APS

Palo Verde Nuclear
Generating Station

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102-04899-CDM/TNW/RAB
March 11, 2003

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

Reference:
1. Letter No. 102-04641-CDM/RAB, dated December 21, 2001, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations"
2. Letter, dated June 14, 2002, from J. N. Donohew, USNRC, to G. R. Overbeck, "Palo Verde Nuclear Generating Station, Unit 2 – Request For Additional Information Regarding Power Uprate License Amendment Request (TAC No. MB3696)"
3. Letter No. 102-04847-CDM/TNW/RAB, dated October 11, 2002, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request"

Dear Sirs:

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

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

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During the NRC review of Reference 3, which provided responses to the Reactor Systems Branch questions in Reference 2, the Staff determined that several responses needed clarification and requested a phone call, which was held on January 16, 2003. Attachment 2 to this letter documents the responses to all of the clarifications requested by Reactor Systems Branch.

APS requests the opportunity to review the draft Safety Evaluation Report (SER) for the technical discussion in the SER. APS understands that the staff currently expects to complete the draft SER by late March or early April, 2003. APS will return comments to the NRC within six weeks, approximately early to mid May and, thus agrees that the final SER may be issued as late as June 30, 2003.

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

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

Sincerely,

David Mauldin

CDM/TNW/RAB

Attachments:

1. Notarized Affidavit
2. Clarification of Responses to the Request for Additional Information from the Reactor Systems Branch

| | | |
|---|---|---|
| cc: | E. W. Merschoff | (NRC Region IV) |
| | J. N. Donohew | (NRC Project Manager) |
| | Bo Pham | (NRC Project Manager) |
| | N. L. Salgado | (PVNGS) |
| | A. V. Godwin | (ARRA) |

# ATTACHMENT 1

## Notarized Affidavit

Attachment 1

STATE OF ARIZONA )
) ss.
COUNTY OF MARICOPA )

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 11 Day Of March, 2003.

Cassandre Justiss

Notary Public

OFFICAL SEAL
Cassandre Justiss
NOTARY PUBLIC - STATE of ARIZONA
MARICOPA COUNTY
MY COMM EXPIRES October 30, 2006

Notary Commission Stamp

## ATTACHMENT 2

## Clarification of Responses to Request for Additional Information from the Reactor Systems Branch

Clarification of Responses to Request for Additional Information

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

Question 18 of Reference [3] - The licensee states that core average gap conductance values are used in the analyses. Does use of a core average value rather than an absolute maximum or minimum ensure a conservative result? How do the absolute maximum and minimum gap conductance values compare to the core average maximum and minimum values.

APS Response:

The determination, selection, and use of core gap conductance values described in the Power Uprate License Report (PURLR) provided in Reference 1 are consistent with the existing licensing basis described in the UFSAR. The methods used to determine the values are described in the topical reports provided as references to the response to Question 18a of Reference 3.

Use of an absolute maximum or minimum gap conductance value ensures a conservative result, and the following paragraph describes how this conservatism is maintained in the analyses:

The core "average" gap conductance values are the maximum or minimum gap conductance values that are determined at a specific core "average" power and core "average" burnup. As stated in the response to Question 18b of Reference 3, the core "average" power level used to derive the corresponding gas gap conductance value encompasses the core power levels observed during the entire transient, not just the core average power at the beginning of the transient. It is overly conservative to use a constant maximum or minimum gap conductance applied to entire core during the entire transient. However, it simplifies the analyses that are performed to evaluate the response of the Nuclear Steam Supply System (NSSS).

The use of a dynamically calculated local gap conductance value versus a constant minimum or maximum core average gap conductance value for the events where NSSS response was evaluated was explored earlier by performing sensitivity studies. For example, the CEA Ejection - Reactor Coolant System (RCS) Peak Pressure event was evaluated by using both methods during the development of the System 80 Combustion Engineering Standard Safety Analysis Report (CESSAR). Comparison of the studies demonstrated that using a constant core average gap conductance value is more conservative than using dynamically calculated gap conductance values with respect to RCS peak pressure. For this event, RCS peak pressure was calculated to be 2757 psia using the maximum average gap conductance value, and 2707 psia using the dynamically calculated gap conductance value.

NRC Question 2:

Question 22 of Reference [3] - Please be more specific regarding the appropriate NRC-approved methodologies used to calculate the CPC and COLSS overall uncertainty penalty factors.

APS Response:

The Modified Statistical Combination of Uncertainties (MSCU) analysis derives the overall uncertainty penalty factors for Core Protection Calculator (CPC) and Core Operating Limit Supervisory System (COLSS).

The methodology for MSCU analysis is described in CEN-356(V)-P-A, Revision 01-P-A, Modified Statistical Combination of Uncertainties, dated May 1988. This methodology has been previously reviewed and approved by the NRC, as stated in the Unit 2 Core Operating Limits Report (COLR).

NRC Question 3:

Question 23.a of Reference [3] - Please identify the COLR physics methods used to determine the reactivity insertion rates, and provide more detail on the plant data used. Also, the discussion on the methodology used to determine the source neutron power and initial power fraction is not clear. Is an NRC-approved methodology used to determine the source neutron power and initial power fraction?

APS Response:

The bounding reactivity insertion rates are determined by performing a parametric study on CEA bank worth and axial power shape for various core designs. These studies utilize ROCS or SIMULATE codes. The range of bank worths and axial power shapes used in the parametric study encompassed the actual plant values observed in the previous core designs at the time of development of the method. These parameters are verified to remain bounding for each reload cycle in accordance with the approved methodology discussed below. Based on the bounding bank worth and the bounding top peak axial shape, bounding differential rod worths are determined using the HERMITE code. The bounding differential rod worth and rod speed is then input into the CENTS code to establish the reactivity insertion rate that is used in the transient.

As described in the response to Question 23.a of Reference 3, the source neutron power at subcritical conditions represents the power that corresponds to the neutron production of the fuel material present at a given $k_{eff}$. This power is represented by the textbook formula: $P = S / (1 - K)$. In this equation, the source strength, S, represents the effective power from neutron production at $k_{eff} = 0.0$ (no fissionable material considered), and K is the value of $k_{eff}$ for a given configuration. The source strength is calculated by utilizing the industry standard code ORIGEN 2. The initial power level for the transient simulation is based upon this source strength and the subcriticality imposed by the withdrawn bank, and subcritical multiplication considerations.

The methods being followed are the same as those used in the current licensing basis. This methodology is consistent with the approved Reload Analysis Methodology (NRC letter to Mr. William Conway, APS dated June 14, 1993, Approval of Reload Analysis Methodology Report – Palo Verde Nuclear Generating Station (TAC Nos. M85153, M85154, AND M85155)). As stated in the Unit 2 COLR, "Analytical Methods", these physics methods have been previously reviewed and approved by the NRC. The license amendments authorizing the COLR, and use of the analytical methods therein, were approved in NRC letter to Mr. William Conway, APS, dated December 30, 1992, "Issuance of Amendments for the Palo Verde Nuclear Generating Station Unit No. 1 (TAC No. M83092), Unit No. 2 (TAC No. M83093), and Unit No. 3 (TAC No. M83094)".

NRC Question 4:

Question 23.b of Reference [3] - For the CEAW from subcritical and HZP events, the licensee assumes a minimum β value. Does a minimum β value produce the limiting results for all acceptance criteria for these events (fuel temperature, DNBR and RCS pressure)?

APS Response:

The use of a minimum delayed neutron fraction, (β) leads to the shortest reactor period upon achieving criticality. The shorter period leads to the most adverse 'spike' in core power production. This will be the most adverse selection for all of the acceptance criteria examined for the subject events.

NRC Question 5:

Question 26 of Reference [3] - Does the licensee plan to correct the titles of the effected Figures?

APS Response:

The affected figure titles will be corrected during the incorporation of the approved Power Uprate (PUR) amendment request into the UFSAR.

NRC Question 6:

Question 27 of Reference [3] - Which Palo Verde Technical Specification ensures that the thermal margin assumed in the analyses is maintained? Has the licensee performed sensitivity studies to verify that a case starting from a higher initial core inlet temperature would result in a faster trip time, and thus, lower thermal degradation? If so, please provide results to demonstrate this.

APS Response:

Technical Specification Limiting Conditions for Operation (LCO) 3.2.4 requires that steady state operation of the PVNGS units Departure from Nucleate Boiling Ratio (DNBR) shall be maintained by one of the following methods

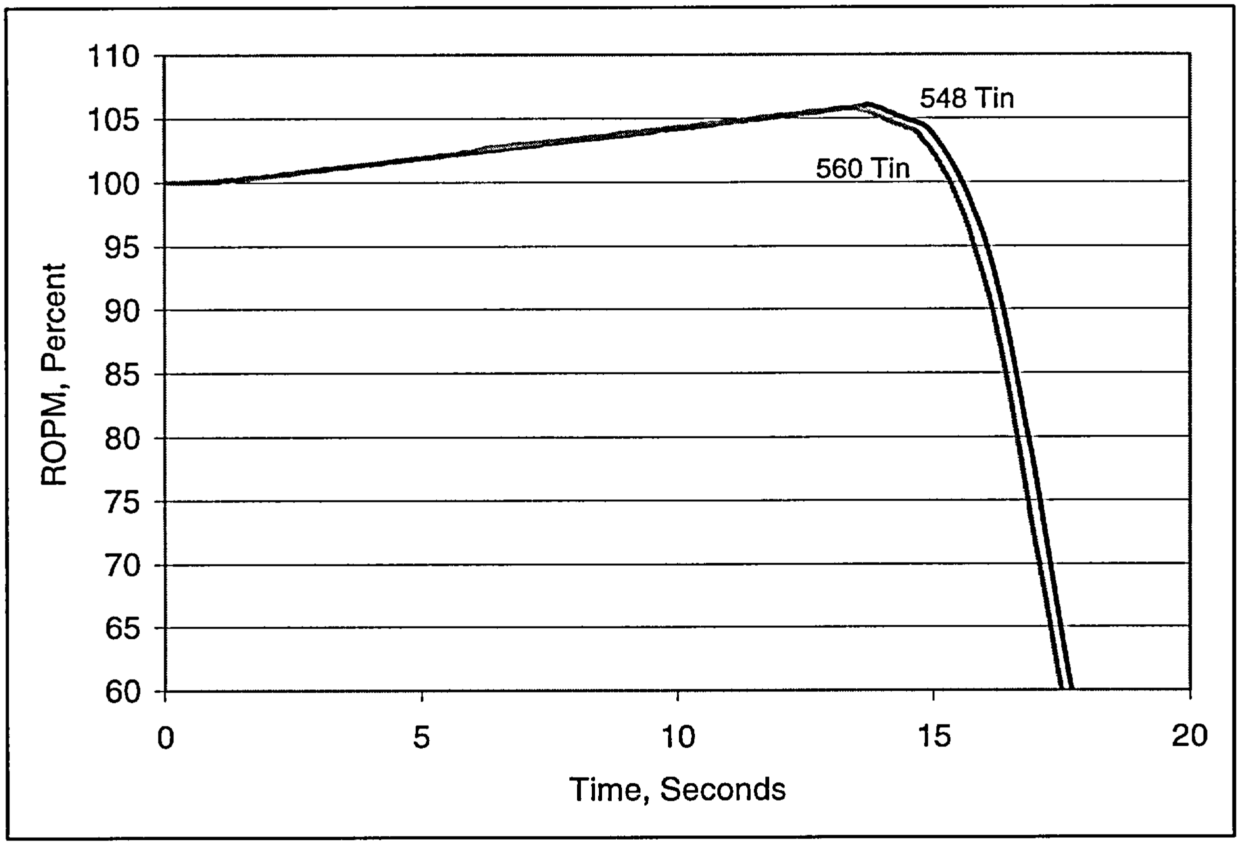
- COLSS and control element assembly calculators (CEACs) Operable
- COLSS Operable/CEACs Inoperable
- COLSS Out-of Service/CEACs Operable
- COLSS Out-of Service/CEACs Inoperable

These configurations are addressed by LCO 3.2.4.a through 3.2.4.d, respectively.

Thermal margin degradation during any given transient is calculated in terms of core power and is called the Required Over Power Margin (ROPM). The limiting transient ROPM is incorporated into the monitoring system, thus preserving sufficient thermal margin to meet the acceptance criteria of all design basis events.

The reactor trip responding to the withdrawal of the high worth CEA banks from high initial power conditions is the CPC Variable Overpower Trip (VOPT) function. The important inputs to VOPT are the neutron power signals and the change in, not initial value of, core inlet temperature. Figure 1 shows the ROPM resulting from initiating the event at 548 °F and 560 °F. As shown in the figure, the actual value of initial temperature does not affect the trip system as a result of the transient, and the ROPM is essentially unaffected by the selection of initial core inlet temperature.

Figure 1
100% Power CEAW ROPMs




NRC Question 7:

Question 28.b of Reference [3] - The staff would like additional clarification regarding the conversion of the predicted radial distortion factor to thermal margin units. Also, please provide a Reference to the NRC-approved methodology used.

APS Response:

A 95/95 upper bound of the CETOP calculated sensitivity of Over Power Margin (OPM) to changes in radial Peaking Factor (Fr) of the sample cases is determined. This upper bound is used to convert radial distortion factors to thermal margin (ROPM) units.

The overall methodology followed is that used in the COLSS and CPC Overall Uncertainty Analysis methodology, which was reviewed and approved by the NRC in Topical Report CEN-356(V)-P-A Revision 01-P-A, Modified Statistical Combination of Uncertainties, dated May 1988.

The method employed is the same as that used in the current licensing basis. This methodology is consistent with the approved Reload Analysis Methodology (NRC letter to APS dated June 14, 1993, "Approval of Reload Analysis Methodology Report - PVNGS (TAC Nos. M85153, M85154, AND M85155)").

NRC Question 8:

Question 29.d of Reference [3] - The licensee states that, "a loss of offsite power does not adversely affect the results of the peak RCS pressure and fuel performance cases." Please provide the technical basis for this conclusion. Is the assumption of loss of offsite power for only the radiological dose consequence case consistent with the current Palo Verde licensing basis?

APS Response:

As stated in the response to Question 29.d of Reference 3, the CEA Ejection event consists of three parts: RCS peak pressure, fuel performance, and dose consequence. Consideration of a Loss of Offsite Power (LOP) is not a requirement of the Standard Review Plan (SRP) Section 15.4 and it is not in the licensing basis for PVNGS for the RCS Peak Pressure and Fuel Performance cases. Regardless, the LOP does not adversely affect the results for the peak RCS pressure case since the peak pressure occurs relatively quickly following a high pressurizer pressure trip. A LOP occurring prior to the High Pressurizer Pressure Trip (HPPT) would result in an earlier trip on Reactor Coolant Pump (RCP) speed by the Core Protection Calculator System (CPCS) resulting in a more benign peak pressure. The effects of the timing of a later LOP was evaluated by performing a parametric study for rapid pressurization events. The study demonstrated that the no-LOP cases resulted in peak RCS pressures that were either the same as, or more adverse than, the LOP cases. Also, the fuel performance case is not affected by the LOP since the power spike occurs very early in the event and the energy rise in the fuel peaks before the effect of the reduced RCS flow, due to the LOP, becomes significant.

On the other hand, consideration of a LOP for dose consequences of a CEA Ejection event is required by Regulatory Guide 1.77. The dose consequences of a CEA Ejection event is adversely impacted due to the unavailability of the condenser and the increased steam release to the atmosphere. Therefore, a LOP is assumed to occur for the CEA Ejection radiological consequence case. This assumption is consistent with the methodology described in the UFSAR Section 15.4.8, and is explained in Section 6.4.4.1 of the PURLR (Reference 1).

NRC Question 9:

Question 30 of Reference [3] - Based on the Staff RAI, the licensee has changed from a Peak Linear Heat Rate Safety Limit to a Fuel Centerline Temperature Safety Limit. Does the licensee plan to revise Sections 6.3.4.1 of its submittal to reflect this change in the Safety Limit?

APS Response:

Section 6.3.4.1 of the PURLR (Reference 1) will not be revised, since the amendment request and the NRC Safety Evaluation Report (SER) established the correlation between the Peak Linear Heat Generation Rate (PLHGR) Safety Limit and the Fuel Centerline Temperature Safety Limit.

The UFSAR will be revised in accordance with 10 CFR 50.71 (e)

NRC Question 10:

Please provide actual procedure requirements for the time of initiation of hot side injection.

APS Response:

Emergency Operating Procedures (EOPs):

- 40EP-9EO03, Loss of Coolant Accident, and
- 40EP-9EO09, Functional Recovery, Appendix HR-2

instruct the operators as follows:

> "**IF** the elapsed time from the start of the LOCA is between 2 and 3 hours, **AND ANY** of the following conditions exist:
>
> - RCS subcooling is less than 24°F [40°F], based on Representative Core Exit Thermocouple (REP CET) temperature.
> - Pressurizer level is less than 10% [15%].
> - Reactor Vessel Level Monitoring System (RVLMS) indicates that the Reactor Vessel Upper Head (RVUH) level is less than 16%.
>
> **THEN** PERFORM Appendix 100, Hot Leg Injection."

The procedure 40EP-9EO10, Standard Appendices, Appendix 100, "Hot Leg Injection" provides instructions for establishing hot leg injection.

The EOPs are consistent with the sequence of events described in UFSAR Sections 6.3.2.7 and 6.3.3.

NRC Question 11:

Question 19 of Reference [3] – The staff requested additional information regarding Sections 7.1 and 7.2 of the Palo Verde submittal. The RAI response included information addressing only Section 7.1. Please provide information regarding Section 7.2, "Core Design." Is Section 7.2 intended to address SRP Section 4.3, "Reactor Design?" Where is SRP Section 4.3 addressed in the submittal?

APS Response:

The PURLR sections and their corresponding SRP and UFSAR Sections are summarized in the table below:

| PURLR Section | SRP Section | UFSAR Section |
|---|---|---|
| Section 7.1 | Section 4.4 | Section 4.4 |
| Section 7.2 | Section 4.3 | Section 4.3 |
| Section 7.3 | Section 4.2 | Section 4.2 |

With respect to Section 7.2, the following paragraph is provided to complete the response to Question 19 of Reference 1:

A sample set of assumptions and parameters of importance for the core design that have been used and observed for the existing plant configuration (i.e. 3876 $MW_t$) are presented in the PURLR Table 7-1. Comparing these parameters to the projected parameters for PUR core designs, which are also presented in the table, shows that the existing parameters and assumptions will remain within the code limitations and restrictions. These parameters are verified to remain within the code restrictions and limitations for each specific reload cycle. APS 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.

PURLR Section 7.2 states that core design methodology will be consistent with the existing approved Reload Analysis Methodology (NRC letter to APS dated June 14, 1993, "Approval of Reload Analysis Methodology Report - PVNGS (TAC Nos. M85153, M85154, AND M85155)").

NRC Question 12:

Is Figure 6.3-18 correct? The submittal states that the maximum pressure stays below the 110% limit of the design pressure. However, Figure 6.3-18 indicates that in another 200 sec could exceed 110% of the design pressure. Please explain.

APS Response:

Figure 6.3-18 presents the RCS pressure behavior for Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with Loss of Offsite Power following Turbine Trip (IOSGADV + LOP) event during the first 30 minutes of the transient when no operator action is taken. Beyond 30 minutes, the operator may take action to initiate a controlled cooldown, such as manually controlling the Auxiliary Feedwater (AFW) and Atmospheric Dump Valves (ADVs). In addition, the automatic plant response, such as opening of Primary Safety Valves (PSVs) and Main Steam Safety Valves (MSSVs) provide protection against exceeding the safety limits (110% of the design pressure).

The automatic response of PSVs that limits the maximum RCS pressure to the opening setpoint pressure of the valves is demonstrated within Figure 6.3-18 for the 3876 $MW_t$ case, for which the overpressure protection is provided by the PSVs lifting at about 1700 seconds into the transient. For the 3990 $MW_t$ case, this effect is not visible, since the pressurizer pressure does not reach the PSV lift setpoint before 1800 seconds. If operators did not take action beyond 30 minutes, the PSVs would eventually lift for the uprated power case resulting in a similar RCS pressure response beyond 1800 seconds. With no operator action, the maximum pressurizer pressure will remain at or below PSV lift pressure, and RCS pressure will remain below 110% of the design pressure.

Presenting the first 30 minutes of the transient is consistent with the SRP and Regulatory Guide 1.170 requirements, and the current licensing basis presented in the UFSAR.

NRC Question 13:

Page 6-85 and elsewhere. The HERMITE code: do you use an approved version in your application? If yes, provide a reference.

APS Response:

The reference for the HERMITE code is provided in PURLR Section 9, Reference 9-42. HERMITE was approved by the NRC in letter to Mr. William Conway, APS, "Approval of Reload Analysis Methodology Report – Palo Verde Nuclear Generating Station TAC M85153, M85154, and M85155)", dated June 14, 1993.

NRC Question 14:

Figure 6.3-40, why are the MDNBR values so different for a 2.96% power increase? Please provide a physical explanation of this phenomenon.

APS Response:

PURLR Section 6.3.1.5 presents the analyses for Post-Trip Main Steam Line Break (MSLB) events that are examined to evaluate the potential for Return-to-Power (R-t-P) following a reactor trip, and subsequent DNBR degradation.

The initial reactor power level does not significantly impact the Post-Trip MSLB event. The initial plant parameters that have first-order effects on the results of the event are those parameters that affect the amount of positive reactivity insertion and subsequent R-t-P following the reactor trip. Of those parameters, the Moderator Temperature Coefficient of reactivity (MTC) has the greatest effect on the R-t-P. With a negative MTC, increased heat removal by the secondary system due to the MSLB will result in a positive reactivity insertion and may result in R-t-P after reactor trip. Therefore, the cooldown of the RCS by the secondary system is the dominant phenomena. As stated in PURLR Section 6, the safety analyses in support of the PUR are performed with Replacement Steam Generators (RSGs), which are larger than the Original Steam Generators (OSGs). The larger water mass in the RSGs results in a longer and larger heat removal by the secondary system which results in a larger positive reactivity insertion during the transient. As shown in PURLR Table 6.3-13, for the 3990 $MW_t$ case which has larger Steam Generators (SGs), Hot Full Power (HFP) MSLB with LOP results in R-t-P, while for the 3876 $MW_t$ case, the reactivity insertion is not sufficient to result in R-t-P. The power increase observed during the MSLB transient for the 3990 $MW_t$ case results in significantly larger degradation of the Departure from Nucleate Boiling (DNB) than the 3876 $MW_t$ case. This difference is observed in PURLR Figure 6.3-40.

NRC Question 15:

Section 6.3.1.5.6. The acceptance criteria for this transient included that the vessel will not be subject to brittle fracture. In the conclusions section there is no mention of the potential for brittle fracture. However, Table 6.3-16 indicates that the scenario includes a substantial cold leg injection under considerable pressure. Please address the vessel brittle fracture acceptance criterion.

APS Response:

The conclusion drawn in PURLR Section 6.3.1.5.6 is consistent with the existing licensing basis as stated in UFSAR Section 15.5.1.4. In the NRC's review and approval of the methodology, the conclusion states that the consequences of postulated Steam Line Breaks (SLBs) meet the requirements of General Design Criteria (GDC) 27, 28, 31, and 35, as documented in NUREG-0852, Combustion Engineering Standard Safety Analysis Report (CESSAR) Safety Evaluation Report, Supplement 2, Section 15.3.1 and Appendix H, dated September 1983.

The protection against reactor vessel brittle fracture is not demonstrated through explicit Chapter 15 analyses. In general, the vessel is protected against brittle fracture during severe cooldown events, such as the MSLB event, through application of the Pressurized Thermal Shock (PTS) screening criteria (refer to PURLR Section 7.5, and response to Question 17 of Reference 3). As indicated in response to Question 17, End of Life (EOL) Reference Nil Ductility Temperature ($RT_{NDT}$) for the PVNGS Unit 2 reactor vessel is 78 $^{o}$F. This is well below the PTS screening criterion of 270 $^{o}$F.

In addition to the information provided in PURLR Section 7.5, the thermal stress and brittle fracture analyses are reported in PURLR Sections 3, 4, and 5. The Low Temperature Overpressure Protection (LTOP) analysis is reported in PURLR Section 8.13.

NRC Question 16:

Figure 6.3-28, page 6-98 and 6-100. At the end of 10 minutes the pressure keeps rising yet you concluded that the pressure remains within the acceptance limits. How did you arrive in that conclusion, since the operator action is 20 minutes away?

APS Response:

The long-term NSSS response for the "increased heat removal by the secondary system" events is similar for each event in this category due to the same physical phenomena driving the transient. The initial decrease in the RCS pressure due to cooldown is reversed later in the transient due to the R-t-P and /or NSSS recovery from the excess heat removal. Thus, the long term RCS pressure response for MSLB event is similar to the IOSGADV+LOP event presented in PURLR Figures 6.3-13 through 6.3-25 and Table 6.3-8. Figure 6.3-18 illustrates the long-term RCS pressure response for the IOSGADV+LOP event and represents a typical response for any increased heat removal by the secondary system event. As demonstrated in this figure, the automatic plant response (i.e. opening of PSVs) limits the RCS pressure, and the safety limit (110% of the design pressure) is not exceeded. Extended plots of RCS pressure versus time for the 3876 $MW_t$ and 3990 $MW_t$ MSLB-HFP with LOP cases are provided on the following pages.

Figure 1:
MSLB HFP with LOP – RCS Pressure vs. Time (3876 $MW_t$)

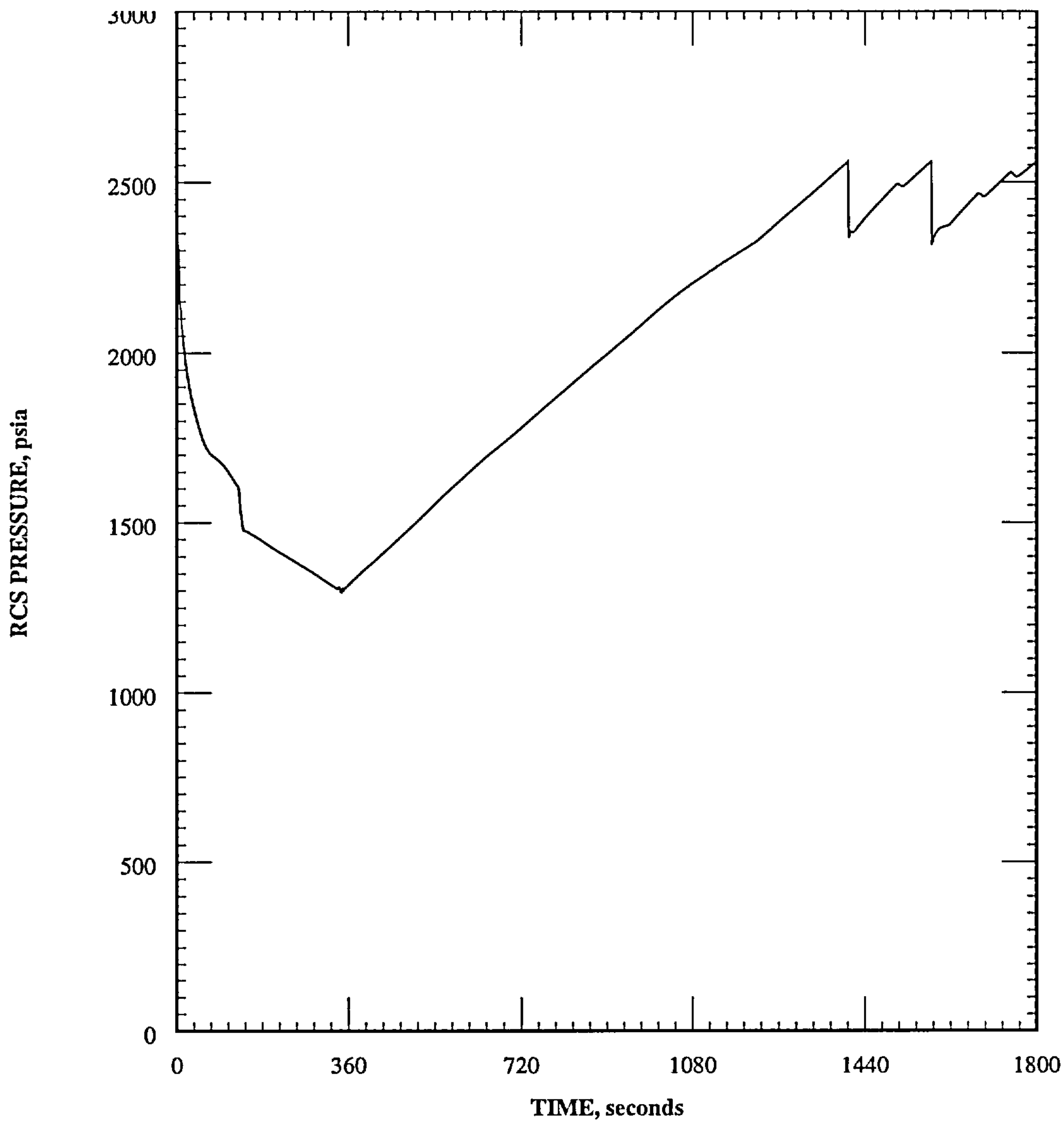
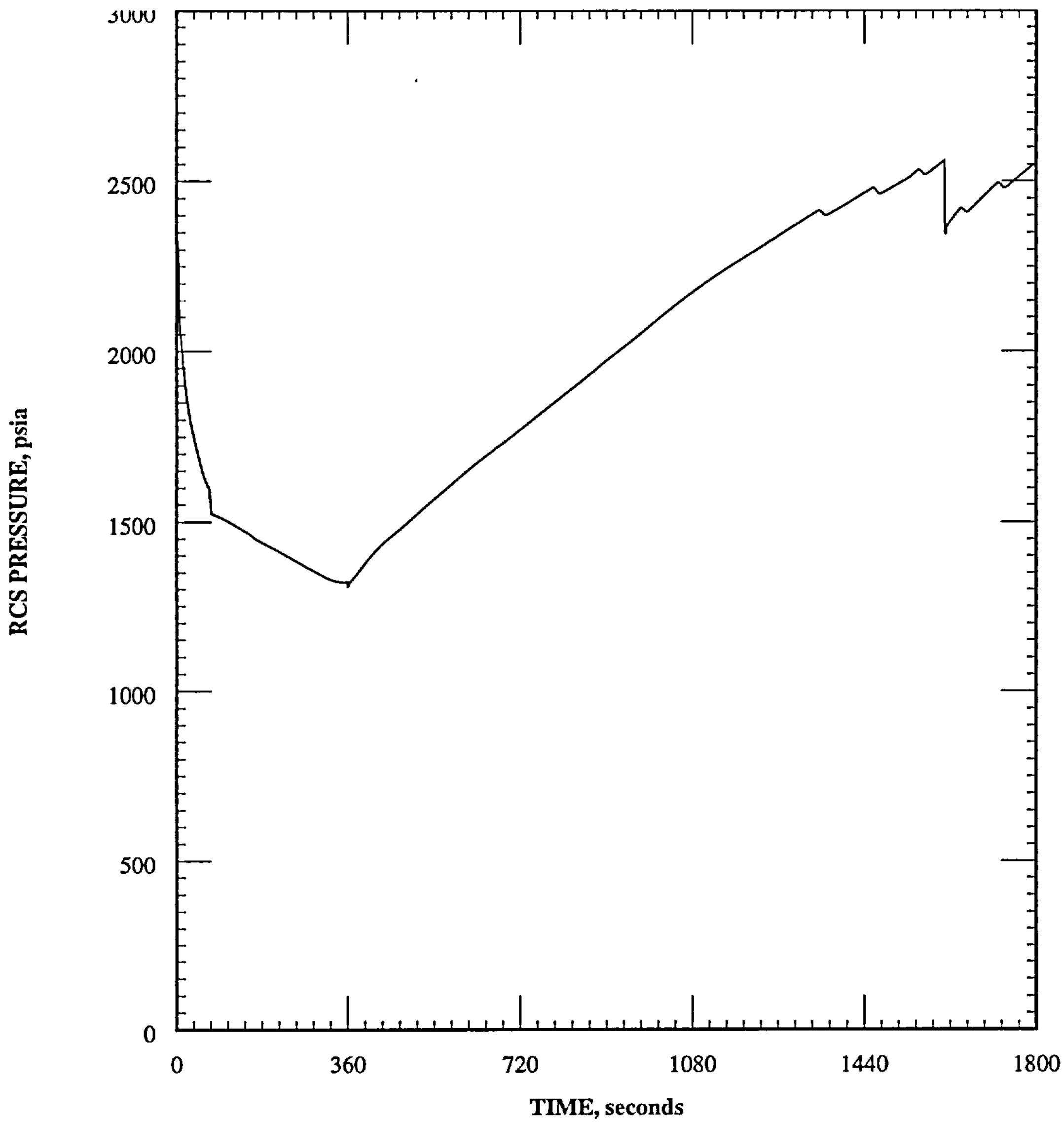


Figure 2:
MSLB HFP with LOP – RCS Pressure vs. Time (3990 $MW_t$)



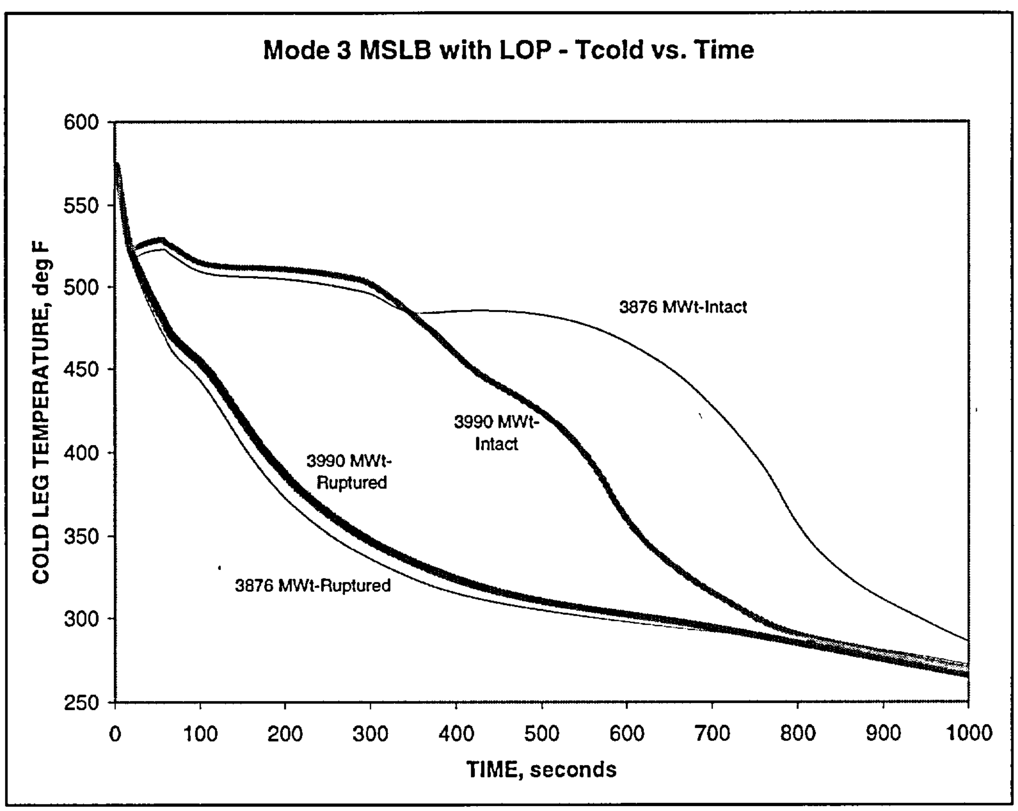
NRC Question 17:

Figures 6-42/43/44. The intact steam generator displays a large discontinuity in behavior, yet there is nothing in the scenario to justify it. Please provide a physical explanation of these phenomena.

APS Response:

The PURLR Figures 6.3-42 thru 44 illustrate the behavior of the NSSS parameters during a Mode 3 MSLB event analysis used to demonstrate the adequacy of shutdown margin. The figures demonstrate that the positive reactivity insertion following the reactor trip results in a period of power increase, which is later compensated by the shutdown margin provided.

The discontinuity in Figure 6.3-42 occurring at approximately 10 seconds into the transient is due to the isolation of the SGs upon receipt of the Main Steam Isolation Signal (MSIS). The Mode 3 MSLB analysis for the 3876 $MW_t$ case receives a MSIS at a steam pressure setpoint of 810 psia, while the 3990 $MW_t$ case receives a MSIS at 875 psia (see Attachment 3 of Reference 1 for proposed Technical Specification changes). The lower SG pressure of the 3876 $MW_t$ case results in moderator temperatures that are lower during the initial phases of the transient. The difference in R-t-P between the two cases as a result of the cold leg temperature differences is shown in PURLR Figure 6.3-42. For both cases, the difference in cold leg temperatures disappears once subcriticality is reestablished via the start of the safety injection system. The response of the cold leg temperatures on a longer simulation of the event is shown below:



NRC Question 18:

Pages 6-185/7, 3rd paragraph. You let the SG dry out to simulate the worst conditions for the RCS. Then you actuate the AFW to fill up the SG. This scenario is for the purposes of analysis. In real life do you have a procedure to refill the SG?

APS Response:

The EOPs direct the operator to isolate the affected SG and to maintain level in the unaffected SG by using the appropriate feedwater source. For this event, the affected SG would be isolated to the greatest extent possible. AFW would be used to maintain level in the unaffected SG.

In the event that both SGs are dry, the EOPs direct the operator to determine which SG is affected using various control room indications, and then isolate the affected SG. The operator is then directed to establish feedwater flow to the unaffected SG and recover and maintain level in that generator.

NRC Question 19:

Pg. 6-185 par. 6.3.2.8.1.2. The SRP has brittle fracture (GDC 31) as one of the acceptance criteria. Please address vessel brittle fracture. Same question par. 6.3.2.8.1.6.

APS Response:

Please see response to NRC Question 15.

NRC Question 20:

Pg. 6-189, Table 6.3-27, last entry. FWLB = 0.14 $ft^2$ this is a small break (by your own definition in the same section) for which the FW margin should be sufficient to keep the SG operating. How does this break size maximize RCS pressure? Intuitively one would expect that a larger break (or a DEGB) would be the most conservative. Different values of the break size appear on pages 6-190 and 6-184. What is your break optimization procedure?

APS Response:

As stated in response to Question 11 of Reference 3, for the FWLB events, a spectrum of break sizes was analyzed by performing sensitivity studies to determine the most limiting break size. The sensitivity study methodology on the break size is described in CESSAR Appendix 15B and NUREG-0852, Supplement 2, Appendix G. As explained in CESSAR Appendix 15B and UFSAR Section 15.2.8, a larger break results in an earlier Low Steam Generator Level Trip (LSGLT) which in turn results in lower RCS pressure at the time of reactor trip. This results in a more benign RCS peak pressure. On the other hand, a smaller break results in an earlier High Pressurizer Pressure Trip with more SG liquid inventory available for the heat removal at the time of trip, which also results in a lower RCS peak pressure. Therefore, the worst break size is the one that would result in simultaneous HPPT and LSGLT, thereby maximizing the RCS peak pressure.

For the Small Feedwater Line Break (SFWLB), break sizes less than or equal to 0.20 $ft^2$ are evaluated using various initial parameters. The most limiting break size for PUR was determined to be the largest break size of the spectrum, i.e. 0.2 $ft^2$. For the FWLB with a LOP and a Single Failure (SF), the spectrum of all break sizes is evaluated with various initial parameters. The largest break size used is 1.4 $ft^2$ which corresponds to DEGB of the largest size pipe. Note that other initial conditions, besides break size, have an effect on the peak RCS pressure and the maximum pressurizer level during the transient. Therefore, the peak RCS pressure and maximum pressurizer level cases are analyzed separately since the worse case parameters are not mutually conservative. Although determination of the limiting break size is the same as explained above for both cases, i.e. simultaneous HPPT and LSGLT, a conservative combination of the other initial parameters results in different limiting break sizes for the peak RCS pressure and maximum pressurizer level cases.

NRC Question 21:

Tables 6.3-33, 6.3-35 and 6.3-47 show different values for MTC. Fig. 6.3-154 shows $\Delta\rho = 0$ vs time. Where do these values come from? Is the feedback small, irrelevant or $\Delta\rho = 0$ for conservatism? Or am I reading something wrong?

APS Response:

The MTC value selected is that which provides the most adverse moderator temperature reactivity feedback for the event being analyzed. For example, for heat-up events, the most positive (the least negative) value of MTC results in the most adverse feedback, while the most negative value of MTC results in the most adverse feedback for the cooldown events. Based on the Technical Specification 3.1.4, Moderator Temperature Coefficient (MTC), and the COLR MTC limits, the most positive or most negative MTC values are selected for each transient.

For the first two transients in the question (Loss of Flow (LOF) and RCP Sheared Shaft Events), the most positive (or least negative) value of MTC is more adverse due to the local increases in core coolant temperature, which results in the greatest degradation of thermal margin. For the LOF event, a MTC value of $0.0 \times 10^{-4}$ $\Delta\rho/°F$ is selected, which is more conservative than that allowed by the COLR.

As explained in the response to Question 13 of Reference 3, the selected MTC value ($-0.18 \times 10^{-4}$ $\Delta\rho/°F$) is the COLR Beginning-Of-Cycle (BOC) MTC value at 95% power level. In terms of the transient having the potential to degrade DNBR to the point at which the DNBR Specified Acceptable Fuel Design Limits (SAFDL) is exceeded, both the full power and 95% power cases are equivalent since the same initial thermal margin is preserved between those power levels by Technical Specifications. The RCP Sheared Shaft event analysis conservatively combines the MTC at 95% power with an initial power of 100% for the limiting case. This combination of initial conditions bounds the possible thermal margin degradation over the range of power levels between 95% and 100%.

Regarding the last transient in question, Letdown Line Break, the most negative value of MTC results in the most adverse reactivity feedback effect. Thus, the most negative value of COLR MTC ($-4.2 \times 10^{-4}$ $\Delta\rho$/°F at 100% and End-Of-Cycle (EOC)) is selected for this transient.

NRC Question 22:

Figures 6.3-154 and -241 show the same MDNBR yet 6.3-241 represents an all bounding case. Does this make physical sense? Please comment.

APS Response:

Even though both accidents result in the same minimum DNBR, they are not in the same accident classification categories. Refer to the response to Question 16 of Reference 3 for a detailed explanation.

NRC Question 23:

Regarding the Inadvertent Deboration event:

1. The licensees Analysis of Record (AOR) indicates that MODE 5 results in the least time available between detection and time to loss of shutdown margin (time available for the operator to take action). For the proposed uprated power level (MODE 1 operation), does the available operator action time remain bounded by the MODE 5 AOR time? Provide a technical basis for this conclusion. If not bounded, provide justification that the available operator action time is greater than the 15 minute requirement of NUREG-0800, Standard Review Plan.
2. The licensee states that, "the PUR has no impact on the ID event." Please discuss the technical basis for this conclusion that the proposed power level will not have any impact on operator action time during MODE 1 operation?

APS Response:

1. For Mode 1 and 2 operation, an inadvertent deboration results in a slow increase in core power and RCS temperatures. This critical event is slower than other power excursions analyzed, i.e. Uncontrolled CEA Withdrawal, and the reactor trips in time to prevent violation of the safety limits. The Technical Specification for Power Dependent Insertion Limit (PDIL) ensures enough trippable CEAs are available such that Shutdown Margin is preserved in Modes 1 and 2. The reactor trip ensures a second dilution period of at least 15 minutes in Modes 3 and lower. There is no change in the reactor trip functions and/or PDIL LCOs for power uprate, thus the limiting scenario, i.e. the Mode 5 case remains bounding for PUR.
2. Since Mode 5 operation is the most limiting in terms of operator action time for this event, the change in the reactor power level does not impact the results of the limiting Inadvertent Deboration Event. However, the PURLR Section 6.3.4.6 evaluation includes the consideration of larger RSGs, which increases RCS volume. The increase in RCS volume results in increasing the time for dilution

(or decreased dilution rate), making the inadvertent deboration event more benign for the PUR condition.