containment purge release, ft^3	153,000
Source term	60 uCi/gm Dose Equivalent I-131 Noble gases from table above

Containment Leakage Pathway

Source term	100% gap activity plus Initial RCS activity

Fraction of discharged RCS activity available for release

Noble gases	1.0
Iodines	0.25

Containment volume, ft^3

Main sprayed region	2.27E6
Auxiliary sprayed region	2.0E5
Unsprayed region	1.5E5
Total	2.62E6

Containment release, %/day

0-24 hours	0.1
24-720 hours	0.05

Containment mixing flow, unsprayed volume change per hour	3.3 (8250 cfm)
Duration of release, days	30

Containment air transfer rates, cfm

Main sprayed to unsprayed	758
Auxiliary sprayed to sprayed	668

Containment spray lambda, hr^{-1}	Main	Aux
Elemental	19.6	6.05
Organic	0.0	0.0
Particulate	0.32	0.09
Containment elemental iodine plateout lambda, hr^{-1}		
Main sprayed region		2.14
Auxiliary sprayed region		14.4
Unsprayed region		14.4
Containment spray timings (CSAS=1087 sec)		
Injection spray initiation, sec		33+CSAS
Particulate spray duration, sec		386
Elemental spray duration, sec		386

Primary-to-Secondary Leakage Pathway

Primary-to-secondary leak rate @SG, gpm	0.5
RCS source term	100% gap activity plus Initial RCS activity
Release duration, hours	3
Steam generator partition coefficient	0.01

Initial SG Release Activity Pathway

Secondary source term, μCi/gm dose equivalent I-131	0.1
Total mass release via MSSVs and ADVs, lbm	334,000
Steam generator partition coefficient	1.0

ECCS Leakage Pathway

ECCS leak rate, ml/hr	3000
Start of ECCS leakage, minutes	20
Duration of release, days	30
Containment sump volume, ft^3	7.0E4
Fraction of RCS inventory in sump	0.5
Iodine flash fraction	0.1
Essential ESF Filtration Units Efficiency, %	
Elemental	95
Organic	95
Particulate	95

Assumptions for Control Element Assembly Ejection Accident Analyses

Time to CIAS signal, sec	69
Control room switchover from normal to emergency mode after CIAS, seconds	50
Containment Purge Pathway	
Containment purge duration, sec	77
RCS activity released during purge period, %	100
Initial RCS specific activity	See table above
Containment Leakage Pathway	
Source term	
Radial peaking factor	1.77
Fraction of rods that exceed DNB	0.17
Gap fraction, all nuclide groups	0.10
Fraction of rods that exceed DNB that experience melt	0.0
Containment volume, ft^3	2.62E6
Containment release, %/day	
0-24 hours	0.1
24-720 hours	0.05
Duration of release, days	30
Primary-to-Secondary Leakage Pathway	
Primary-to-secondary leak rate @SG, gpm	0.5
Steam release via MSSV, ADVs, lbm	1,176,800
Release duration, hours	2.67
Steam generator partition coefficient	0.01
ECCS Leakage Pathway	
ECCS leak rate, ml/hr	3000
Start of ECCS leakage, minutes	20
Duration of release, days	30
Containment sump volume, gal	522,000
Fraction of RCS inventory in sump	0.5
Iodine flash fraction	0.1

Assumptions for SGTR with LOP and Single Failure Analyses

(** data interpreted from licensee-provided graphs)

Initial RCS activity		1.0 μCi/gm dose equivalent I-131
Initial secondary activity		0.1 μCi/gm dose equivalent I-131
Pre-incident iodine spike activity		60.0 μCi/gm dose equivalent I-131
Co-incident spike multiplier		500
Iodine appearance rates, Ci/hr		
I-131		13624
I-132		12005
I-133		24341
I-134		16000
I-135		19790
Iodine spike duration, hrs		8
Event timing, sec		
Reactor trip		100
LOP		103
Time to SIAS signal, sec		245
Control room switchover from normal to emergency mode after CIAS, seconds		50
Break flow flash fraction**		
0-2400 sec		1.0
2400 sec-8 hrs		0.05
Break flow to affected SG**, lbm/sec		
0-60 s		60
60-360 s		46.5
360-1080 s		53.5
1080-3000 s		63
3000-4200 s		57
4200-5760 s		48
5760-7200 s		40.5
7200-12000 s		36
12000-26400 s		31.5
26400-28800 s		30
Primary-to-secondary leakage to unaffected SG, gpm		1.0
Steam generator mass** @, lbm	Affected	Unaffected
0-60 s	100,000	100,000
60-360 s	70,000	100,000
360-1080 s	55,000	128,000
1080-3000 s	170,000	185,000
3000-4200 s	300,000	262,000
4200-28800 s	300,000	303,000
Steam release from ruptured SG[2], lbm		
0-2 hours		550,000
2-8 hours		775,000

Steam release from unaffected SGs**, lbm	
0-2 hours	25000
2-8 hours	50000
Steam partition coefficient	0.01
Main condenser DF (prior to LOP)	100

Assumptions for SGTR with LOP

(** data interpreted from licensee-provided graphs)

Initial RCS activity		1.0 μCi/gm dose equivalent I-131
Initial secondary activity		0.1 μCi/gm dose equivalent I-131
Pre-incident iodine spike activity		60.0 μCi/gm dose equivalent I-131
Co-incident spike multiplier		500
Iodine appearance rates, Ci/hr		
I-131		13624
I-132		12005
I-133		24341
I-134		16000
I-135		19790
Iodine spike duration, hrs		8
Event timing, sec		
Reactor trip		760
LOP		763
Time to SIAS signal, sec		780
Control room switchover from normal to emergency mode after CIAS, seconds		50
Break flow flash fraction**		
0-740 s		0.11
740-2700 s		0.035
2700-3000 s		0
3000-3600 s		0.027
3600-5400 s		0.025
5400-6600 s		0.01
6600 s - end		0.00
Break flow to affected SG**, lbm/sec		
0-50 s		60
50-400 s		49.5
400-740 s		45.7
740-1000 s		36
1000-1200 s		40.5
1200-2700 s		42
2700-3000 s		42
3000-3600 s		43.5
3600-5400 s		47.5
5400-6600 s		42.0
6600-7200 s		33
7200-14400 s		15
14400-28800 s		7.5
Primary-to-secondary leakage to unaffected SG, gpm		1.0
Steam generator mass** @, lbm	Affected	Unaffected
0-50 s	115,000	110,000

50-400 s	125,000	110,000
400-740 s	145,000	110,000
740-1000 s	125,000	85,000
1000-1200 s	125,000	85,000
1200-2700 s	185,000	125,000
2700-3000 s	280,000	165,000
3000-3600 s	315,000	210,000
3600-5400 s	330,000	300,000
5400-6600 s	350,000	320,000
6600-7200 s	380,000	300,000
7200-14400 s	450,000	300,000
14400-28800 s	570,000	300,000
Steam release from ruptured SG**, lbm		
0-90 minutes		135,000
Steam release from unaffected SGs**, lbm		
0-2 hours		160,000
2-8 hours		1,015,000
Steam partition coefficient		0.01
Main condenser DF (prior to LOP)		100

Assumptions for Inadvertent ADV Opening Analysis

Initial RCS activity	1.0 μCi/gm dose equivalent I-131
Initial secondary activity	0.1 μCi/gm dose equivalent I-131
Fuel clad damage fraction	0.055
Gap fraction inventory, all nuclides	0.1
Release duration, sec	
Affected SG	1800
Unaffected SG	28800
Primary-to-secondary leak rate to @SG, gpm	0.5
Release holdup	
Affected SG	none
Unaffected SG	yes
Steam partition coefficient	
Affected SG	1.0
Unaffected SG	0.01
Steam generator mass @, lbm	180,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam release from affected SG (0-30min), lbm	180,000

Assumptions for MSLB Analyses

Initial RCS activity (1.0% F.F)	3.6 μCi/gm dose equivalent I-131
Initial secondary activity	0.1 μCi/gm dose equivalent I-131
Pre-incident iodine spike activity	60.0 μCi/gm dose equivalent I-131
Co-incident spike multiplier	500
Iodine appearance rates, Ci/hr	
I-131	13624
I-132	12005
I-133	24341
I-134	16000
I-135	19790
Iodine spike duration, hrs	8
Faulted SG blowdown (100%) duration, minutes	30
Primary-to-secondary leakage @SG, gpm	0.5
Primary to secondary leakage duration, days	30
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000
Steam release from faulted SG, lbm	300,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam partition coefficient	
Affected SG	1.0
Unaffected SG	0.01
Time to SIAS signal, sec	49
Control room switchover from normal to emergency mode after CIAS, seconds	50

Assumptions for FWLB Analyses

Initial RCS activity	1.0 μCi/gm dose equivalent I-131
Initial secondary activity	0.1 μCi/gm dose equivalent I-131
Primary-to-secondary leakage @SG, gpm	0.5
Primary to secondary leakage duration, hrs	8
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000
Steam release from faulted SG (8-hours), lbm	300,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam partition coefficient	
Affected SG	1.0
Unaffected SG <30 minutes	1.0
Unaffected SG >30 minutes	0.01
Time to SIAS signal, sec	28.4
Control room switchover from normal to emergency mode after CIAS, seconds	50

Assumptions for Sheared RCP Shaft Analysis

Initial RCS activity	1.0 μCi/gm dose equivalent I-131
Initial secondary activity	0.1 μCi/gm dose equivalent I-131
Fuel clad damage fraction	0.17
Peaking Factor	1.72
Gap fraction inventory, all nuclides	0.1
RCS mass, lbm	560,000
Primary-to-secondary leak rate to @SG, gpm	0.5
Release duration, hrs	8
ADV sticks open on affected SG, sec	1800
Time to restore affected SG level, sec	5400
Steam partition coefficient	
Affected SG (<90 minutes)	1.0
Affected SG (>90 minutes)	0.01
Unaffected SG	0.01
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
0-8 hours	2,550,000
Steam release from affected SG(0-90 min), lbm	180,000
Time to SIAS signal, sec	351
Control room switchover from normal to emergency mode after CIAS, seconds	50

Assumptions for AOO Analysis

Initial RCS activity	1.0 μCi/gm dose equivalent I-131
Initial secondary activity	0.1 μCi/gm dose equivalent I-131
Fuel clad damage fraction	0.10
Peaking Factor	1.72
Gap fraction inventory, all nuclides	0.1
RCS mass, lbm	560,000
Primary-to-secondary leak rate to @SG, gpm	0.5
Release duration, hrs	8
Steam partition coefficient	0.01
Steam generator mass @, lbm	180,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
0-8 hours	2,550,000

ACRONYMS

ACRONYM	DEFINITION
AAC	Alternative Alternating Current
ABVS	Auxiliary Building Ventilation System
AC	Alternating Current
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
ANO	Arkansas Nuclear One
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating procedure
AOR	Analysis of Record
APS	Arizona Public Service
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BSAP	Bechtel Structural Analysis Program
BTP	Branch Technical Position
CBHVACS	Control Building Heating, Ventilation, and Air Conditioning System
CE	Combustion Engineering
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CENTS	Combustion Engineering Nuclear Transient Simulator
CEOG	Combustion Engineering Owners Group
CESSAR	Combustion Engineering Standard Safety Analysis Report
CFR	Code of Federal Regulations
CFWS	Condensate and Feedwater System
CFWS	Condensate and Feedwater System
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
CHVACS	Containment Heating, Ventilation, and Air Conditioning Systems
CPC	Core Protection Computer
CSS	Containment Spray System
CST	Condensate Storage Tank
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Event
DBLLOCUS	Double-Ended Break of the Letdown Line Outside Containment Upstream of the letdown line control valve
DC	Direct current
DEDLSB	Double-Ended Discharge Leg Slot Break
DEHLSB	Double-Ended Hot Leg Slot Break
DESLSB	Double-Ended Suction Leg Slot Break
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling

DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECWS	Essential Cooling Water System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Year
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESPS	Essential Spray Pond System
FAC	Flow Accelerated Corrosion
FHA	Fuel Handling Accidents
FIV	Flow-induced Vibration
FTC	Fuel Temperature Coefficient
FW	Feedwater
FWIV	Feedwater Isolation Valve
FWLB	Feedwater Line Break
GDC	General Design Criterion
GL	Generic Letter
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HJTC	Heated Junction Thermocouple
HLR	Head Lift Rig
HPPT	High Pressurizer Pressure Trip
HPSI	High Pressure Safety Injection
HVAC	Heating Ventilation and Air Conditioning
I&C	Instrumentation & Controls
ICI	In-Core Instrumentation
ID	Inadvertent Deboration
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
LBB	Leak Before Break
LBLOCA	Large Break Loss-of-Coolant Accident
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
LPZ	Low Population Zone
LTC	Long-term Cooling
LTOP	Low Temperature Over Pressure
M&E	Mass and Energy
MCL	Main Coolant Loop
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSIVBV	Main Steam Isolation Valve Bypass Valve
MSLB	Main Steam Line Break
MSSS	Main Steam Supply System

MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MWt	Megawatts Thermal
NCWS	Nuclear Cooling Water System
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
OSG	Original Steam Generator
P_a	Peak Calculated Containment Internal Pressure for the Design Basis Loss of Coolant Accident
PCT	Peak Cladding Temperature
PCWS	Plant Cooling Water System
PLHR	Peak Linear Heat Rate
PMF	Probable Maximum Flood
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTS	Pressurized Thermal Shock
PUR	Power Uprate
PURLR	Power Uprate Licensing Report
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
ROPM	Required Over Power Margin
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSGs	Replacement Steam Generator
RTP	Rated Thermal Power
RT_{NDT}	Nil Ductility Temperature
RVI	Reactor Vessel Internals
RVID	Reactor Vessel Integrity Database
SAFDL	Specified Acceptable Fuel Design Limit
SBCS	Steam Bypass Control System
SBLOCA	Small Break Loss-of-Coolant Accident
SBO	Station Blackout
scfm	Standard Cubic Feet per Minute
SCU	Statistical Combination of Uncertainties
SE	Safety Evaluation
SFP	Spent Fuel Pool
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank
SRP	Standard Review Plan

SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
TAV	Turbine Admission Valve
T_{ave}	average reactor coolant temperature
T_{hot}	hot leg reactor coolant temperature
TBHVACS	Turbine Building Heating, Ventilation, and Air Conditioning System
TCWS	Turbine Cooling Water System
TS	Technical Specifications
TBV	Turbine Bypass Valve
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
UHS	Ultimate Heat Sink
VOPT	Variable Over-Power Trip