APS

Palo Verde Nuclear
Generating Station

David Mauldin
Vice President
Nuclear Engineering
and Support

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

10 CFR 50.90
Mail Station 7605
P O Box 52034
Phoenix, AZ 85072-2034

102-04847-CDM/TNW/RAB
October 11, 2002

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

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

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

Dear Sirs:

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

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

Attachment 2 to this letter provides written responses to the questions from the Reactor Systems Branch. Responses to questions from the remaining branches have previously been submitted.

A member of the **STARS** (Strategic Teaming and Resource Sharing) Alliance

Callaway • Comanche Peak • Diablo Canyon • Palo Verde • South Texas Project • Wolf Creek

A001

The information in Tables 31.a-1 through 31.a-9 and the response to question 35 in Enclosure 1 are considered proprietary by the Westinghouse Electric Corporation. Westinghouse requests that this information be withheld from public disclosure, in accordance with 10 CFR 2.790. Attachment 3 contains an affidavit from the Westinghouse Electric Corporation stating the reasons that this information should be considered proprietary.

Attachment 4 replaces Section 6.3.0.3.1 of Attachment 6 to Reference 1, in its entirety. The section was rewritten to provide a clearer and more thorough discussion of the Core Protection Calculator Dynamic Filter Coefficients.

Since the submittal of Reference 1 in December 2001, two issues have arisen as a result of work being done to support implementation of the CENTS Code at PVNGS. The first of these issues concerns the discovery by Westinghouse that a correction factor for pressurizer safety valve (PSV) orifice size, previously used when CESEC was the code of record, was inadvertently omitted when the analyses were converted to the CENTS code. Attachment 5 describes the issue and presents the corrected results for the affected analyses.

The second issue concerns the Analysis of Record (AOR) for Feedwater Line Break (FWLB) with Loss of Offsite Power (LOP) – Long Term Cooling Event. To address these issues the event has been reanalyzed. The new analysis affects Section 6.3.2.8.1 of Power Uprate Licensing Report attached to Reference 1. Results of the new analysis demonstrate that the acceptance criteria are satisfied for operation at 3990 MWt as well as at 3876 MWt. A description of the issue and results of the new analysis will be submitted to the NRC by November 22, 2002.

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/kg

Attachments:

1. Notarized Oath or Affirmation
2. Reactor Systems Branch Questions and APS Responses
3. Affidavit from Westinghouse Electric Corporation Submitted Pursuant to 10 CFR 2.790 to Consider Tables 31.a-1 Through 31.a-9 SER Limitations/Constraints Associated with the LBLOCA and SBLOCA Evaluation Models Used for the PUR ECCS Performance Analysis and NRC Question 35 on the Description of the Long Term Cooling, Boron Precipitation Model as Proprietary Information
4. Replacement Pages for Power Uprate Licensing Report Section 6.3.0.3.1
5. Pressurizer Safety Valve Orifice Sizing Correction Factor

Enclosures:

1. Tables 31.a-1 Through 31.a-9 SER Limitations/Constraints Associated with the LBLOCA and SBLOCA Evaluation Models Used for the PUR ECCS Performance Analysis and NRC Question 35 on the Description of the Long Term Cooling, Boron Precipitation Model
2. Revised Tables and Figures for the Pressurizer Safety Valve Orifice Sizing Correction Factor

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

Attachment 1

Notarized Oath or Affirmation

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 October, 2002.

Nora E. Meador

Notary Public

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

Notary Commission Stamp

Attachment 2

NRC Reactor Systems Branch Questions and APS Responses

Reactor Systems Branch

NRC Question 1:

Attachment 2, Section 2, of the application: The proposed uprate from 3876 MW_t to 3990 MW_t will add 114 MW_t. The submittal states that correspondingly 55 megawatts electric (MW_e) will be added. How will you achieve a 48.2 percent conversion for the 114 MW_t while the original thermal efficiency is about 32 percent?

APS Response:

Currently, to extend Original Steam Generator (OSG) tube life, PVNGS operates with a High-Pressure (HP) Feedwater (FW) heater bypass line open. This line, which is in parallel with the two HP FW heater strings, bypasses approximately 17% to 25% of the FW flow around the HP FW heaters. The resulting lower FW temperature extends the OSG tube life, but reduces the overall thermal efficiency.

The HP FW heater bypass line will be closed after the installation of the Replacement Steam Generators (RSG). Also, the new RSGs will contribute to increased plant efficiency because steam header pressure will be approximately 100 psi higher. Approximately 18 MW_e of the expected 55 MW_e power increase is attributed to these two factors. The requested 114 MW_t increase associated with Power Uprate (PUR) is expected to generate an additional 37 MW_e, which yields an efficiency of approximately 32.5 percent.

NRC Question 2:

Attachment 6, Section 2.1.3: The proposed uprate is based on the anticipated performance of the RSGs. However, the submittal does not state any provisions to verify (during initial operation) that the parameters chosen in the analysis stage are indeed those present in the operation of the plant.

APS Response:

Startup testing will be performed in accordance with the integrated startup test plan described in the Power Uprate Licensing Report (PURLR), Section 9.12. The startup tests will verify the design basis and expected operating parameters associated with RSG/PUR. The test plan will be performed during initial plant startup and ascension to full power operations following 2R11.

NRC Question 3:

Attachment 6, Sections 6.1.2 and 6.1.3: For the loss of coolant accident (LOCA), there is the large break LOCA (LBLOCA) and small break LOCA (SBLOCA). For the LBLOCA and SBLOCA of record, the planar linear heat generation rate (PLHGR) is listed as 13.1 kW/ft for the LBLOCA, and 13.5 kW/ft for the SBLOCA. What caused the difference in PLHGR in the two cases?

What codes have been used for the LBLOCA and SBLOCA? Where have these codes been reviewed and approved by the NRC staff? For the SBLOCA, have any changes to the code been made and, if yes, what would the effect be on the peak clad temperature and the amount of oxidation? (See also question 32b.)

APS Response:

The Emergency Core Cooling System (ECCS) performance analysis for the PURLR is the analysis performed in support of the Technical Specification amendment request for a 2% increase in the rated core power from 3800 MW_t to 3876 MW_t (Reference 36) that was submitted to the NRC in 1995. This analysis consisted of three parts: LBLOCA, SBLOCA, and post-LOCA Long Term Cooling. Although it is not explicitly described in Reference 36, all three parts of the ECCS performance analysis for the amendment request were performed for a 5% increase in the rated power to 3990 MW_t (4070 MW_t with a 2% power measurement uncertainty). Therefore, ECCS performance was not reanalyzed for this PUR request. That analysis, herein referred to as stretch power ECCS performance analysis, is the latest ECCS performance Analysis of Record (AOR) reviewed by the NRC. The NRC's Safety Evaluation Report (SER) for the stretch power ECCS performance analysis is documented in Reference 37.

As described in Reference 36, both the LBLOCA and SBLOCA stretch power analyses were performed for a PLHGR of 13.5 kW/ft. The LBLOCA portion of the ECCS performance analysis has since been revised to evaluate changes in plant configuration parameters, correct an error in the model (See response to the Question 31, Item b) and to address computer operating systems and hardware changes. As a result of the revisions, PLHGR was reduced from 13.5 kW/ft to 13.1 kW/ft. This change is reflected in the Core Operating Limits Report (COLR). All revisions were implemented under the provisions of 10 CFR Part 50.59 and/or 10 CFR Part 50.46.

Unlike the LBLOCA analysis, no revisions to the SBLOCA analysis that was performed in support of the stretch power were required. However, the results of the SBLOCA analysis for the stretch power remain applicable as the current SBLOCA ECCS performance analysis. When the PLHGR limit was reduced for LBLOCA analysis, the SBLOCA analysis was not revised since the LBLOCA analysis (versus the SBLOCA analysis) limits the maximum allowable value for the PLHGR in the ECCS performance analyses performed with the Westinghouse evaluation models for Combustion Engineering designed Pressurized Water Reactors (PWRs). In light of that fact, the SBLOCA analysis was not performed at a lower PLHGR as was done in the LBLOCA

analysis in order to create a bounding analysis. These changes resulted in the difference in PLHGR between the LBLOCA and SBLOCA cases.

The PUR LBLOCA and SBLOCA ECCS performance analyses use the 1985 EM and S1M versions of the Westinghouse ECCS performance evaluation models for Combustion Engineering designed PWRs, respectively. The computer codes used in these evaluation models, the associated topical reports, and the NRC SERs are listed in Tables 3-1 and 3-2. The codes have been reviewed and approved by the NRC.

Table 3-1: 1985 EM LBLOCA Evaluation Model Topical Reports and SERs		
Subject	Topical Report Reference	SER Reference
LBLOCA Evaluation Model, CENPD-132	1	26
Supplement 1	2	26
Supplement 2	3	27
Supplement 3	4	28
CEFLASH-4A, CENPD-133 (Blowdown Thermal Hydraulics)	5	26
Supplement 2	6	26
Supplement 4	7	29
Supplement 5	8	28
COMPERC-II, CENPD-134 (Refill/Reflood Thermal Hydraulics)	9	26
Supplement 1	10	26
Supplement 2	11	28
STRIKIN-II, CENPD-135 (Hot Rod Heatup)	12	26
Supplement 2	13	26
Supplement 4	14	30
Supplement 5	15	31
PARCH, CENPD-138 (Steam Cooling Heat Transfer Coefficients)	16	26
Supplement 1	17	26
Supplement 2	18	32
HCROSS, LD-81-095, Encl. 1, App. A (Blocked Channel Flow Fractions)	19	28
COMZIRC, CENPD-134, Sup. 1, App. C (Core-Wide Cladding Oxidation)	10	26
Application of FLECHT Correlation to 16x16 Fuel Assemblies, CENPD-213	20	33
Application of NUREG-0630 Cladding Rupture and Swelling Models, Enclosure 1 to LD-81-095	19	28

Table 3-2: S1M SBLOCA Evaluation Model Topical Reports and SERs		
Subject	Topical Report Reference	SER Reference
SBLOCA Evaluation Model, CENPD-137 Supplement 1	21 22	26 34
CEFLASH-4AS, CENPD-133 (Blowdown Thermal Hydraulics) Supplement 1 Supplement 3	 23 24	 26 34
STRIKIN-II, CENPD-135 (Hot Rod Heatup, Forced Convection Period) Supplement 2 Supplement 4 Supplement 5	 12 13 14 15	 26 26 30 31
PARCH, CENPD-138 (Hot Rod Heatup, Pool Boiling Period) Supplement 1 Supplement 2	 16 17 18	 26 26 32
Application of Cladding Rupture and Swelling Models to 16x16 Fuel Assemblies, CENPD-185	25	35

One error has been discovered and corrected in the SBLOCA evaluation model computer codes since the current AOR was performed. The error was in the CEFLASH-4AS computer code. The error and its impact on peak cladding temperature and cladding oxidation are discussed in the response to Question 33.

In addition to correcting the error described in the response to Question 33, two other changes were made to the SBLOCA evaluation model computer codes since 1995 due to changes associated with computer operating systems and hardware. The changes were made in 1997 (Reference 38) and in 2001 (Reference 39). The 1997 changes also included the automation of data transfer between the blowdown thermal hydraulics computer code, CEFLASH-4AS, and the pool-boiling hot rod heatup code PARCH. The operating system/hardware changes had an insignificant impact on peak cladding temperature. As reported in Reference 38, the automation of the data transfer between CEFLASH-4AS and PARCH resulted in less than a 3 °F decrease in peak cladding temperature.

References for Question 3:

1) CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.
2) CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.
3) CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.
4) 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.
5) CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.
6) CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.
7) CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.
8) CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.
9) CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.
10) CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.
11) CENPD-134, Supplement 2-A, "COMPERC-II, Program for Emergency Refill-Reflood of the Core," June 1985.
12) CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.
13) CENPD-135P Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.
14) CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.
15) CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.
16) CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.
17) CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.
18) CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.
19) LD-81-095, Enclosure 1-P-A, "C-E ECCS Evaluation Model, Flow Blockage Analysis," December 1981.
20) CENPD-213-P, "Reflood Heat Transfer, Application of FLECHT Reflood Heat Transfer Coefficients to C-E's 16x16 Fuel Bundles," January 1976.

21) CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.
22) CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.
23) CENPD-133P, Supplement 1, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," August 1974.
24) CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977.
25) CENPD-185-P-A, "Clad Rupture Behavior, LOCA Rupture Behavior of 16x16 Zircaloy Cladding," May 1975.
26) O. D. Parr (NRC) to F. M. Stern (C-E), NRC Staff Review of the Combustion Engineering ECCS Evaluation Model, dated June 13, 1975.
27) O. D. Parr (NRC) to A. E. Scherer (C-E), NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model Changes, dated December 9, 1975.
28) D. M. Crutchfield (NRC) to A. E. Scherer (C-E), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986.
29) S. A. Richards (NRC) to P. W. Richardson (Westinghouse CENP), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model' (TAC No. MA5660)," December 15, 2000.
30) K. Kniel (NRC) to A. E. Scherer (C-E), "Combustion Engineering Emergency Core Cooling System Evaluation Model," November 12, 1976.
31) R. L. Baer (NRC) to A. E. Scherer (C-E), "Evaluation of Topical Report CENPD-135, Supplement No. 5," September 6, 1978.
32) K. Kniel (NRC) to A. E. Scherer (C-E), "Evaluation of Topical Report CENPD-138, Supplement 2-P," April 10, 1978.
33) K. Kniel (NRC) to A. E. Scherer (C-E), Staff Evaluation of CENPD-213, dated August 2, 1976.
34) K. Kniel (NRC) to A. E. Scherer (C-E), "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P," September 27, 1977.
35) O. D. Parr (NRC) to A. E. Scherer (C-E), October 30, 1975.
36) 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.
37) Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 108 to Facility Operating License No. NPF-41, Amendment No. 100 to Facility Operating License No. NPF-51, and Amendment No. 80 to Facility Operating License No. NPF-74, Arizona Public Service Company, et al, Palo

Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Docket Nos. STN 50-528, STN 50-529, and STN 50-530, dated May 23, 1996.

38) Letter 102-04135-JML/SAB/RMW, J.M. Levine (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN-50-528/529/530, Errors in Emergency Core Cooling System (ECCS) Performance Evaluation Models, 10 CFR 50.46(a)(3)(ii) Annual Report," dated June 17, 1998.

39) Letter 102-04631-SAB/TNW/JAP, Scott A. Bauer (APS) to Document Control Desk (NRC), "Emergency Core Cooling System (ECCS) Performance Evaluation Models, 10 CFR 50.46(a)(3)(ii) Annual Report," dated December 13, 2001.

NRC Question 4:

Attachment 6, Section 6.3.0.1, "Methods and Computer Codes": For the sheared reactor coolant pump shaft with loss of power transient, the proposed analyses assume that the operator will manually refill the affected SG. Previous analyses did not assume manual action for this event. Why did the uprate require operator action? How was the operator response time estimated? Does this assumption meet regulatory guidance for operator action for design-basis events?

APS Response:

The change in the operator actions is not due to the PUR. The change reflects operator actions performed in accordance with approved plant procedures, the guidance provided in ANSI/ANS 58.8-1984, "Time Response Design Criteria for Safety-Related Operator Actions," and NRC-approved time lines for other events. Some of these actions, such as the early (as early as two minutes after trip) opening of the Atmospheric Dump Valves (ADVs), result in more adverse consequences than previous analyses. Other actions that are credited for the mitigation of the event, such as refilling the Steam Generator (SG) with the stuck-open ADV, are taken after the first 30 minutes. A discussion of the operator actions assumed in the new analysis and their bases is presented below:

- The new analysis assumes an early operator action of opening an ADV at two minutes after the reactor trip. This represents a more realistic time at which operators are most likely to first open an ADV for secondary pressure control than that which was previously assumed to occur 30 minutes after the trip. This assumption is more limiting in terms of the dose consequences since it results in an increase in the integrated ADV steam release to the atmosphere. This assumption is also consistent with the expected operator actions following a trip based on operating experience and simulator scenarios.
- Crediting the operator action to close the ADV on the unaffected SG two minutes after recognizing a stuck open ADV on the affected SG is consistent with the Emergency Operation Procedure (EOP) guidelines. The two-minute delay time between consecutive operator actions is consistent with the requirements of

ANSI/ANS 58.8. This operator action time line has been previously reviewed and approved by the NRC for the Steam Generator Tube Rupture with Loss of Power (SGTRLOP) with a single failure (stuck open ADV) (see NUREG 0857, Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Stations, Units 1, 2, and 3, Supplements 6 and 7).

- No further operator action is credited for 30 minutes into the event. After 30 minutes, an operator action is taken to direct the Auxiliary Feedwater (AFW) flow to the affected SG, and to refill and maintain level in the affected SG. The 30-minute time line for this action complies with the requirements of ANSI/ANS 58.8. In addition, this action is consistent with the guidelines provided in the EOPs. This action is similar to the one previously reviewed and approved by NRC for SGTRLOP with a single failure (stuck open ADV).

NRC Question 5:

Attachment 6, Section 6.3.0.1, "Methodology and Computer Codes" paragraph on Methods and Assumption Changes: Are the proposed changes in the assumptions within the scope of the approved methodologies?

APS Response:

The proposed changes have been evaluated and compared to the appropriate topical reports, NRC SERs, and SRP. The proposed changes are within the scope of the approved methodologies, and are consistent with the reviewed and approved EOPs and guidelines. The following paragraphs provide additional information regarding these changes:

- More Realistic Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV) with Loss of AC Power (LOP) and the Limiting Anticipated Operational Occurrence (AOO) with a single failure: This change constitutes a more realistic way of analyzing the two events that previously had been combined in order to simplify the simulation of the transients. While separating the two events, the new analysis is consistent with the approved methods. This change is included in the PURLR Section 6.3.0.1.

  Previous analyses conservatively ignored the Core Protection Calculator (CPC) Variable Overpower Trip (VOPT) and low SG pressure trips that would occur during an excess load with uncontrolled steaming, as well as the initial thermal margin in excess of the degradation associated with an 11% excess load. These analyses also conservatively forced the hot channel Departure from Nuclear Boiling Ratio (DNBR) to the Specified Acceptable Fuel Design Limits (SAFDL) at the onset of the LOP following turbine trip and simply modeled the loss of forced flow portion of the event. In the new analysis, the conservative assumptions that were made only to simplify the simulation of the event are removed. The IOSGADV + LOP event is separated from the simplified loss of flow from SAFDL methodology, and analyzed as increased heat removal by the secondary system

with LOP and single failure. The previously approved methodologies for an increased heat removal by the secondary system are used for IOSGADV with single failure (LOP), and the existing methodology of bounding all infrequent events is still maintained. A composite event was created to bound the DNBR degradation for all infrequent events, including AOOs in combination with a single active failure. This composite event assumes that an unspecified initiating event degrades all the thermal margin preserved by Core Operating Limit Supervisory System (COLSS) and brings the core conditions right to the DNBR SAFDL. The details of this evaluation are presented in Section 6.3.8 of the PURLR.

- Post-trip Main Steam Line Break (MSLB) detailed reactivity calculation: The CENTS code is used to analyze the MSLB event. The CENTS code has been previously reviewed and approved for use in non-LOCA transients generically (see "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, approved in a letter from Martin J. Virgilio, NRC to SA Toelle, ABB Combustion Engineering, dated 3/17/94), and for Palo Verde in Operating License Amendment No. 137 (Letter NRC to APS, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments Re: Various Administrative Controls (TAC Nos. MB1668, MB1669, and MB1670), dated October 15, 2001). The CENTS three-dimensional reactivity feedback option incorporates the capability of three-dimensional reactivity effects with local changes in moderator density, and is used for the post-trip MSLB. The three-dimensional reactivity contribution is based on approved physics methods as a function of core inlet plane temperature tilt (difference between hot and cold leg temperatures), core flow, and core fission power. A comparison of the limiting MSLB scenario, evaluated with and without this reactivity feedback option to demonstrate its application, is provided in Section 6.3.1.5.5.1 of the PURLR.
- Operator actions for Single Reactor Coolant Pump (RCP) Sheared Shaft with LOP event: see response to Question 4.
- Use of Iodine Decontamination Factor (DF) of 100 for unaffected SG: The use of an iodine DF of 100 when the SG tubes are submerged is described in the Standard Review Plan (SRP) for MSLB and Steam Generator Tube Rupture (SGTR). Based on the review of the SG mass and level fluctuations for a given UFSAR Chapter 15 event, the severity of level fluctuations in the unaffected SG due to the transient is similar or less than what is observed in the MSLB and SGTR events. Therefore, it has been determined that the use of a DF of 100 for other Chapter 15 events is appropriate. The only exception is the FW system pipe break in which the mass in the unaffected SG decreases until main steam isolation and delivery of AFW occurs. For that event, a DF of 1.0 is assumed for the unaffected SG. In addition, after 30 minutes, the operator is assumed to initiate a controlled cooldown of the Reactor Coolant System (RCS) to Shutdown Cooling (SDC) conditions. During the controlled cooldown portion of the events, the operators maintain the appropriate subcooled margin as well as the SG level in accordance with the EOPs. Use of a DF of 100 for the unaffected generator during the phase where the SG level is re-established is deemed acceptable.

NRC Question 6:

Attachment 6, Section 6.3.0.3, Table 6.3-3, low SG pressure: The numerical values in the table are in conflict with those on Page 4-12, please explain.

APS Response:

Table 6.3-3 presents the Reactor Protection System (RPS) analytical trip values used in the safety analysis. These values were selected to encompass the potential total loop uncertainties associated with the process instrumentation in normal and harsh environmental conditions. The values bound the allowable range for the Plant Protection System (PPS) bistable surveillance testing acceptance criteria. For example, the Instrumentation and Control (I&C) design setpoint for Low Steam Generator Pressure (LSGP) trip is 960 psia for PUR design conditions. The surveillance testing acceptance criterion associated with this setpoint is 5 psi. Therefore, the Technical Specification allowable value for the LSGP for PUR will be $\geq$955 psia. This is the value presented in PURLR Section 4.3.2 on page 4-12. On the other hand, the analytical values used in the safety analysis are based on the design setpoint of 960 psia, and include bounding instrument loop uncertainties of 45 psid and 85 psid for normal and harsh environmental conditions, respectively. Thus, the values presented in PURLR Table 6.3-3 for the LSGP setpoint are 915 psia (960 psia minus 45 psid) and 875 psia (960 minus 85 psid) for normal and harsh environmental conditions.

NRC Question 7:

Attachment 6, Section 6.3.1.3, "Increased Main Steam Flow": Table 6.3-6 indicates an automatic main steam isolation valve (MSIV) closure; however, the description of the transient simulation indicates manual closure of the MSIVs. Please explain this discrepancy.

APS Response:

The last sentence of the first paragraph of Section 6.3.1.3.3.1 should read: "The excess steam demand is terminated by automatic closure of the MSIVs." The MSIVs begin closing automatically on a main steam isolation signal (MSIS) that is initiated by the LSGP at 26 seconds for the PUR case, and are fully closed within 5.6 seconds (at 30.6 seconds). No manual action is assumed for the first 30 minutes of the transient, therefore the sentence mentioning manual closure is in error.

NRC Question 8:

Attachment 6, Section 6.3.1.4.3 and elsewhere, use of CENTS computer code for non-loss of coolant accident (non-LOCA) transient simulation: It is stated that the Unit 2 Amendment 137 qualified the CENTS code for non-LOCA transient analysis. It is also stated that there are limitations in the code. Was the code qualified for plant analysis for operation at 4070 MW_t (3990 x 1.02 = 4070) and if yes, how was it accomplished?

APS Response:

No limitation on power level was specified for the CENTS computer code. CENTS was qualified by comparison to other NRC approved safety analysis computer codes; principally CESEC-III, CEFLASH-4A, and RELAP5/MOD3. The response to question 5 contains the reference for NRC approval for generic use of the CENTS code.

CESEC-III is a non-LOCA safety analysis code and CEFLASH-4A is a LOCA safety analysis code that complies with 10 CFR 50, Appendix K. RELAP5/MOD3 is the NRC developed code used for non-LOCA and LOCA safety analyses that has been extensively benchmarked.

Neither CENTS, nor the computer codes against which it was benchmarked, contain phenomenological models which would inherently limit the operating power level at which they can be applied. The NRC SERs for generic use and plant specific use (see response to question 5) do not include any restrictions regarding the power level at which CENTS may be used.

NRC Question 9:

Attachment 6, Section 6.3.1.6, "Steam System Piping Failures Inside and Outside Containment - Mode 3 Operation" and elsewhere: The computer code HRISE is used for the estimation of the departure from nucleate boiling ratio (DNBR). Has this code been reviewed and approved by the NRC staff? Is the associated hand calculation of the linear heat generation rate at the time of return-to-power part of the approved process?

APS Response:

The HRISE computer code is a computer utility code that is used to assess the MacBeth DNBR correlation based on the transient code output. The use of HRISE in MSLB methodology has been reviewed by the NRC for the CE System 80 design. This review is documented in NUREG-0852, Safety Evaluation Report CESSAR System 80, Supplement 2, Appendix H.

The associated hand calculation of Linear Heat Generation Rate (LHGR) is separate from the examination of the core response with the MacBeth correlation. This calculation merely converts the predicted fractional core fission power production

reached during the Return-to-Power (R-t-P) phase from the transient code output to the core average LHGR value at full power conditions. This calculation may be expressed as:

CALHGR = [fractional core power * rated thermal power(kW)]/total length of fuel rods

and multiplies this result by Fq, the 3-D peaking factor to obtain the ratio of the rate of power production in the hottest node of the core to the average rate of power production in the core.

This formulation is consistent with the calculations inherent in the Core Protection Calculators (CEN-305-P, "Functional Design Requirement for a Core Protection Calculator", July 1985), and the evaluations performed for other Design Basis Events (Topical Report, "Reload Analysis Methodology for the Palo Verde Nuclear Generating Station," Revision 00-P-A).

NRC Question 10:

Attachment 6, Section 6.3.1.7.3, "Description of Analysis": This section identifies the limiting scenario (for "Pre-trip Main Steam Line Break Power Excursions") as the "...full power event with offsite power available". However, Table 6.3-21, which lists the parameters used for the analysis, indicates a 95 percent power level. Do the conclusions listed in Section 6.3.1.7.6 reflect the full power run, or the 95 percent run, as indicated in Table 6.3-21?

The DNBR calculation was performed using the CETOP-D and TORC computer codes. Have these codes been reviewed and approved by the NRC staff? How does the analyses using CETOP-D and TORC differ from analyses using the HRISE code elsewhere in this submittal? Discuss why CETOP-D and TORC were used instead of HRISE?

APS Response:

The analysis of the pre-trip SLB represents the largest increased main steam flow from high power conditions. Both the 95% and full power cases have the same initial thermal margin preserved by Technical Specifications and the 95% power case has the potential for a larger change in power before CPC VOPT trip. Therefore, the 95% initial power case is more limiting than full power case, and is presented.

The analysis credits the initial conditions of the event being a 'distance' from the DNBR SAFDL, which is the Required Over Power Margin (ROPM) that is maintained by operating in compliance with the plant Technical Specifications. The amount of ROPM preserved by the Technical Specifications is the same at 95% power as it is for 100% power. Thus, in terms of the transient having the potential to degrade DNBR to the point at which the DNBR SAFDL is exceeded, both the full power and 95% power cases are equivalent. Below 95% power, the plant Technical Specification's begin setting aside additional ROPM.

The most limiting scenario evaluated also credits the action of the CPC VOPT to generate a reactor trip. The CPC VOPT response is defined by various CPC constants. The setpoints are such that at full power conditions, the CPC VOPT setpoint is limited by a fixed ceiling. At 95% power conditions, the CPC VOPT setpoint is limited by an offset term above the current power level. Additionally, as actual core power starts to rise in response to a transient from 95%, the CPC VOPT setpoint is allowed to rise. The combination is such that the 95% initial power condition results in the possibility of a larger change in power before trip than the full power case.

The core response of the pre-trip SLB was examined with TORC and CETOP computer codes. Both codes were submitted to the NRC in the topicals CENPD-160-S for CETOP-D, and CENPD-161-P-A and CENPD-206-P for TORC. Please refer to Questions 20 and 21 for NRC correspondence that implicitly approved the use of these codes for PVNGS.

The pre-trip SLB involves the examination of the thermal hydraulic response of the core during the initial seconds of the SLB transient while the other SLB scenarios presented in Section 6.3 involve the evaluation of the thermal hydraulic response of the core in the R-t-P phase of the transient. The thermal hydraulic conditions during a pre-trip SLB transient are within the limitations and restrictions of the CE-1 correlation that is used to calculate the DNBR in CETOP-D and TORC. During the time examined for a pre-trip SLB transient, the CE-1 correlation provides conservative predictions of the thermal hydraulic state of the core. On the other hand, during the post-trip R-t-P or Mode 3 SLB scenarios, RCS temperatures and pressures decrease below the limitations and restrictions of the CE-1 correlation. Therefore, a different DNBR correlation (MacBeth) is used that is valid for the predicted thermal hydraulic conditions of these scenarios. Use of the MacBeth correlation has received NRC approval for these applications. The HRISE code is used to calculate the MacBeth DNBR based on the output of the transient predictions for the post-trip SLB and Mode 3 SLB transients.

NRC Question 11:

Attachment 6, Section 6.3.2.8.2.4 "Input Parameters, initial Conditions, and Assumptions": This section discusses the most limiting size break. Has this changed from the previous power level? Discuss how the most limiting break size was determined (e.g., by review of existing data, new analysis, or some other means).

In the same section, the 4th assumption, the primary to secondary heat transfer was assumed degraded. What is the physical basis for this degradation, and by what amount is the degradation?

Discuss if the values of the moderator temperature coefficient (MTC) in Tables 6.3-31 and 6.3-33 should be negative?

APS Response:

In accordance with the previously approved methodology for the FW system pipe break events (Small Feedwater Line Break (SFWLB) and Feedwater Line Break with Loss of Power and Single Failure (FWLBLOPSF)), a spectrum of breaks was analyzed in the sensitivity studies that were performed using the CENTS code to determine the most limiting break size.

For the SFWLB, small breaks less than or equal to 0.20 ft^2 were 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, which is the same size as that used in the previous AOR. Therefore, the limiting SFWLB break size did not change.

For the FWLBLOPSF, the spectrum of all break sizes is evaluated with various initial parameters. The limiting break size is different at different power levels.

The primary to secondary heat transfer degradation assumptions did not change for the FW system pipe break events. For SFWLB, the degradation model reviewed and approved for CE System 80 CESSAR Appendix B is maintained. The NRC's SER that demonstrated the approval of the method assumed for the degradation model and its technical basis are presented in Appendix G of NUREG-0852, Supplement 2. In accordance with the method, the heat transfer degradation was initiated at a SG liquid inventory of 35,000 lb_m, linearly decreasing to total loss of heat transfer when the SG empties of liquid. This method is not based on defined, physical properties of the SGs. Instead, the method conservatively models primary to secondary heat transfer degradation.

As noted in the Section 6.3.2.8 of the PURLR, depending on the enthalpy of the flow through the line break and the affected SG heat transfer characteristics, the break flow may induce either a RCS cooldown or heatup. However, the cooldown potential of a Feedwater Line Break (FWLB) is less severe than a MSLB. Therefore, the FWLB event is analyzed as a heatup event. For heatup events, it is more limiting to use the most positive (or least negative) MTC. The FWLB analyses use a MTC value of 0.0E-04 $\Delta\rho/^{\circ}F$ which is more conservative than the Core Operating Limits Report (COLR) value of -0.20E-04 $\Delta\rho/^{\circ}F$, at full power, Beginning of Cycle (BOC).

NRC Question 12:

Attachment 6, Section 6.3.3.1.5, Table 6.3-34: At 28.8 sec, the main steam safety valves (MSSVs) begin to cycle. The pressure-time function indicates a wavy pattern as shown in Figure 6.3-141. Discuss, if this is caused by MSSV cycling, is it not likely that the MSSVs will fail. Unlike the Power operated relief valves, the MSSVs are not designed for prolonged cycling.

APS Response:

The wavy pattern in secondary and primary pressures in the Figures 6.3-141 and 6.3-145 are due to the opening and closing of the first bank (low set pressure) MSSVs during the transient. The question was raised that MSSVs are not designed for prolonged cycling. This is a true statement; in fact, the valves are designed not to cycle rapidly but to actuate (open/close) within their specified nameplate blowdown capacity. This is accomplished through valve design as discussed below.

The Palo Verde MSSVs are Dresser Model 3707, spring loaded, self-actuating, pressure relief valves. The design encompasses an enclosed spring housing, huddle chamber, and upper and lower adjustment rings. This design maximizes the lift effort required to open the valve, which results in extended blowdown characteristics. The adjustment rings are set to meet system-operating requirements. The lower adjustment ring has a function in both the valve's opening and closing operation. It allows the valve to rapidly open to 70%. This in turn exposes the steam to a larger surface area on which to act, allowing the valve to more slowly go into full lift. As the steam is exhausted through the valve discharge and the over pressure condition is relieved, the valve begins to close. The position of the lower adjustment ring is set such that it cushions the disc upon re-closing preventing any damage to the valve internals or it's ability to subsequently re-actuate or cycle.

However, in the safety analyses, the MSSV's opening and closing characteristics are modeled conservatively to produce results that are more adverse. For example, for events where later opening and early closing of the MSSVs are more adverse, the energy release through the MSSVs is minimized. This conservatism is established by setting the pop-open to 70% open at the valve setpoint plus the maximum tolerance of 3%. A constant valve area is maintained until the valve is fully opened. As steam pressure decreases, the valve area is assumed to linearly decrease to 70% area at blowdown pressure. Below the blowdown pressure, the valves are modeled to shut closed. The more restrictive and abrupt opening and closing characteristics of the valve model result in more cycling of the valves than would be expected under actual conditions.

For valve cycling and its impact on the MSSV design functions, Dresser has performed cycle testing on Model 3707 safety valves. In 1988, Dresser subjected a Model 3707 MSSV to a 250 full flow cycle test. The test valve was installed on the Wyle Labs full flow steam test loop; instrumentation was installed to measure inlet steam, body, spring, and ambient temperatures. Pressure transducers were installed to measure inlet steam pressure, and a Linear Variable Differential Transformer (LVDT) was installed on the upper end of the valve's spindle to measure valve lift. The test environment simulated normal plant conditions. The test concluded after 250 valve cycles. The test was completed without incident with no anomalies noted, except for audible leakage at the conclusion of the test run.

In 1993, APS Engineering performed extensive testing of 12 Model 3707 MSSVs at the Westinghouse Service Center. The tests were performed as part of an effort to develop

a correlation between off-site and on-line testing methods. Similar to the Dresser tests mentioned above, these valves were instrumented, brought to simulated plant operating conditions, and repeatedly cycled (with restricted lift). Seven valves were cycled between 25 and 43 times every 5 to 10 minutes with no valve operating (actuating) anomalies noted.

Similarly, MSSVs were cycled during comparative testing efforts performed at the Diablo Canyon facility in 1992 and at the Westinghouse facility in 1993. No valve actuating anomalies due to cycling were noted during these tests.

MSSVs are designed to operate repeatedly as demanded by system operating conditions without detrimental effects. Extensive valve testing and years of operating history has demonstrated that the valves employed in the Palo Verde design function properly when subjected to overpressure conditions and will not fail during repeated cycling.

NRC Question 13:

Attachment 6, Section 6.3.3.4.4, sheared shaft event, "Input Parameters, Initial Conditions, and Assumptions": Table 6.3-35 includes an MTC value of -0.18E-04 Δρ/°F. Discuss how this value was selected.

APS Response:

The -0.18 $\times 10^{-4}$ Δρ/°F MTC value is the COLR BOC MTC value at 95% power level.

Due to the local increases in core coolant temperature during the RCP sheared shaft event, a most positive (or least negative) value of MTC results in the greatest degradation of thermal margin. In terms of the transient having the potential to degrade DNBR to the point at which the DNBR 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 sheared shaft 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% at which the plant Technical Specifications are preserving the same initial thermal margin.

NRC Question 14:

Attachment 6, Section 6.3.6.3.2.3.1, "Transient Simulation": The section credits a 100 sec manual reactor trip to maximize integrated steam flow out of the (assumed) stuck open automatic depressurization valve. However, recent experience with a steam generator tube rupture (SGTR), at Indian Point Unit 2, indicates that the operator was not able to trip the reactor for several minutes. Discuss how the time of this trip is justified in view of this actual operating experience. How would the transient have evolved should a more realistic manual trip time be assumed?

APS Response:

There are three analyses that, collectively, form the PVNGS licensing basis for Steam Generator Tube Rupture events: Steam Generator Tube Rupture without Loss of Offsite Power (SGTR), Steam Generator Tube Rupture with Loss of Offsite Power (SGTRLOP), and Steam Generator Tube Rupture with Loss of Offsite Power and a limiting Single Failure (SGTRLOPSF).

The SGTRLOPSF analysis conservatively assumes a manual reactor trip since it results in more adverse dose consequences, while the scenarios analyzed for the SGTR and/or SGTRLOP do not credit a manual reactor trip. The impact of the timing of the manual reactor trip for SGTRLOPSF is discussed below:

The SGTRLOPSF analysis assumes that the operator opens an ADV following the reactor trip and that the ADV on the affected SG fails open for the duration of the event. Thus, an earlier reactor trip results in an earlier opening of an ADV that in turn maximizes the integrated steaming out of the affected SG to the atmosphere, increasing the dose consequences of the event. On the other hand, a later reactor trip yields more adverse dose consequences since it allows the activity generated from the event Generated Iodine Spike (GIS) to build up. Therefore, for the SGTRLOPSF event, parametric studies have been performed to evaluate the impact of these competing effects of the trip time on the dose consequences. These studies have demonstrated that the increased integrated steaming from the stuck open ADV on the affected SG has more adverse impact on the dose consequences than the GIS build up. Thus, an earlier trip is more adverse for SGTRLOPSF event. As a result of this conclusion, the analysis used a manual reactor trip time that would encompass the earliest automatic trip and the expected initial operator actions. The expected automatic reactor trip for all SGTR events is the trip on CPC hot leg saturation. Had the CPC based automatic trip been retained, or had a later (more realistic) manual trip been assumed, the dose consequences of the SGTRLOPSF event would have become more benign.

Although the SGTR and SGTRLOP analyses do not credit a manual reactor trip, the scenarios studied in these analyses are, in terms of the sequence of events and consequences, more similar to the SGTR event experienced at Indian Point 2 on February 15, 2000, than the SGTRLOPSF scenario. For the cited Indian Point 2 SGTR event, the event initiated at 19:15 as a small SG tube leak (3.4 gallons/day) that gradually degraded with time. The operator action to manually scram was documented

at 19:29, i.e., 14 minutes into the event. The SGTRLOP analyses credited the automatic reactor trip on the CPC hot leg saturation, which occurred at 759 seconds (~12.3 minutes) and 898 seconds (~15 minutes) for the 3990 MW_t and 3876 MW_t cases, respectively. Thus, the sequence of events presented in PURLR Table 6.3-52 is comparable to the operating experience for actual SGTR event. A later reactor trip yields more adverse consequences for SGTR/SGTRLOP events since it allows the activity generated from the GIS to build up. Had a manual reactor trip prior to the CPC based automatic trip been assumed, the dose consequences would have been more benign for these cases.

NRC Question 15:

Attachment 6, Section 6.3.6.3.2.4, SGTRLOP, [steam generator tube rupture loss of offsite power] "Input Parameters, Initial Conditions, and Assumptions": As in Section 6.3.6.3.2.3.1 above, an assumption is made for a 100 sec. operator trip. Discuss how the time of this trip is justified in view of actual operating experience, and being earlier than the core protection calculator (CPC) action?

APS Response:

The assumptions listed in Section 6.3.6.3.2.4 represent the SGTRLOPSF event that assumed a manual trip at 100 seconds. The manual trip does not apply to the SGTRLOP event that is presented in Section 6.3.6.3.3. Please refer to the response to Question 14 for the justification of trip times in view of actual operating experience and CPC action for both SGTRLOP and SGTRLOPSF events.

NRC Question 16:

Attachment 6, Section 6.3.8, "Limiting Infrequent Events": In the selection of the limiting abnormal operating occurrence (AOO) with a single active failure, the LOP from DNBR at the specified acceptable fuel design limit (SAFDL) value was selected. The calculated minimum DNBR was 1.17. However, the sheared shaft transient (Figure 6.3-154) resulted in the same DNBR value of 1.17 but started from normal operation DNBR.

Discuss would it be reasonable to conclude that the sheared shaft is the limiting AOO, because it would result in a lower DNBR value than 1.17 had it started from the SAFDL DNBR value?

APS Response:

It would not be reasonable to conclude that the RCP sheared shaft event as the limiting Anticipated Operational Occurrence (AOO). RCP sheared shaft and seized rotor events are Limiting Fault Design Basis Accidents (DBA) as defined in the Standard Review Plan, Regulatory Guide 1.70, and ANSI/ANS-51.1. Limiting Fault DBAs are

occurrences that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. The sheared shaft event, as a limiting fault, may violate the DNBR and result in fuel damage provided that radiological consequences remain acceptable.

An AOO in combination with a single failure is an infrequent event. Infrequent events are the events, which may occur during the lifetime of a facility for which DNBR SAFDL violation is acceptable provided that the radiological consequences are acceptable.

During normal operation, conformance with plant Technical Specifications will ensure that initial conditions from which design basis events are postulated to occur are a known amount of thermal margin 'away' from the DNBR SAFDL condition.

The analysis of the sheared shaft event credits the ROPM of the transient at the initiation of the event. The thermal margin degradation of this event is more adverse than either the AOO alone or the flow coastdown resulting from the assumed single failure (LOP) alone. The total thermal margin degradation for the sheared shaft event is equal to that of the AOO with single failure, resulting in the same value of minimum DNBR.

The analysis of the limiting infrequent event is that of the most adverse possible AOO in combination with the most adverse single active failure. The underlying AOO is postulated to have degraded all of the initial thermal margin preserved by the Technical Specifications and the plant is on the verge of generating a CPC low DNBR trip. The most adverse single active failure would then be the failure that would result in the most rapid decrease in DNBR. This would be the LOP and the resultant coast down of the RCPs.

In summary, even though both accidents result in the same minimum DNBR, they are not in the same accident classification categories. Therefore, it is not reasonable to assume that the RCP sheared shaft transient is the limiting AOO. It is, however, a limiting fault event.

<u>NRC Question 17:</u>

Attachment 6, Section 7.5, "Neutron Fluence": The vessel fluence in the analysis of record (AOR) was calculated for a power level of 4200 MW_t. Discuss the following: (1) does the AOR satisfy the guidance in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (2) what is the value of the end of life reference temperature for null ductility transition (RT_{NDT}), and (3) in the RT_{NDT} computation process, were any adjustments made on the calculated fluence value based on dosimetry measurements? If yes, for Item (3), provide the data for the adjustment.

APS Response:

The AOR was not performed following the guidance in Regulatory Guide 1.190. The AOR dates back to the initial safety analysis, before issuance of Regulatory Guide 1.190, and is the basis for the design of the reactor pressure vessel. The AOR end-of-life fluence is 3.29E+19 n/cm^2 for the vessel inside surface. In 1994, WCAP-13935, "Analysis of 137 Degree Capsule from the Arizona Public Service Company Palo Verde Unit No. 2 Reactor Vessel Radiation Surveillance Program," was issued. Comparing the WCAP-13935 analysis to the PURLR showed that the projected end-of-life (32 EFPY) fluence was approximately one-third lower than the value in the AOR (i.e., one-third more conservative than the assessment done for the PURLR). The large difference between the AOR and the WCAP-13935 analysis is based on the fact that the latter did account for actual plant operation, and much of the difference is a reflection of the low leakage fuel management program employed.

The maximum predicted value of the end-of-life RT_{NDT} for the Palo Verde Unit 2 reactor pressure vessel is 78 °F and applies to the intermediate shell plate F 765-6. This value was based on the predicted peak fluence calculated in the AOR for the vessel inside surface, 3.29E+19 n/cm^2, at end-of-life. The predicted value of end-of-life RT_{NDT} was calculated using Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, and includes the initial RT_{NDT}, the predicted shift, and an uncertainty value of 34 °F.

In the RT_{NDT} computation process, no adjustments were made on the calculated fluence value based on dosimetry measurements. The AOR end-of-life fluence, 3.29E+19 n/cm^2 for the vessel inside surface, was used directly, and that value was based solely on calculations.

NRC Question 18:

Attachment 6, Section 6.3: For the analyses of various design basis transient events described in the section, the values of the fuel rod gap conductance range from 500 Btu/hr-ft^2-°F for the sheared RCP shaft event, 6,100 Btu/hr-ft^2-°F for the steam bypass control system malfunction event, to 6,984 Btu/hr-ft^2-°F for the CEA ejection event.

NRC Question 18.a:

Discuss how these various values of fuel rod gap conductance are calculated, and what are the bases for determining which value to use as an initial condition of a particular event.

APS Response:

The FATES3A fuel performance code (References 1 through 3) is used to derive the fuel rod gap conductance data for the PVNGS fuel. The predicted core average maximum and minimum gap conductance data is used in the events where a conservative estimate of core average behavior is required. The maximum core average gap conductance is generated for the case where fuel and clad are in firm contact at the maximum contact pressure at a specified Core Average Linear Heat Rate (CALHR). The maximum gap conductance, as a function of CALHR for both Hot Full Power (HFP) and Hot Zero Power (HZP), is derived for a given fuel and core design. The minimum gap conductance values are derived as a function of CALHR and core average burnup. For a given fuel and core design, the minimum core average gap conductance is determined at Beginning of Cycle (BOC) and End of Cycle (EOC) at full power by using the minimum BOC and EOC core average burnup and the minimum CALHR. The calculated generic and bounding set of minimum and maximum core average gap conductance data are verified to remain bounding for each reload cycle.

For a particular event and the specific criterion evaluated for that event, the minimum or maximum average gap conductance is selected as an initial condition to make the event consequences more adverse. The basis for the selection is either the evaluation of the physical phenomena (delay or promptness of energy release from the fuel to clad to coolant, or duration of the energy release after a reactor trip, etc.), or the results of sensitivity studies. The difference in the values stated in the question is due to the selection of either minimum or maximum core average conductance for a particular event as an initial condition.

The selection of the gap conductance may also differ depending on the criterion that is being evaluated for the same event. For example, the CEA ejection event is evaluated for both fuel/cladding integrity (fuel temperature and enthalpy case) and RCS peak pressure (peak pressure case). The initial conditions for the gap conductance are different for each CEA ejection event case. In the CEA ejection fuel temperature and enthalpy case, the STRIKIN-II code that evaluates the hot channel based on the average channel, is used. In this case, the maximum gap conductance is used for the average channel so that the fuel temperature rise is reduced due to the heat loss to the coolant, minimizing the Fuel Temperature Coefficient (FTC) feedback in order to maximize the power increase. The hot channel power is then calculated by multiplying the average channel power by the radial peaking factor. In order to maximize the energy deposit in the hot pin fuel pellets, the minimum gap conductance is used for the hot channel so that the heat transfer to the coolant is minimized. On the other hand, for the CEA ejection RCS peak pressure case, where the CENTS code is used to predict the Nuclear Steam Supply System (NSSS) response, a conservative estimate of core average behavior is determined. For this case, the maximum gap conductance is used in order to increase the heat transfer to the coolant so that the energy transferred to the RCS is maximized.

It is also true that the minimum and maximum values differ in the different events. As in the example given in the question, the maximum gap conductance is 6,100 Btu/hr-ft^2- °F

for the Steam Bypass Control System (SBCS) malfunction event, while 6,984 Btu/hr-ft^2-°F is used to determine RCS peak pressure for the CEA ejection event. These differences are due to the discretion of the analyst in choosing a more bounding initial condition for the event. A maximum core average gap conductance that is calculated for a CALHR of 8 kW/ft is selected for the SBCS Malfunction event to bound the reload core designs which have a predicted CALHR of about 5-6 kW/ft. The value selected for the CEA ejection event also bounds the reload core designs, but was selected to be the maximum gap conductance value calculated for a CALHR in excess of 20 kW/ft.

References for Question 18.a:

1) "C-E Fuel Evaluation Model Topical Report," Combustion Engineering, Inc., CENPD-139-P-A, July 1974.
2) "Improvements to Fuel Evaluation Model," Combustion Engineering, Inc. CEN-161(B)-P, July 1981.
3) Letter, R.A. Clark (NRC) to A.E. Londvall, Jr. (BG&E),"Safety Evaluation of CEN-161 (FATES3)," March 31, 1983.

NRC Question 18.b:

Does the gap conductance value change or is it held constant during a transient and what is the basis for this?

APS Response:

The gap conductance value is either calculated dynamically during the transient or held constant depending on the event. In the events where a conservative estimate of core average behavior is sufficient, either maximum or minimum initial gap conductance is selected and held constant to encompass the range of the power level (or CALHR) experienced during the event for simplification of analysis. For the events that require localized effects of the gap conductance to be evaluated, a dynamic calculation of the gap conductance is performed. As explained in the response to the Question 18a, the CEA ejection event is one example where both cases can be observed. For the CEA ejection fuel temperature and enthalpy case, the STRIKIN-II code solves (radially) the one-dimensional cylindrical heat conduction equation for each axial region along the rod. The conduction model explicitly represents the gas gap region and dynamically calculates the gap conductance in each axial region. On the other hand, for the CEA ejection RCS peak pressure case, the initial gap conductance is held constant since the calculated gap conductance is well below this initial value and remains below this value during the transient.

NRC Question 18.c:

Provide sensitivity study results, which show the effects of the input values of gap conductance on the analysis results of various events.

APS Response:

The following tables illustrate the effects of initial gap conductance on the analysis results of various events:

Table 18.c-1: Sensitivity Study on Initial Gap Conductance for FWLB RCS Peak Pressure	
Gap Conductance (Btu/hr-ft^2-°F)	RCS Peak Pressure (psia)
5000	2676.4
3000	2677.6
1000	2680.1
500	2682.5

Table 18.c-2: Sensitivity Study on Initial Gap Conductance for Loss of Condenser Vacuum (LOCV) Event		
Gap Conductance (Btu/hr-ft^2-°F)	RCS Peak Pressure (psia)	Secondary Peak Pressure (psia)
5000	2696.5	1381.0
3000	2697.1	1382.1
1000	2698.9	1382.4
400	2702.8	1382.5

Table 18.c-3: Sensitivity Study on Initial Gap Conductance for Post-Trip MSLB Event		
Gap Conductance (Btu/hr-ft^2-°F)	Minimum Departure from Nucleate Boiling Ratio (MDNBR)	Time of MDNBR (seconds)
6527	3.58	50
594	2.94	272

NRC Question 19:

Attachment 6, Sections 7.1 and 7.2: With respect to the impacts of the proposed PUR on the core thermal-hydraulic design and core design, confirm that all parameters and assumptions to be used for analyses described in Sections 7.1 and 7.2 remain within any code limitations or restrictions. Describe the process used to support the conclusions.

APS Response:

Section 7.1 of the PURLR describes the core thermal-hydraulic analyses performed in support of operation of Unit 2 at an uprated power of 3990 MW_t. The methods employed in these analyses are consistent with the approved reload analyses methods. All the generic assumptions and parameters that are used for the current plant configuration (i.e. 3876 MW_t) and uprated power (i.e. 3990 MW_t) are listed in Table 19-1. These generic parameters and assumptions remain within the code limitations and restrictions. In accordance with the reload methods, these parameters are verified to remain within the code restrictions and limitations for a specific reload cycle. APS will continue to perform cycle specific thermal-hydraulic 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.

Table 19-1: Core Thermal-Hydraulic Parameters at Full Power			
General Characteristic	Units	3876 MW_t	3990 MW_t
Total Heat Output (core only)	MW_t	3,876	3,990
Fraction of Heat Generated in Fuel Rod	—	0.975	0.975
Primary System Pressure (nominal)	psia	2,250	2,250
Inlet Temperature (nominal)	°F	554	557
Total RCS Flow	gpm	423,320	423,320
Technical Specification minimum flowrate	lb_m/hr	155.8E+06	155.8E+06
Coolant Flow Through Core (minimum)	lb_m/hr	151.1E+06	151.1E+06
Hydraulic Diameter (nominal channel)	ft.	0.039	0.039
Average Mass Velocity	lb_m/hr-ft^2	2.49E+06	2.49E+06
Minimum Pressure Drop Across Core (steady state flow irreversible ΔP over entire fuel assembly)	psid	16.3	16.3
Total ΔP Across Vessel (based on nominal dimensions and minimum steady state flow)	psid	52.6	52.5
Core Average Heat Flux (based on the fraction of heat added in fuel rods)	Btu/hr-ft^2	182,073	187,429
Total Heat Transfer Area	ft^2	70,834	70,834
Film Coefficient at Average Conditions	Btu/hr-ft^2-°F	6,220	6,240
Average Film Temperature Difference	°F	29.3	30.0
Average Linear Heat Rate of Undensified Fuel Rod (account for fraction of heat generated in fuel rod)	kW/ft	5.32	5.48
Average Core Enthalpy Rise	Btu/lb_m	87.5	90.1
Maximum Clad Surface Temperature	°F	656.7(1)	656.7(1)

Note 1: These values are the same due to the rounding.

NRC Question 20:

Attachment 6, Section 7.1: References 7-2 and 7-4 are letters requesting that the NRC review and approve a revised core inlet flow distribution methodology and the specific application of the CETOP-D computer code for Palo Verde Units I, 2 and 3. For the revised core inlet flow distribution methodology, the safety evaluation report (SER) was written specifically for Unit I and included a statement that the licensee plans to submit a generic application addressing a revised maximum departure from nucleate boiling ratio (MDNBR) setpoint for all three units. Has the generic application been submitted and reviewed, and approved by the NRC? Please provide a reference to the NRC SERs, which granted these approvals for Unit 2. Also, for the PUR conditions, are all parameters within the restrictions or limitations of this methodology?

APS Response:

References 7-2 and 7-4 have been submitted for review and approved by the NRC as follows:

Reference 7-2:

NRC letter to APS dated May 26, 1994, "Issuance of Amendments for the Palo Verde Nuclear Generating Station Unit No. 1 (TAC No. M88679), Unit No. 2 (TAC No. M88680), and Unit No. 3 (TAC No. M88681)." This letter issued amendments 76, 62, and 48 to the Facility Operating Licenses for Unit 1, 2, and 3, respectively. These amendments approved a change in the DNBR limit and use of the core inlet flow distribution methodology.

Reference 7-4:

NRC letter to APS dated June 14, 1993, "Approval of Reload Analysis Methodology Report - PVNGS (TAC Nos. M85153, M85154, AND M85155)." This letter provided NRC approval allowing APS to perform reload designs. Implicit in this letter is NRC approval for APS to use the CETOP-D Code.

All the assumptions and parameters that are used for uprated power (i.e. 3990 MW_t) remain within the limitations and restrictions of this methodology. APS will continue to perform cycle specific thermal-hydraulic analyses and verify applicability of the assumptions and parameters to future reload cycles, including the uprated unit reloads, in accordance with the current reload process and methods.

NRC Question 21:

Attachment 6, Section 7.1: The steady state departure from nucleate boiling (DNB) analysis was performed using the methodology of Reference 7-5, CEN-356(V)-P-A, Revision 01 -P-A, "Modified Statistical Combination of Uncertainties," dated May 1988. The NRC SER for this methodology was written specifically for Palo Verde Unit 1. For clarification, please provide the technical justification for the application of this methodology to Unit 2, or provide a reference to the approved SER for Unit 2. Have any modifications been made to the methodology since implementation on Unit I? If so, please describe and provide the technical basis to support the change.

APS Response:

As stated in the Core Operating Limits Report (COLR) for Unit 2, "Core Operating Limits Report, Palo Verde Nuclear Generating Station, Unit 2," Revision 6, Section "Analytical Methods," this analytical method has been previously reviewed and approved by the NRC for use in Unit 2. The licensing amendments authorizing the COLR, and use of the analytical methods therein, were approved in NRC letter to Mr. William Conway (APS Co.), "Issuance of 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)."

No revisions have been made to the methodology described in CEN-356(V)-P-A, Revision 01-P-A. However, the DNBR limit has been changed through subsequent license amendment requests (refer to response to the Question 22).

NRC Question 22:

Attachment 6, Section 7.1: The section provides a TORC computer code calculated 95/95 DNBR limit of 1.34. This value is not consistent with the Updated Final Safety Analysis Report (UFSAR) value of 1.30, or with the Reference 7-5 value of 1.24. Please provide the technical basis for this difference. Also, discuss the impact of this change on the CPC and core operating limit supervisory system overall uncertainty penalty factors.

APS Response:

The DNBR Safety Limit (SL) provided in Technical Specification (TS) 2.1.1.1 and Reactor Protection System (RPS) DNBR RPS Trip Setpoint provided in Technical Specification Table 3.3.1-1 has been changed from 1.30 to 1.34 for the core operating cycles 11 and later. Therefore, the DNBR limit of 1.34 is used for the PUR cycle. The license amendment approving this change is documented in NRC letter from Jack N. Donohew (NRC) to Gregg R. Overbeck (APS), dated March 28, 2001, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments to Changing Minimum Departure form Nuclear Boiling Ratio (TAC Nos. MB0745, MB0746, and MB0747)". The UFSAR update which will reflect this change is due to be submitted to the NRC by June 2003 in accordance with 10 CFR 50.71(c).

The uncertainties in inlet flow to the hot assembly and the adjacent assemblies that was first encountered in operating cycle 7 for Unit 1 resulted in a change in the methodology to evaluate the core inlet flow distribution and associated uncertainties. As a result of the re-evaluation, the DNBR safety limit was increased from the 1.24 value used in Reference 7-5 to 1.30. The license amendment approving this change was documented in NRC letter from Brian E. Holian (NRC) to William F. Conway (APS), dated May 26, 1994, "Issuance of Amendments for the Palo Verde Nuclear Generating Station Unit No. 1 (TAC No. M88679), Unit No. 2 (TAC No. M88680), and Unit No. 3 (TAC No. M88681)."

The overall uncertainty penalty factors applied to the Core Protection Calculators (CPC) and the Core Operating Limit Supervisory System (COLSS) are calculated in accordance with the appropriate NRC-approved methodologies, and are applicable to PUR cycles. Therefore, the CPC and COLSS penalty factors will continue to provide the 95/95 probability/confidence level that the hot rod will not experience DNB during normal operations and anticipated operational occurrences for PUR cycles.

NRC Question 23:

Attachment 6, Section 6.3.4 - Reactivity and Power Distribution Anomalies:

NRC Question 23.a:

Discuss the methodology used to calculate the reactivity insertion rates used in these analyses and provide the technical basis for the initial power level assumptions for the UCEAW from subcritical and Hot Zero Power.

APS Response:

A bounding set of reactivity insertion rates were developed based on the approved physics methods identified in the COLR. These bounding values were selected based on the review of existing plant data at the time the bounding analysis was performed. These values are verified to remain bounding for each reload cycle.

There are two inputs for the initial power level for the subcritical Control Element Assembly Withdrawal (CEAW) case that impact the total reactivity insertion and subsequent peak power during the transient: the source neutron power and initial core power fraction. The source neutron power at subcritical condition represents the power that corresponds to the neutron production of the fuel material present at a given k_{eff}. The limiting value for the initial power fraction is determined by iterating the initial power fraction until the calculated value of the inserted reactivity is equal to the difference between the calculated subcritical margin and the CEA worth.

The minimum initial power for HZP CEAW case is chosen to maximize the reactivity insertion before reaching the reactor trip setpoint. For HZP CEAW scenario, the minimum initial power is the same as the high log power trip bypass setpoint, which is 1×10^{-4} % of HFP. However, the initial power is adjusted to 0.19×10^{-4} % of HFP to include instrumentation uncertainties in the analysis.

NRC Question 23.b:

All reactivity transients assume that no SG tubes are plugged. For each of the transients, please discuss the impact of 10 percent SG tube plugging on the results of these events and compare to the corresponding acceptance criteria. What is the basis for the upper limit of 10 percent SG tube plugging for the PUR conditions?

APS Response:

For the CEAW events, the worst-case values of DNBR and fuel centerline temperature are based on the minimum heat removal from the core by the secondary systems. In the analysis of CEAW events, secondary side flow through the SGs is minimized such

that secondary heat removal effects are not credited in the analysis. This is accomplished by manually inputting an arbitrarily small value for secondary system flow into the transient code. Therefore, SG tube plugging has no effect on the results of the analysis and this parameter should be removed from Tables 6.3-37 and 6.3-39 of the PURLR. Tables 23.b-1 and 23.b-2 provide the revised Tables 6.3-37 and 6.3-39, respectively.

Table 23.b-1: Revised Table 6.3-37
Parameters Used for the CEAW from Subcritical Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	6.27E-10	6.27E-10
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	1785	1785
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	0.5E-04	0.5E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Reactivity insertion rate (%$\Delta\rho$/in)	0.065	0.065
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6530	6530
Single failure	none	none
LOP	no	no

Table 23.b-2: Revised Table 6.3-39 Parameters Used for the CEAW from HZP Event		
PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	0.19E-04	0.19E-04
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	2000	2100
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC (Δρ/°F)	0.5E-04	0.5E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Reactivity insertion rate (%Δρ/in)	0.040	0.040
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6530	6530
Single failure	none	none
LOP	no	no

NRC Question 24:

Attachment 6, Section 6.3.4.1.1: For the UCEAW from subcritical event, there is a rather significant difference in the duration of this event when compared to the UFSAR AOR. The UFSAR analysis shows this event terminates at approximately 300 seconds after initiation, while the new analyses performed for the PUR terminates at approximately 55 seconds. Discuss the changes in key parameters as a result of the PUR, which would explain this difference.

APS Response:

Section 15.4.1 of UFSAR, Revision 11, describes the analysis of record (AOR) for the UCEAW from Subcritical Conditions event. Table 15.4.1-1 provides the sequence of events for this event. Comparing this table to that provided in PURLR Table 6.3-38 demonstrates that the duration of the event for the current AOR and the PUR AOR are essentially the same.

Previous AORs took into consideration the movement of the shutdown CEA banks to achieve criticality. As a result of this input, the time to achieve criticality occurred much later in the transient due to the high worth of these banks. Movement of the shutdown CEA banks was later removed from the analysis to incorporate the requirements of Technical Specification 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed, Limiting Condition of Operation 3.1.2c. This condition states "Reactor criticality shall not

be achieved with shutdown CEA movement." Since shutdown CEA banks are no longer considered in the AOR, the duration of the transient has significantly decreased.

NRC Question 25:

Attachment 6, Section 6.3.4.1.1: For the UCEAW from Subcritical event, Figure 6.3-158 shows that the RCS pressure oscillates for approximately 15 seconds following the reactor scram. Discuss the expected magnitude, frequency, and the physical phenomenon causing these oscillations. Oscillations are not evident for any other key parameters for this event.

APS Response:

The oscillations observed in the RCS pressure are the result of the natural frequency of the system.

The subcritical CEAW results in a brief but dramatic spike of reactor power around the time of the reactor trip. This spike results in the heat-up of a portion of the RCS. The RCS undergoes a rapid increase in pressure as the coolant attempts to expand as a result of the heat-up. The expansion of the RCS results in an in-surge to the pressurizer. The momentum of the in-surge compresses the steam space of the pressurizer resulting in the pressurizer temporarily having a higher pressure than the RCS as a whole. This pressure difference across the surge line reverses the surge line flow and an out-surge from the pressurizer occurs.

The magnitude of the pressure oscillations rapidly decays as the system achieves equilibrium. After approximately 8 cycles of diminishing oscillations during the first 15 seconds, the oscillations cease. The maximum RCS pressure during the transient is less than 100 psi above the initial pressure.

NRC Question 26:

Attachment 6, Section 6.3.4.1.1: For the UCEAW from Hot Zero Power event, the resulting maximum RCS pressure in Table 6.3-40 is not consistent with the results shown in Figure 6.3-164. Explain this discrepancy.

APS Response:

Table 6.3-40 provides the peak RCS pressure, as calculated at the downstream node of the RCP. Figure 6.3-164 is a plot of the pressurizer pressure as calculated in the pressurizer node. Thus, the title of Figure 6.3-164 incorrectly states the pressurizer pressure as RCS pressure, causing the discrepancy. Also, the same discrepancy exists for Table 6.3-38 and Figure 6.3-158.

The correct title of Figure 6.3-158 is "Uncontrolled CEAW from Subcritical - Pressurizer Pressure vs. Time" and the correct title of Figure 6.3-164 is "Uncontrolled CEAW from HZP-Pressurizer Pressure vs. Time."

NRC Question 27:

Attachment 6, Section 6.3.4.2 - UCEAW at Power: The UFSAR states that parametric studies performed for initial condition determinations indicate that minimum DNBR during the CEAW at power is most sensitive to initial core inlet temperature. As such, the UFSAR analysis of record assumed the maximum allowable core inlet temperature (580 °F). In the PUR reanalysis for this event, an initial core inlet temperature at the lower limit of the range (548 °F) was assumed. Provide a justification for assuming the lower limit of the core inlet temperature range and discuss the impact on DNBR results if the higher temperature is assumed.

APS Response:

The statement in the UFSAR is based on the original AOR of Cycle 1. The analyses for subsequent cycles credit the CPC Variable Overpower Trip (CPC VOPT) to generate a reactor trip. The analysis also credits the initial conditions of the event being a known amount of thermal margin "away" from the DNBR SAFDL condition during the normal operation. This known initial thermal margin is maintained by the plant Technical Specifications.

The initial values of the six important thermal hydraulic parameters (RCS flow, radial peaking factor, Axial Shape Index (ASI), core inlet temperature, average heat flux, and RCS pressure) are adjusted so that the initial thermal margin is minimized. Only if the 'sense' (minimum/maximum, etc.) of a parameter adversely affects the transient simulation, an effort is made to select that parameter as the limiting initial condition. Other parameters would be allowed to 'float' until the minimum thermal margin condition existed at time equal to zero.

The CPC VOPT is dependent on core inlet temperature effects on core neutron power as measured by the excore instrumentation. A lower core inlet temperature results in a cooler coolant in the reactor vessel downcomer region resulting in temperature shadowing of these detectors. As a result, the measured core power will be lower than the actual core power. Therefore, a lower core inlet temperature results in a later reactor trip, which allows the core power to increase further prior to the reactor trip. This effect is maximized in the analysis by using the minimum core inlet temperature, and the other parameters are allowed to 'float' until the minimum thermal margin condition existed at time equal to zero.

A case starting from a higher initial core inlet temperature would result in an earlier CPC VOPT than a case starting from a lower initial core inlet temperature. This, in turn, would result in lower thermal degradation during the transient. Thus, the minimum

DNBR would be further away from the SAFDL providing the same minimum initial thermal margin exists for both cases.

NRC Question 28:

Attachment 6, Section 6.3.4.3 - Full Length Control Element Assembly (CEA) Drop Event:

NRC Question 28.a:

Provide the technical basis for assuming the initial core power of 95 percent of rated thermal power. Discuss if this assumption bounds the requested uprate power of 3990 MW_t?

APS Response:

The initial thermal margin (ROPM) preserved by the Technical Specifications is a function of core power level. The same amount of ROPM is preserved between 100% and 95% power levels. Below 95% power level, additional amounts of ROPM are preserved. Also, the radial power distribution distortion potential of a dropped CEA increases as the core power level is decreased. Therefore, the radial distortion factor associated with the dropped CEA is evaluated at several power levels (20%, 50%, 65%, and 95%) for each reload cycle to determine the power dependent ROPM values that the Technical Specifications must preserve for that specific core design.

The limiting case for both 3876 MWt and 3990 MWt rated power units is the 95% power level case. This will be verified for the uprated power cycles during the reload process in accordance with the current methods.

NRC Question 28.b:

Provide the technical basis for reactivity parameter values assumed for this event and listed in Table 6.3-43.

APS Response:

With the exception of dropped CEA reactivity worth, the other reactivity parameters are selected in accordance with the PURLR Section 6.3.0.2.

The dropped CEA reactivity worth used in the presented analysis is typical of the single CEA drop worth for the core that was examined as a potential uprate core, and is based on the existing data from the previous core designs. In reality, the negative reactivity

worth associated with the dropped CEA does not impact the results. This analysis assumes that the core has totally recovered initial pre-drop power due to MTC feedback no matter how much negative reactivity was initially added to the core during the drop. Thus, the quantification of thermal margin degradation associated with the single CEA drop is performed via a conversion of the predicted radial distortion factor to thermal margin units in accordance with approved methodology.

NRC Question 28.c:

Provide a plot of the acceptance criteria (MDNBR and linear heat generation rate) vs. time for this event.

APS Response:

The requested plots for MDNBR and LHGR are contained in Figures 28.c-1 and 28.c-2.

The responses of the MDNBR and LHGR are shown for 15 minutes following the single CEA drop. After 15 minutes, the operators will act to reduce core power in accordance with Section 3.1.5, "Control Element Assembly (CEA) Alignment," of the COLR to maintain thermal margin.

Figure 28.c-1: CEA Drop DNBR vs. Time

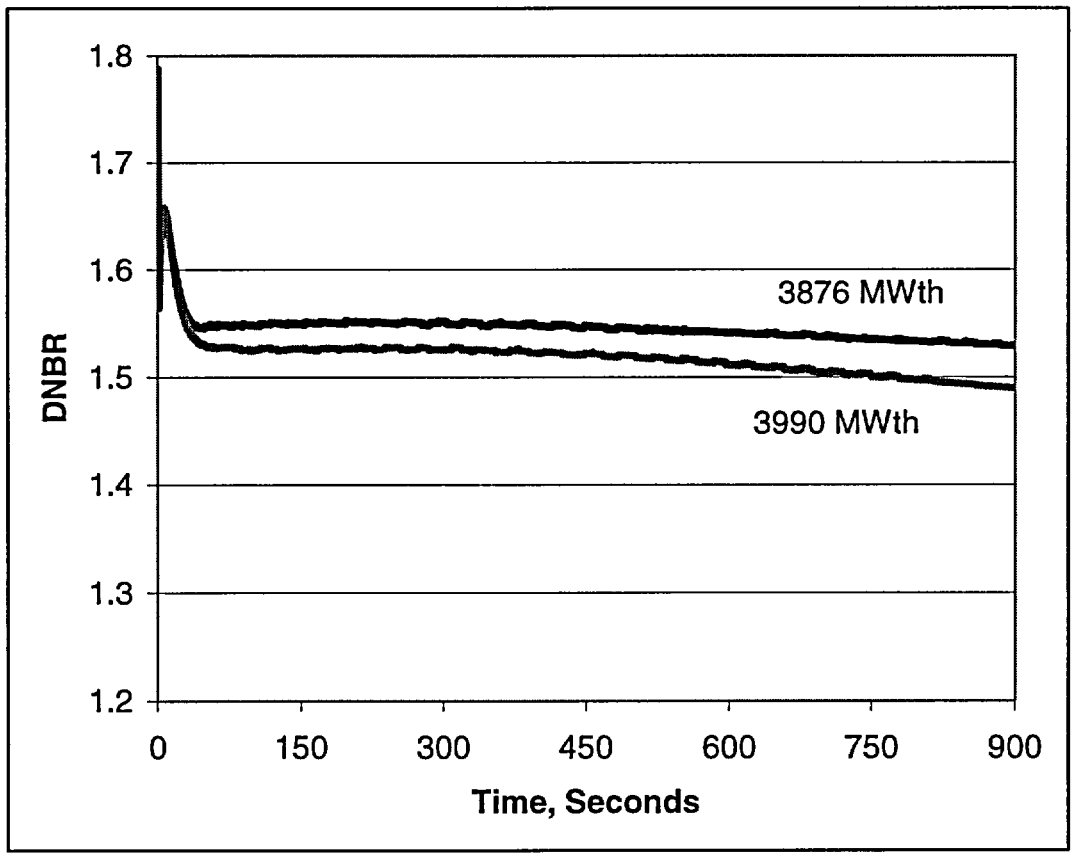
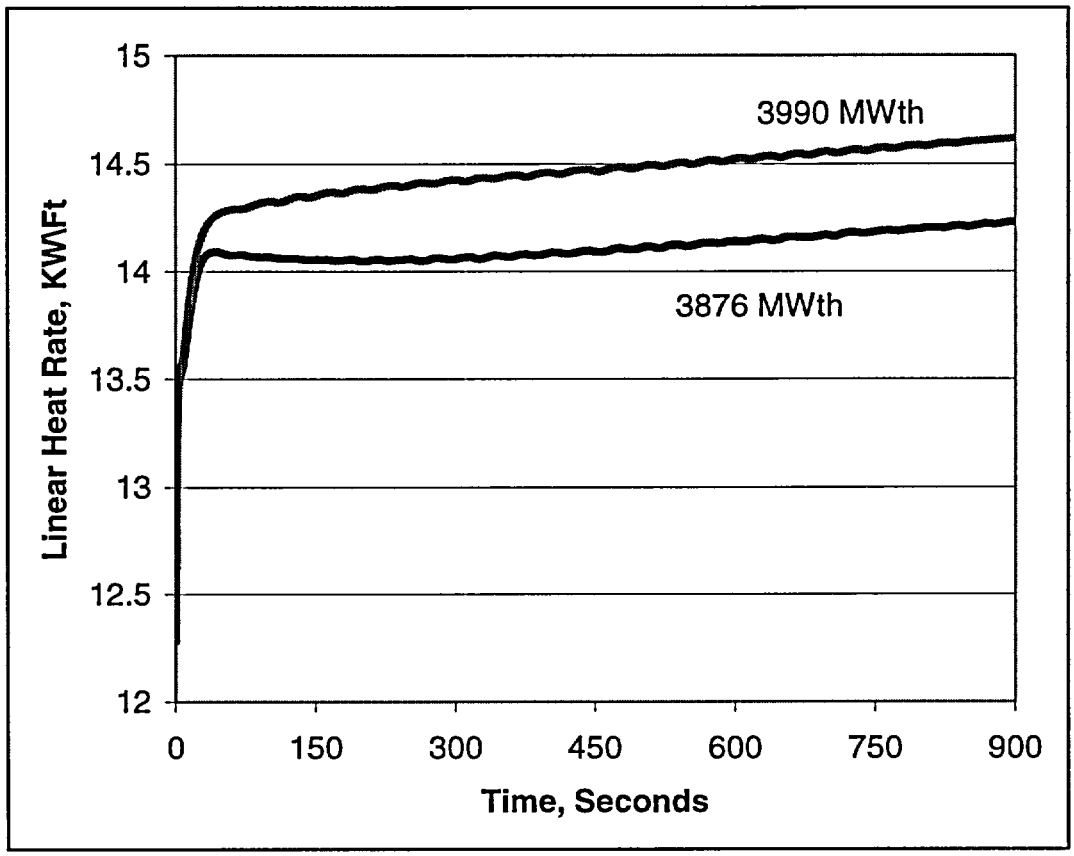


Figure 28.c-2: CEA Drop LHR vs. Time



NRC Question 29:

Attachment 6, Section 6.3.4.8 - CEA Ejection Event:

NRC Question 29.a:

For evaluating the fuel performance and peak RCS pressure cases, the licensee assumed that the Control Element Drive Mechanism (CEDM) rupture was plugged by the ejected CEA. Because this is an unusual assumption, is there a design feature of this plant that justifies this assumption? Does this assumption provide the limiting results for the fuel performance case? Also, what break size (ft^2) was assumed in the analysis?

APS Response:

One of the acceptance criteria for the CEA ejection event (per SRP Section 15.4.8) is the evaluation of peak RCS pressure. The methodology for peak RCS pressure (per Regulatory Guide 1.77, 'Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors', Revision 0, May 1974, Appendix A, Item 13) is that " No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing." Thus, despite the fact that there is no design feature of the plant that justifies this assumption of CEDM rupture plugged by the ejected CEA, the plugged rupture assumption is part of the NRC approved methodology and is consistent with Regulatory Guide 1.77 guidelines. This assumption applies only to the evaluation of peak RCS pressure.

The fuel performance evaluation of the CEA ejection event models the power excursion due to the CEA being ejected. This case uses minimum RCS pressure as the initial condition, and only the hot pin and the average pin are modeled, without taking the credit for the NSSS response since the power excursion occurs fast relative to the NSSS response. The CEDM being ruptured does not impact the fuel performance results.

The break size, however, affects the dose consequence evaluation of the CEA ejection event. The primary system mass/energy releases and the RCS depressurization and cooldown that results from the break impacts the secondary system mass/energy releases which affect the radiological consequences. For the dose consequence evaluation of the CEA ejection, a break size of 0.04 ft^2 is assumed. This break size is consistent with the existing methodology.

NRC Question 29.b:

The initial SG level assumed in this analysis is outside the range of initial conditions listed in Table 6.3-2. Provide the technical basis for this assumption, and discuss its impact on the results of the analysis.

APS Response:

The range for the initial SG level given in Table 6.3-2 is defined to conservatively envelope the steady state operational range by assuming that the initial SG levels are at the high or low SG level alarm setpoints (including the uncertainties associated with the instrumentation). The analysis for the CEA ejection event RCS peak pressure case assumed a lower initial SG level than the lower limit given in Table 6.3-2 by setting the initial SG level at the low SG level trip (LSGLT) setpoint (including the uncertainties associated with the instrumentation). However, neither Table 6.3-2 nor the AOR was revised to eliminate this difference since it is conservative (a lower SG level minimizes the heat removal by the secondary system maximizing the RCS peak pressure), and it does not impact the results significantly (the RCS peak pressure occurs quickly in this event, prior to secondary system heat removal effects becoming significant).

NRC Question 29.c:

Discuss the methodology used to determine the ejected CEA worth.

APS Response:

The most limiting ejected CEA worth is calculated using approved physics codes as defined in the COLR, in accordance with the reload methodology for each reload cycle. All CEAs allowed in the core by the COLR are surveyed at the extreme ends of the operating cycle to determine the most worthy ejected CEA. This is done at 20%, 50%, 65%, and 95% of rated power conditions. The worth calculations take into account the CEA Power Dependent Insertion Limits (PDIL), ASI range, allowable azimuthal tilt, calculational uncertainty, and possible monitoring modes (COLSS in service or COLSS out of service). In addition, since a VOPT trip condition is reached within a few seconds of the limiting CEA ejection, the worth calculations do not take credit for thermal reactivity feedback effects.

NRC Question 29.d:

The UFSAR analysis of record for this event assumed a coincident loss of offsite power, while the PUR analyses do not. Provide the technical basis for changing the licensing basis for this case. Discuss the impact that this assumption would have on the PUR results for this event.

APS Response:

As explained in response to the Question 29.a, the CEA ejection event consists of three parts: peak RCS pressure evaluation, fuel performance evaluation, and the dose consequence evaluation. A loss of offsite power (LOP) does not adversely affect the results of the peak RCS pressure and fuel performance cases. On the other hand, the dose consequence of a CEA ejection event is adversely impacted due to the unavailability of the main condenser, and the increased steam release to the atmosphere. Therefore, a LOP is not modeled for the peak RCS pressure and the fuel performance cases. However, a LOP is assumed to occur for the CEA ejection radiological consequence case, which is consistent with the methodology described in the UFSAR Section 15.4.8 and Section 6.4.4.1 of the PURLR.

NRC Question 29.e:

For the fuel performance case, provide a table of the initial parameter assumptions, sequence of events, and corresponding results.

APS Response:

The initial parameters used for the CEA Ejection fuel performance case, and the sequences of events are presented in Tables 29.e-1, and 29.e-2, respectively. The sequence of events table also presents the results for the maximum average fuel enthalpy, maximum clad surface temperature, and the maximum fuel centerline temperature. These tables represent typical values for the CEA Ejection fuel performance case and are verified to remain bounding for each reload cycle core design.

Table 29.e-1: Initial Parameters Used for the CEA Ejection Fuel Performance Case

PARAMETER	VALUE (see note 2)	
	3990 MW_t	3876 MW_t
Initial Core Power (% of rated)	102	102
Initial Core Inlet Temp (°F)	566	562
Initial Pressurizer Pressure (psia)	2100	2100
Initial RCS Flow (% of design)	95	95
MTC ($x10^{-4}$ $\Delta\rho$/°F)	0.0	0.0
FTC	least negative	Least negative
Kinetics	minimum β	Minimum β
Ejected CEA Worth (%$\Delta\rho$)	0.131	0.131
Postulated CEA Ejection Time (sec)	0.05	0.05
Scram Worth at Trip, N-2 (%$\Delta\rho$)	5.5	5.5
Gap conductance (Btu/hr-ft^2-°F)	See note 1	See note 1

Note 1: For the CEA Ejection fuel performance case, the STRIKIN-II code solves (radially) the one-dimensional cylindrical heat conduction equation for each axial region along the hot rod. The conduction model explicitly represents the gas gap region and dynamically calculates the gap conductance in each axial region (please also refer to the response to Question 18).

2. Performed for 20%, 50%, 65%, and 100% of rated power conditions accounting for power uncertainty. Values present the limiting case (100% power level) for expected PUR core design.

Table 29.e-2: Sequence of Events for CEA Ejection Fuel Performance Case		
Time (sec)	Event	Value
0.00	Mechanical failure of CEDM causes CEA to eject.	---
0.04	High power trip setpoint (fractional)*	1.29
0.05	CEA fully ejected.	---
0.1	Maximum core power (fractional)	1.49
0.7	Reactor trip breakers open	---
1.3	Scram CEAs begin falling.	---
2.6	Maximum radially average fuel enthalpy in the hot node (cal/gm)	141
3.3	Maximum clad surface temperature in the hot node (°F)	1086
3.4	Maximum fuel centerline temperature in the hot node (°F)	4808

Note *: Since STRIKIN-II cannot model the CPC VOPT trip, the VOPT trip may be modeled as an ordinary high power trip.

NRC Question 30:

Section 6.3.4.1 of Attachment 6: The section describes the analysis of uncontrolled CEA withdrawal from a subcritical or low power condition. Sections 6.3.4.1.3.3 and 6.3.4.1.4.3, respectively, describe the results of analyses for the CEA withdrawal from a subcritical and low power conditions, and state that the hot channel minimum DNBR remains above the safety limit, and that the linear heat generation rate (LHGR) exceeded the safety limit as defined in the Technical Specifications (TSs). This occurs for a short time with a resulting peak fuel temperature well below the limiting fuel centerline temperature for melting fuel. The NRC staff agrees that there is no safety concern as the resulting peak fuel temperature is below the fuel melt limit. However, violation of a TS safety limit even for a very short time is not acceptable, as this is a violation of 10 CFR 50.36, which requires that TS Limiting Safety System Settings be in place to prevent safety limits from being exceeded during analyzed events.

It has recently been identified in a number of TSs of CE designed plants that the peak linear heat rate (PLHR) safety limit is violated for the uncontrolled CEA withdrawal from subcritical or low power conditions. Because the PLHR safety limit is defined as a measure to prevent fuel centerline temperature from reaching the fuel melt temperature, some licensees have resolved this issue by converting from a PLHR safety limit to a fuel centerline temperature safety limit. This issue must be resolved prior to NRC staff approval of the proposed Unit 2 PUR amendment. Discuss how you propose to resolve this issue.

APS Response:

APS letter 102-04836-CDM/TNW/RAB to USNRC, "Palo Verde Nuclear Generating Station (PVNGS) Units 1,2, and 3, Docket Nos. STN-528/529/530 Request for a License Amendment to Revise the Peak Linear Heat Rate Safety Limit, Technical Specification 2.1.1.2," dated September 6, 2002, requested a license amendment to change the safety limit to a peak fuel centerline temperature safety limit.

NRC Question 31:

Confirm that the generically approved LOCA analysis methodologies used for the Palo Verde uprate LOCA analyses apply specifically to Unit 2, and:

APS Response:

The methodologies used in the LOCA analyses are the same as those identified and approved for use in the Unit 2 COLR.

NRC Question 31.a:

Identify all restrictions placed on the LBLOCA and SBLOCA models and show how they are resolved for this uprate. Specifically, the NRC staff wants to understand the resolution of the SER limitation to 3800 MW_t and the process used to justify by extension of the methodology to power levels in excess of 3800 MW_t.

APS Response:

Enclosure 1 list the SER limitations/constraints imposed by the SERs for the 1985 EM and S1M versions for the LBLOCA and SBLOCA evaluation models, respectively. The response also lists how the LBLOCA and SBLOCA PUR analyses address the limitations/constraints. This enclosure is Westinghouse Electric Co. proprietary information and should be withheld from public disclosure pursuant to 10 CFR 2.790.

The following discussion addresses the resolution of the SER limitation that limits the acceptance of the LBLOCA and SBLOCA evaluation models to plants with power ratings up to 3800 MW_t.

The applicability of the LBLOCA and SBLOCA evaluation models to plants with power ratings up to 3800 MW_t is based on the core power level limit set by Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," December 1973. Regulatory Guide 1.49 states that licensed power levels should be limited to a reactor core power level of 3800 MW_t or less until January 1, 1979, at the earliest. The intent of this regulatory guidance was to stabilize the maximum size of nuclear plants until sufficient experience was gained with design, construction, and operation of large plants.

As described in the 1996 stretch power license amendment request (Reference 1) and the associated SER (Reference 2), the staff has reviewed the operating experiences of large plants since the issuance of Regulatory Guide 1.49 in 1973. Furthermore, based on these reviews, the staff has concluded that sufficient experience exists to license plants with powers that are in excess of the administrative limit of 3800 MW_t and, subsequently, has granted such licenses. For example, the NRC licensed Combustion Engineering System 80+ standard design for a power level of 3914 MW_t (Reference 3). Thus, Reference 2 justifies that the license amendment request for power levels in excess of 3800 MW_t is approved for PVNGS.

NRC Question 31.b:

Address all other restrictions on, conditions of applicability, or changes to the Palo Verde LBLOCA and SBLOCA methodologies by NRC staff SERs or other report findings (e.g., Part 21 or 10 CFR 50.59 reports, error adjustments, etc) in a table, listing the item in sufficient detail to identify the concern, the date of the SER or other report, and its disposition (e.g., not applicable to Palo Verde, reanalysis reflecting error correction, or Palo Verde is in the applicable class and how the condition is satisfied for Palo Verde).

APS Response:

Enclosure 1 lists the SER limitations/constraints imposed by the SERs for the 1985 EM and S1M versions for the LBLOCA and SBLOCA evaluation models, respectively. The response also lists how the LBLOCA and SBLOCA PUR analyses address the limitations/constraints. This enclosure is Westinghouse Electric Co. proprietary information and should be withheld from public disclosure pursuant to 10 CFR 2.790.

There are no other restrictions and conditions of applicability to the LBLOCA or SBLOCA methodologies applicable to the PUR ECCS performance analyses. The changes that have been made to the methodologies since the latest review of PVNGS ECCS performance analysis by the NRC in 1995, are explained below:

LBLOCA: There has been one error correction to the 1985 EM methodology that impacted the peak cladding temperature. The error was in the decay heat energy redistribution factor for voided conditions in the model. APS informed the NRC of the error in Reference 4. Decreasing the peak linear heat generation rate by 0.2 kW/ft compensated for the impact of the error. In addition, since 1995, changes/improvements have been made to computer operating systems and hardware. These changes have less than a 1 °F impact on peak cladding temperature.

SBLOCA: There has been one error correction to the S1M methodology. This error was in the coding in the CEFLASH component. Please refer to the responses to Questions 3 and 33 for the detailed description of the error and the resolution.

NRC Question 31.c:

Show that Unit 2, operating at the uprated power is bounded by the assumptions used in the analyses to support the approval of the generic LOCA methodologies.

APS Response:

As described in the following paragraphs, a rated core power of 3990 MW_t (4070 MW_t with a 2% power measurement uncertainty) is inside the range of applicability of the generic LOCA methodologies (evaluation models). Additionally, the values for some core power related parameters, including the hot rod PLHGR, are bounded by values that have been used in previous applications of the evaluation models.

The values for core power related parameters at PUR operating conditions represent, at most, an incremental increase, and for some parameters a decrease, from the originally licensed values for Unit 2 Cycle 1 operating conditions. For example, in the Cycle 1 ECCS performance analysis, the Average Linear Heat Generation Rate (ALHGR) was 5.642 kW/ft. It was based on a core power of 3876 MW_t (including the 2% power measurement uncertainty) and a core design that included approximately 1900 B_4C shims. The B_4C shims have been replaced with integral burnable absorber fuel rods in subsequent core designs. Consequently, at an uprated core power of 4070 MW_t, the ALHGR is 5.735 kW/ft. This is only a 1.6% increase from the Cycle 1 value. It is also noted that the 16x16 fuel assemblies utilized at PVNGS result in an ALHGR that is significantly less than that of Combustion Engineering designed PWRs that use 14x14 fuel assemblies. The ALHGR for the 14x14 fuel assembly plant that was analyzed in the sample break spectrum analysis was 6.2 kW/ft, as presented in the topical report CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of CE and W designed NSSS," which describes the original version of the LBLOCA evaluation model.

With respect to the hot rod PLHGR, the Cycle 1 LBLOCA and SBLOCA analyses were performed using PLHGR of 14.0 kW/ft and 15.0 kW/ft, respectively. The PUR LBLOCA and SBLOCA analyses are performed using PLHGRs of 13.1 kW/ft and 13.5 kW/ft, i.e., significantly lower values than the Cycle 1 values. Since the linear heat rates used in the PUR are bounded by those used in previous analyses, parameters such as initial stored energy and the decay heat power are also bounded by the parameters used in previous applications of the NRC approved evaluation models. Therefore, the LOCA methodologies are used within their range of application for the PUR.

The RCS blowdown thermal hydraulic codes used in the LBLOCA and SBLOCA analyses (CEFLASH-4A and CEFLASH-4AS, respectively) are generalized node/flow path codes. Their core models represent the average core power as well as the hot assembly power (for LBLOCA). The core heat transfer models are essentially the same models that are used in the STRIKIN-II hot rod heatup code to represent the hot rod. Consequently, the RCS blowdown thermal hydraulic codes are applicable for core powers in excess of the core average and hot assembly power levels associated with the PUR.

In summary, there are no features of the power related models used in the LBLOCA and SBLOCA evaluation that limit the applicability of the evaluation models to powers less than 3800 MW_t. As described in the response to Question 31.a, the SER limitation to a maximum core power of 3800 MW_t is not based on any inherent limitation in the evaluation models. Additionally, no changes other than what is described in the responses to Questions 3 and 33 have been made to the evaluation models or to the application of the evaluation models in performing the PUR analysis.

As described above, the generic LOCA evaluation models are applicable to a range of powers that bounds the uprated power. However, "the assumptions used in the analyses to support the approval of the generic LOCA methodologies" (e.g., scoping studies, parametric studies, sample spectrum analyses that are documented in the evaluation model topical reports) are selected to typify, not bound, the Combustion Engineering fleet of PWRs. Use of typical versus bounding assumptions is appropriate because of the similarity of design among the plants in the Combustion Engineering fleet of PWRs. For example, all currently operating plants in the fleet have a 2x4 loop configuration (two hot legs and four cold legs) and have similarly designed safety injection (SI) systems. Despite the large range in core powers, in the context of ECCS performance, all the plants in the fleet operate with similar core and fuel related parameters (e.g., power distributions and fuel stored energies). The evaluation models explicitly address the variations in design that do exist (e.g., 14x14 versus 16x16 fuel assemblies, UO_2 versus urania-erbia fuel pellets, Zircaloy-4 versus ZIRLO® cladding).

For the reasons described above, the Westinghouse evaluation models for Combustion Engineering designed PWRs are applicable for a range of core powers, including an uprated core power of 3990 MW_t (4070 MW_t with a 2% power measurement uncertainty).

NRC Question 31.d:

Provide a comprehensive statement that Palo Verde and its vendor 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, and briefly discuss these processes. (Do not use specific procedures and process components as examples in your response.)

APS Response:

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.

Specifically, APS and Westinghouse use an ECCS performance analysis checklist to provide the assurance that the analysis of record (AOR) bounds the plant configuration parameters prior to every reload cycle. This checklist tabulates the configuration value

(i.e., the as-operated plant value) and the value supported by analysis (i.e., the LOCA analysis input value) for all peak cladding temperature-sensitive parameters. Additionally, the checklist identifies the relationship (i.e., the rule) between the configuration value and the value supported by analysis that must be obeyed in order for the ECCS performance analysis to be applicable to the new reload cycle. APS is responsible for the configuration values while Westinghouse is responsible for the values supported by analysis and defining the rules. Prior to every reload cycle, APS confirms that all the configuration values for the upcoming cycle obey their rules for the current values supported by the analyses. In general, if a configuration value is changed due to plant design or the new core design changes and does not obey its rule, Westinghouse revises the ECCS performance analysis, as appropriate, so that the rule is obeyed given the new configuration value. As part of the process, Westinghouse updates the impacted value(s) supported by analysis in the checklist to reflect the new value(s).

References for Question 31:

1. 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.
2. Letter, C.R. Thomas (NRC) to W.L. Stewart (APS), "Issuance of Amendments for the Palo Verde Nuclear Generating Station Unit No. 1 (TAC No. M94541), Unit No. 2 (TAC No. M94542, and Unit No. 3 (TAC No. M94543)," May 23, 1996.
3. NUREG-1462, Vol. 1, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design Docket No. 52-002," August 1994.
4. Letter, S.A. Bauer (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528/529/530, Error in Energy Redistribution Factors Used in LOCA/ECCS Performance Evaluation Models, 30 Day 10CFR50.46 Report," September 14, 1997.

NRC Question 32:

What are the calculated peak cladding temperature (PCT) and oxidation values for the LBLOCA and SBLOCA per 10 CFR 50.46(b), for Palo Verde at the uprated power? Will this calculation become the official analysis of record for future reporting under Section 10 CFR 50.46?

APS Response:

As explained in the response to Question 3, no new analyses were performed for uprated power since the existing analyses were performed for a rated core power of 3990 MW_t. The existing analyses calculated a peak cladding temperature of 2174 °F at

a PLHGR of 13.1 kW/ft for LBLOCA, and a peak cladding temperature of 1907 °F at a PLHGR of 13.5 kW/ft for SBLOCA. For the LBLOCA, a core-wide cladding oxidation of less than 0.86%, and a maximum cladding oxidation of 8.37% were calculated. For SBLOCA, the core-wide cladding oxidation was less than 0.57%, and maximum cladding oxidation was 3.57%.

After the proposed amendment for the PUR was submitted to the NRC, LBLOCA and SBLOCA ECCS performance was reanalyzed. The reanalyses were performed using the latest NRC accepted versions of the Westinghouse evaluation models for Combustion Engineering designed NSSS (References 1 and 2). In addition, the new analyses explicitly model the ZIRLO® fuel design (Reference 3) that was introduced in Unit 2 during operating cycle 11 which began operation in April 2002. These analyses were reported to the NRC in Reference 4, and will become the official AOR for future reporting. These analyses were performed at a rated core power of 3990 MW_t, and are applicable for operation at PUR conditions. A comparison of the results is provided in Tables 32-1 and 32-2.

Table 32-1: Comparison of Important Results of the Current and New PVNGS Units 1, 2 and 3 LBLOCA ECCS Performance Analysis

Parameter	Current Analysis	New Analysis [b]	New Analysis [c]
Limiting Break Size	0.6 ft^2 DEG/PD [a]	0.6 ft^2 DEG/PD	0.8 ft^2 DEG/PD
Cladding Material	Zircaloy	Zircaloy	ZIRLO®
Peak Cladding Temperature, °F	2174	2110	2087
Time of Peak Cladding Temperature, seconds	250	266	232
Maximum Cladding Oxidation, %	8.37	7.6	12.0
Maximum Core-Wide Cladding Oxidation, %	<0.86	<0.57	<0.73
Time of Cladding Rupture, seconds	36	48	26

a. DEG/PD = Double-Ended Guillotine Break in Pump Discharge Leg
b. Case of Maximum Cladding Temperature
c. Case of Maximum Local Cladding Oxidation

Table 32-2: Comparison of Important Results of the Current and New PVNGS Units 1, 2 and 3 SBLOCA ECCS Performance Analysis		
Parameter	Current Analysis	New Analysis
Limiting Break Size [a]	0.05 ft^2/PD [b]	0.05 ft^2/PD [b]
Peak Cladding Temperature, °F	1907	1618
Time of Peak Cladding Temperature, seconds	1568	1592
Maximum Cladding Oxidation, %	3.57	1.28
Maximum Core-Wide Cladding Oxidation, %	<0.57	<0.2
Time of Cladding Rupture, seconds	No Rupture	No Rupture
a. Break that resulted in the highest peak cladding temperature b. PD = Pump Discharge Leg		

References for Question 32:

1. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.
2. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.
3. CENPD-404-P-A, Revision 0, "Implementation of ZIRLO® Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.
4. APS Letter 102-04699-CDM/TNW/JAP, David Mauldin (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528/529/530, 10CFR50.46 (a)(3)(ii) 30 Day Report for Changes in LOCA/ECCS Performance Evaluation Models," May 03, 2002.

NRC Question 33:

Discuss the effect on the Unit 2 SBLOCA analyses of the recently discovered coding error in the CEFLASH component of the SBLOCA methodology. Provide both PCT and oxidation before and after results.

APS Response:

The recently discovered coding error occurred in the CEFLASH-4AS computer code. CEFLASH-4AS performs the thermal hydraulic analysis of the RCS in the SBLOCA evaluation models for Combustion Engineering designed PWRs. It generates the boundary conditions for core power, pressure, two-phase mixture level, and liquid mass that are used in the hot rod heatup analysis to calculate the peak cladding temperature.

It was discovered that subroutine LEAK, which performs break flow calculations, contained coding that performed operations using subscripts for arrays that exceeded the dimensions of the arrays. In particular, the critical flow tables were copied to arrays

that were dimensioned smaller than the indices of the "do loops" that performed the copy operations. Consequently, data in the arrays were incorrectly ordered and overwritten. In addition, values for variables in subroutine MATPRP, which calculates fuel rod material properties, were also overwritten.

The inconsistencies between the array dimensions and the subscripts were corrected so that all operations are performed with array subscripts that are consistent with the dimensions of the arrays. The code was tested for the occurrence of out-of-range subscripts to confirm that all inconsistencies were corrected.

Calculations were performed with the corrected and erroneous versions of CEFLASH-4AS for all Combustion Engineering designed PWRs impacted by the error. The calculations showed that correcting the error resulted in changes to peak cladding temperature ranging from –38 °F to +43 °F. For PVNGS, correcting the error resulted in a decrease in peak cladding temperature for the limiting SBLOCA. In particular, the calculation with the corrected version of CEFLASH-4AS resulted in a 31 °F decrease in peak cladding temperature, a 0.24% decrease in maximum cladding oxidation, and a 0.04% decrease in core-wide cladding oxidation.

Note that the hot rod heatup portion of the corrected analysis performed for PVNGS did not include a discretionary conservatism that was used in the analysis of peak cladding temperature reported in Section 6.3 of the UFSAR. The discretionary conservatism was added in the UFSAR analysis to create a bounding result. The analysis performed to determine the impact of the error was performed without this discretionary conservatism in order to provide a true prediction of the impact of the error on the analysis.

NRC Question 34:

Discuss the design of the Palo Verde emergency core cooling system (ECCS) switchover from the injection mode to the ECCS sump recirculation mode. What was the decay heat source assumed in the design of the ECCS switchover for the present power? Discuss if the assumed heat source and the timing of the switchover change for the uprated power.

APS Response:

The Palo Verde ECCS is designed to take suction from the Refueling Water Tank (RWT) during the injection mode of operation following a design basis accident. ECCS pump suction is automatically realigned to take suction from the containment sumps (recirculation mode) when the inventory in the RWT is depleted. The switchover occurs when the RWT water level drops to approximately 7% of the indicated span, which generates a Recirculation Actuation Signal (RAS).

The suction switch-over at RAS is designed to occur at the point where the design minimum flow from a single High Pressure Safety Injection (HPSI) pump, with suction aligned to the containment sump, is sufficient to match core boil off and maintain the

core covered. Consequently, sufficient inventory in the RWT is made available to sustain the injection mode until such time that the recirculation mode can be supported (i.e., minimum time to RAS).

The calculation of the minimum time to RAS is predicated on core decay heat that, for PVNGS, is based on the decay heat model of ANSI 5.1-1971. Additional conservatism has been added to the decay heat fractions as prescribed by the standard including a 2% margin for initial power measurement and a 10% margin for decay heat fractions at times greater than 1000 seconds.

Core boil off (mass/time) is subsequently calculated using the ANSI decay heat data and conservatively assuming that the coolant is saturated at the maximum containment pressure conditions. The resulting boil-off rate is compared to the mass addition rate of a single train of HPSI. The minimum time to RAS is established as the required time after the event to where the rates are equal. The resulting RWT volume requirements are calculated using the minimum time to RAS multiplied by the appropriate ECCS flow rates (typically run-out). PVNGS surveillance requirements for the RWT ensure that this volume is maintained during power operation.

The RWT volume requirements for the power up-rate have been evaluated. Increased decay heat increases the time following the accident after which a single train of HPSI can provide sufficient inventory to match boil-off. The increased minimum time to RAS results in requiring a slightly greater volume within the RWT which is dedicated to Emergency Safety Feature (ESF) operation. The resultant ESF reserved volume is well within the RWT volume currently required by the Technical Specifications. Consequently, the proposed power up-rate is considered acceptable relative to the RWT volume requirements and ECCS operation at suction switch-over.

NRC Question 35:

Provide a complete description of the long term cooling, boron precipitation model that is used to establish compliance with 50.46(b)(5).

APS Response:

Enclosure 1, includes the answer to Question 35. This enclosure is Westinghouse Electric Co. proprietary information and should be withheld from public disclosure pursuant to 10 CFR 2.790.

Attachment 3
Affidavit from Westinghouse Electric Corporation
Submitted Pursuant to 10 CFR 2.790 to Consider
Tables 31.a-1 Through 31.a-9
SER Limitations/Constraints
Associated with the LBLOCA and SBLOCA Evaluation Models
Used for the PUR ECCS Performance Analysis
and NRC Question 35 on the Description of the
Long Term Cooling, Boron Precipitation Model
as Proprietary Information

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

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

- Enclosure 1 to WEC letter LTR-OA-02-101, "Responses to PVNGS Unit 2 Power Uprate License Report RAIs Related to ECCS Performance," August 5, 2002

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

1. The information sought to be withheld from public disclosure is owned and has been held in confidence by WEC. It consists of details of how WEC's LOCA methodology complies with SER constraints and a description of aspects of WEC's post-LOCA long-term cooling model and analysis.
2. The information consists of analyses or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to WEC.
3. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements that provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of WEC because:
   a. A similar product or service is provided by major competitors of WEC.
   b. WEC has invested substantial funds and engineering resources in the development of this information. A competitor would incur similar expense in generating equivalent information.

c. The information consists of details of how WEC's LOCA methodology complies with SER constraints and a description of aspects of WEC's post-LOCA long-term cooling model and analysis. The availability of such information to competitors would enable them to design their product or service to better compete with WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.

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

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

Ian C. Rickard
Licensing Project Manager
Westinghouse Electric Company LLC

Sworn to before me this 5th day of August, 2002

Notary Public

My commission expires: May 31, 2003

Attachment 4

Replacement Pages for Power Uprate Licensing Report Section 6.3.0.3.1

Table 6.3-3
RPS Analytical Setpoints Credited in the Transient Analysis

(Page 2 of 2)

Parameter	Units	Analytical Setpoint	Response Time
CPC Asymmetric SG Transient (ASGT) ΔT normal harsh	°F	15.0 N/A	0.75 sec
CPC T_{hot} saturation normal harsh	°F	8.0 N/A	1.0 sec

Notes: (1) SG ΔP low RCS flow trip setpoints (i.e., ramp, step, and floor) selected to ensure breakers open at 2.5 seconds for a sheared shaft event. This is the only event that credits this trip function.

(2) Value in parenthesis represents the analysis setpoint for the existing plant thermal power output (i.e., 3876 MW_t).

Section 6.3.0.3.1 CPC Dynamic Signal Filter Coefficients

The CPC power and temperature signal filters, along with any associated correction factors, were evaluated to demonstrate their continued acceptability at PUR conditions. These signal filters were reviewed for the set of AOOs that provide the greatest challenge to the SAFDLs. This analysis confirms that the filtered CPC responses (and corresponding correction factors), in conjunction with other reactor protection trips and the initial margin preserved by the plant LCOs, result in a CPC reactor trip before the DNB and centerline-to-melt SAFDLs are violated.

The CPC Dynamic Signal Filter evaluation consisted of evaluating the CPC response for a spectrum of AOOs that increase core power, change coolant temperature, and reduce primary system pressure.

Section 6.3.0.3.1.1 Increasing Power Signal Filters

Of the increasing power AOOs (boron dilution, excess loads, single and bank CEAWs), the bank CEAW is examined because it results in the fastest rate of increasing power of all of the CPC's DBEs. The increasing neutron flux signal filter is shown to be conservative by comparing the response of the CPC's compensated core average heat flux to the actual rate of core heat flux increase. For the neutron flux signal filter to be conservative, the CPC's calibrated heat flux should be greater than the actual core heat flux.

To ensure that the CPC calibrated core heat flux conservatively leads actual core heat flux, a thermal margin calculation is performed to ensure that there is enough margin available at time of the trip to prevent the SAFDLs from being violated. The CEAW events were initiated at different power levels along with conservative initial conditions (i.e., bank worth, MTC, trip response, etc.)

Section 6.3.0.3.1.2 Increasing Reactor Coolant System Temperature Signal Filters

The CPC signal filters provide two RCS temperature indications, T_{CMIN} and T_{CMAX} that are used in the protective calculations. T_{CMAX} is used as the core inlet temperature for the CPC's DNBR calculation. To be conservative, the value of T_{CMAX} should be larger than the actual core inlet temperature. T_{CMIN} is used to adjust the raw neutron power signal for reactor vessel downcomer temperature shadowing. To be conservative, the value of T_{CMIN} should be less than the actual RCS temperature existing in the downcomer region of the reactor vessel at any given time in the transient.

The CPC's lead and lag temperature signal filters take the cold leg Resistance Temperature Detector (RTD) signals and adjust them to obtain the desired conservative direction of T_{CMIN} and T_{CMAX}.

Increasing temperature, in the RCS, that is not driven by a power increase can be caused by decreased heat removal events, such as a LOFW or a loss of load. The loss of load is the more adverse heatup scenario and is examined for RCS temperature increases. The loss of load event, however, is typically non-limiting for CPC DNBR responses because of the essentially constant power, and increasing RCS pressure. Consequently, other AOOs that result in an increase in both temperature and power are considered. Specifically, the increase in reactor coolant temperature is evaluated in conjunction with power increases for the CEAW event.

Section 6.3.0.3.1.3 Decreasing Reactor Coolant System Temperature Signal Filters

Decreasing temperature that is not driven by a power decrease can be caused by an excess load event. A spectrum of possible excess loads could be imposed upon the NSSS by the secondary system. An increase in FW flow or a decrease in FW temperature has the potential to increase the heat demand on the RCS. A single ADV or Turbine Bypass Valves (TBVs) has the capability of imposing an increase of rated steam flow. A change in the position of the Turbine Admission Valve (TAV) has the potential to result in larger variations in the steam demand from the initial conditions.

The CPC's lead and lag temperature signal filters take the cold leg RTD signals and adjust them to obtain the desired conservative direction of T_{CMIN} and T_{CMAX}. The excess load event was selected as the CPC's DBE to examine these signal filters for transients resulting in decreasing RCS temperatures.

A spectrum of potential turbine driven increases in steam flow was analyzed to ensure that the signal filters are conservative in the decreasing temperature direction over a spectrum of possible power to load imbalances.

Section 6.3.0.3.1.4 Decreasing Pressure Penalty

A RCS depressurization may be caused by a pressurizer spray malfunction. Failure of the spray system allows the maximum flowrate into the pressurizer. The transient results in a rapid depressurization of the pressurizer and an approach to SAFDL on DNBR.

The RCS depressurization associated with the spray malfunction event was simulated using the CENTS code. Output from the CENTS code was then analyzed and a power equivalent of pressure decrease was calculated to confirm the applicability of the allowance (penalty) for decreasing pressure incorporated into the CPCs.

Section 6.3.0.3.1.5 Results

The CPC transient signal filter analysis was performed to assure that the CPC will conservatively respond to the transients below following PUR:

- the cooldown associated with an excess heat removal event,
- the power increase associated with CEA bank withdrawal, and
- the depressurization associated with a spray malfunction event.

The CPC transient signal filter analysis verifies that the CPC's adjusted process parameters are conservative for the expected values for a given transient event. The CPC coefficients are adjusted as necessary to assure the CPC's action prevents SAFDL violation during these AOOs.

The results of the analysis verified proper response to the significant overcooling, heatup, depressurization, and power increasing transients and conservative CPC actions following PUR. This evaluation ensures that the CPC will provide the necessary trip functions to prevent the SAFDLs from being violated.

Section 6.3.0.4 Engineered Safety Features

The Engineered Safety Features (ESF) systems provide automatic actions to mitigate the consequences of the transient events. Table 6.3-4 lists the ESF analytical setpoints credited in the transient analyses. The analytical setpoints include instrument uncertainties that were applied in a conservative manner intended to increase the severity of a transient. The ESF responses to each transient event are detailed in the event sections.

Attachment 5

Pressurizer Safety Valve Orifice Sizing Correction Factor

Pressurizer Safety Valve Orifice Sizing Correction Factor

Westinghouse has informed APS that during the conversion from CESEC to CENTS for the performance safety analyses, a correction factor was inadvertently omitted from the calculation for pressurizer safety valve sizing. The sizing correction applies to four analyses that have been re-performed. The affected analyses are 1) Loss of Condenser Vacuum, 2) Feedwater Line Break (Peak Pressure Events only), 3) Loss of RCS Flow, and 4) CEA Ejection. Use of the proper correction factor results in slightly less conservative results in peak RCS pressure, however the results continue to meet acceptance criteria.

Enclosure 2 contains updated Sequence of Events Tables and Peak Pressure figures associated with the above listed events. The enclosed Tables and Figures replace the respective existing Tables and Figures in Attachment 6 to Reference 1. Since the revised values are minimal, and changes to the associated figures are also minimal, APS will not provide the remaining figures at this time. All of the corrected figures will be included in the appropriate revision to the UFSAR. Additionally APS will revise the "Overpressure Protection Report for the Palo Verde Nuclear Generating Station."

As noted in the transmittal letter for this attachment, the Large Feedwater Line Break – Long Term Cooling Event re-analysis has been included in Attachment 6 to this letter.

Enclosure 1

Tables 31.a-1 Through 31.a-9
SER Limitations/Constraints Associated with the LBLOCA and SBLOCA
Evaluation Models Used for the PUR ECCS Performance Analysis
and NRC Question 35 on the
Description of the Long Term Cooling, Boron Precipitation Model

Enclosure 2

Revised Tables and Figures for the Pressurizer Safety Valve Orifice Sizing Correction Factor

Index of Revised Tables and Figures

Table 6.3-25
Sequence of Events for LOCV Primary Peak Pressure Case

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOCV, turbine trip, main FW pump trip.	---	---
7.32	7.20	Pressurizer pressure reaches reactor trip setpoint (psia).	2415	2415
7.32	7.20	HPPT signal generated.	---	---
7.82	7.70	Reactor trip breakers open.	---	---
8.42	8.30	Scram CEAs begin falling.	---	---
9.05	8.88	PSVs open (psia).	2549	2549
9.76	9.61	Maximum RCS pressure (psia).	2718	2739
10.1	9.18	MSSV bank 1 open (psia).	1303	1303
13.0	13.0	PSVs close (psia).	2422	2422
12.2	11.3	MSSV bank 2 open (psia).	1344	1344
13.7	13.7	Maximum pressurizer water volume (ft^3).	834	847
14.5	13.4	MSSV bank 3 open (psia).	1370	1370
14.6	13.5	Maximum SG pressure (psia).	1376	1376
18.7	17.8	SG water level reaches AFAS setpoint (%WR).	20	20
18.7	17.8	AFAS generated.	---	---
28.4	28.5	MSSV bank 3 close (psia).	1301	1301
39.2	40.1	MSSV bank 2 close (psia).	1277	1277
>60	>60	MSSV bank 1 close (psia).	1238	1238
>60	>60	AFW flow initiated (gpm).	186	186
1800	1800	Operator initiates the cooldown (min).	30	30

Figure 6.3-71
LOCV Primary Peak Pressure Case-RCS Pressure vs. Time

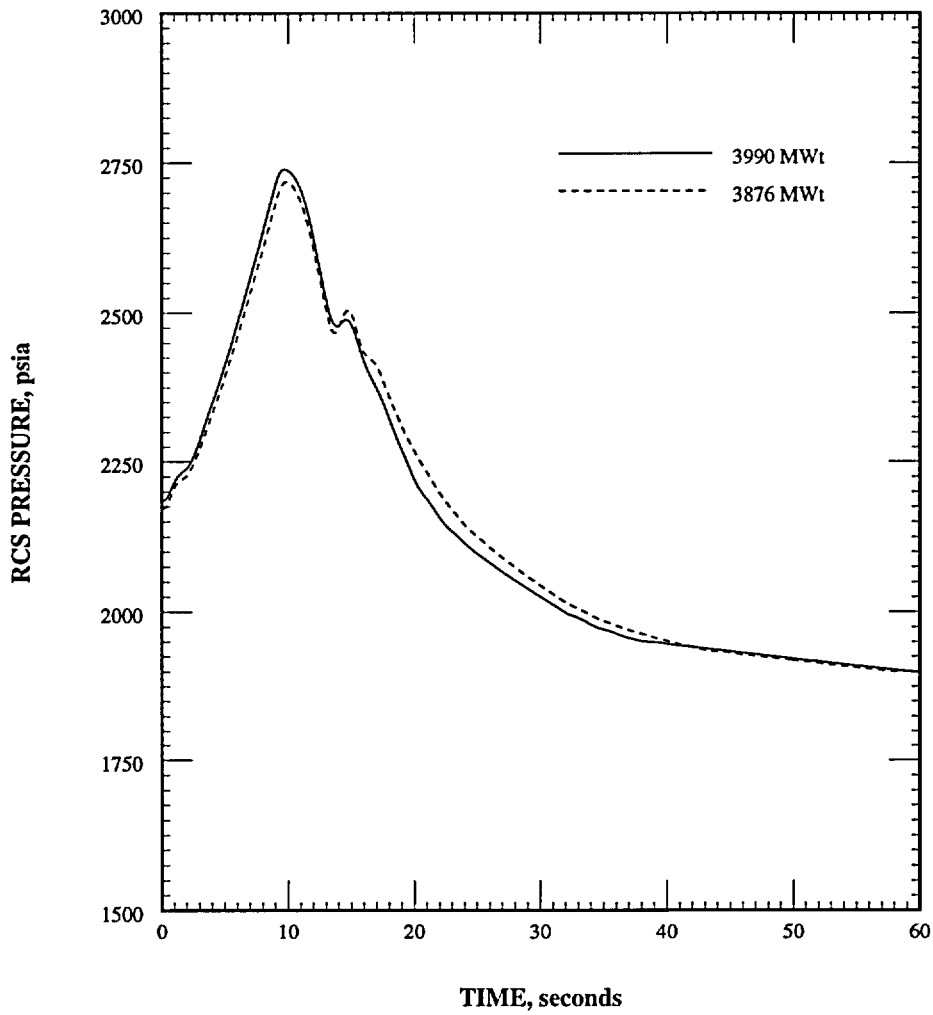


Figure 6.3-72
LOCV Primary Peak Pressure Case-Pressurizer Pressure vs. Time

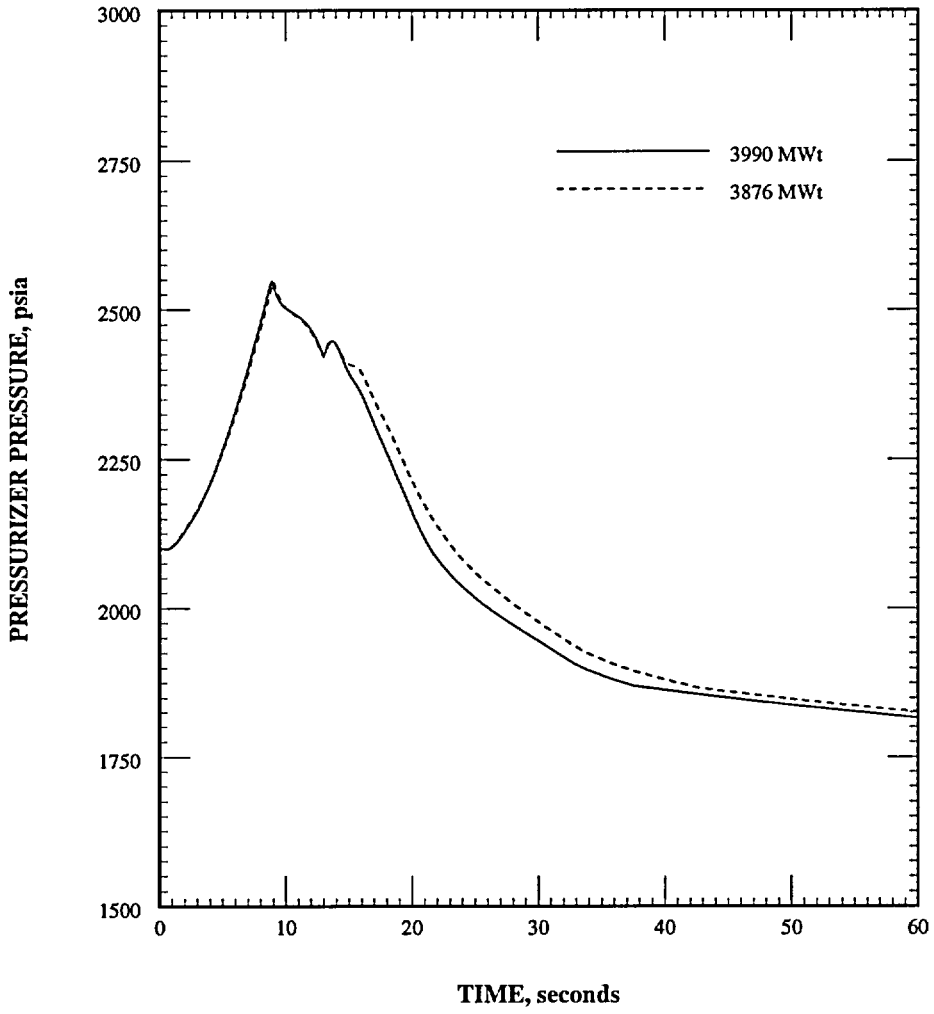


Figure 6.3-74
LOCV Primary Peak Pressure Case-SG Pressure vs. Time

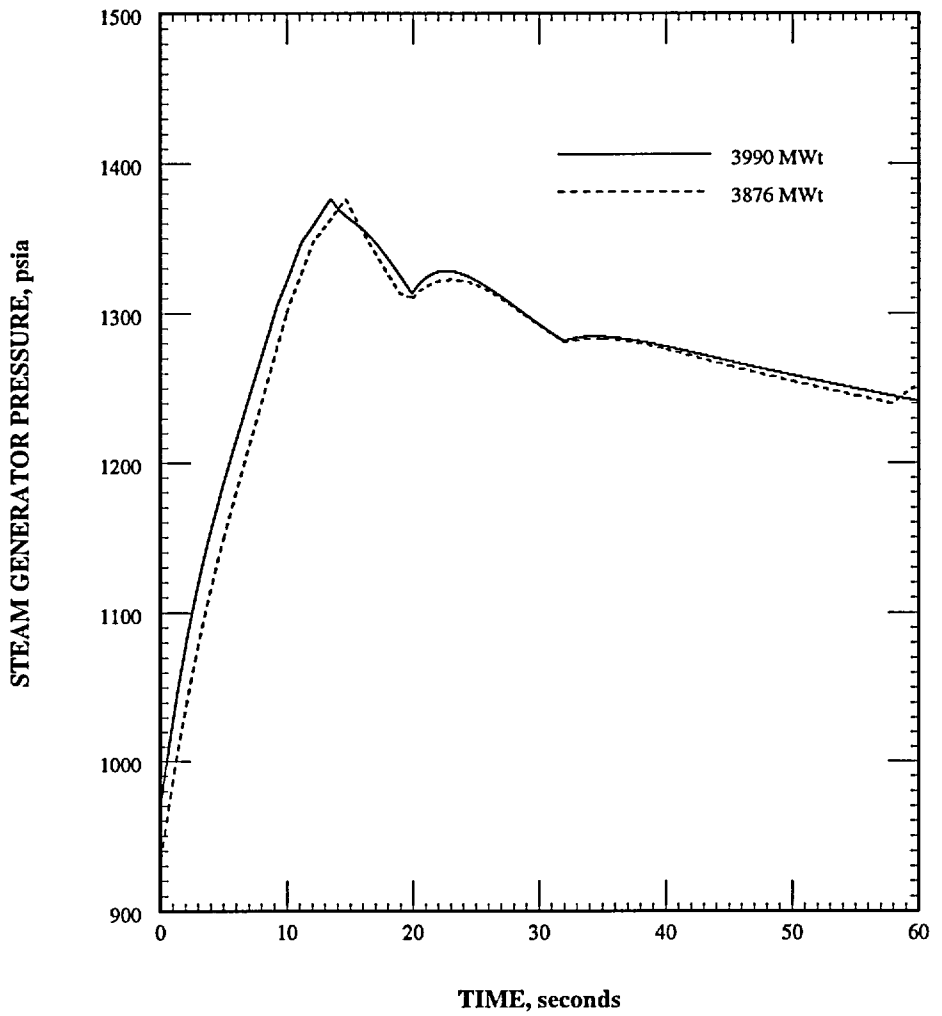


Table 6.3-26
Sequence of Events for LOCV Secondary Peak Pressure Case

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOCV, turbine trip, main FW pump trip.	---	---
6.26	4.17	MSSV bank 1 open (psia).	1303	1303
8.03	5.84	MSSV bank 2 open (psia).	1344	1344
7.81	7.01	Pressurizer pressure reaches reactor trip setpoint (psia).	2415	2415
7.81	7.01	HPPT signal generated.	---	---
8.31	7.51	Reactor trip breakers open.	---	---
8.91	8.11	Scram CEAs begin falling.	---	---
9.36	7.10	MSSV bank 3 open (psia).	1370	1370
9.19	9.62	PSVs open (psia).	2549	2549
9.66	9.95	Maximum RCS pressure (psia).	2659	2631
10.9	10.9	PSVs close (psia).	2422	2422
12.6	12.2	Maximum pressurizer water volume (ft^3).	776	724
16.5	12.9	SG water level reaches AFAS setpoint (%WR).	20	20
16.5	12.9	AFAS generated.	---	---
9.69	14.5	Maximum SG pressure.	1376	1389
30.0	30.2	MSSV bank 3 close (psia).	1301	1301
43.4	44.7	MSSV bank 2 close (psia).	1277	1277
>60	57.3	AFW flow initiated (gpm).	186	186
>60	>60	MSSV bank 1 close (psia).	1238	1238
1800	1800	Operator initiates the cooldown (min).	30	30

Figure 6.3-84
LOCV Secondary Peak Pressure Case-RCS Pressure vs. Time

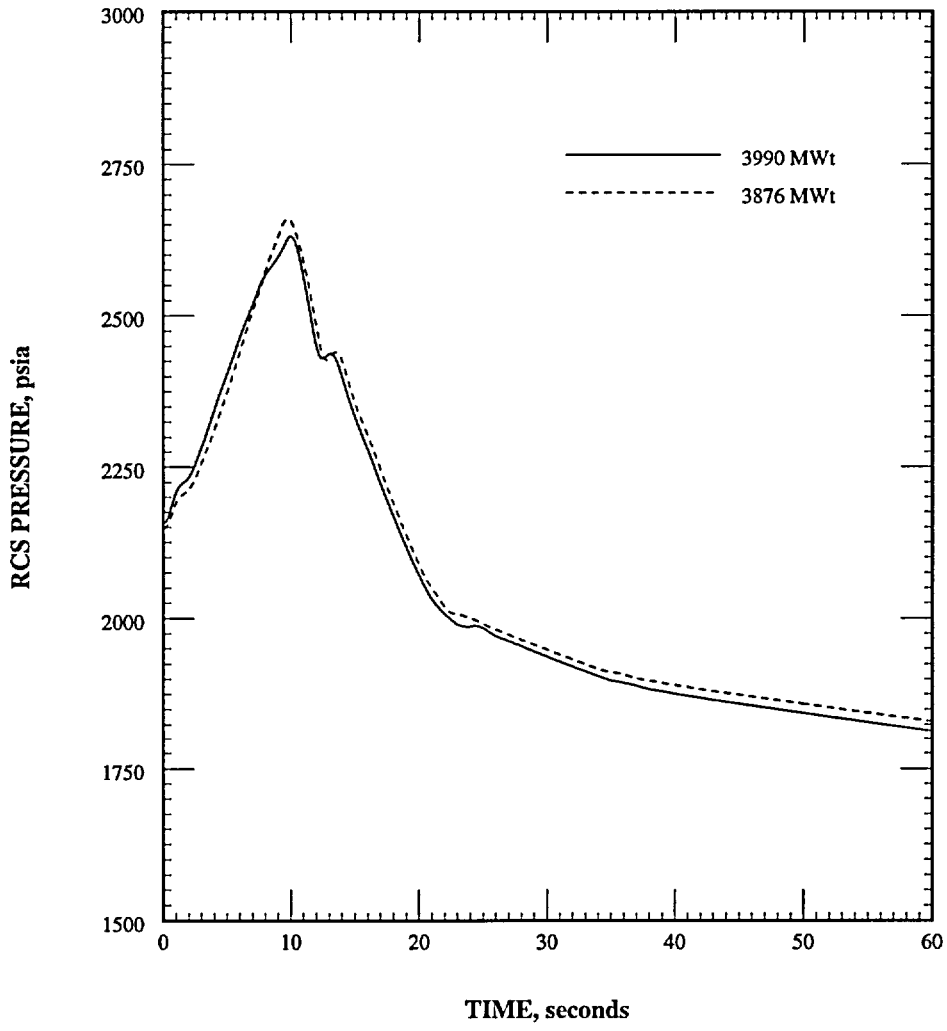


Figure 6.3-85
LOCV Secondary Peak Pressure Case-Pressurizer Pressure vs. Time

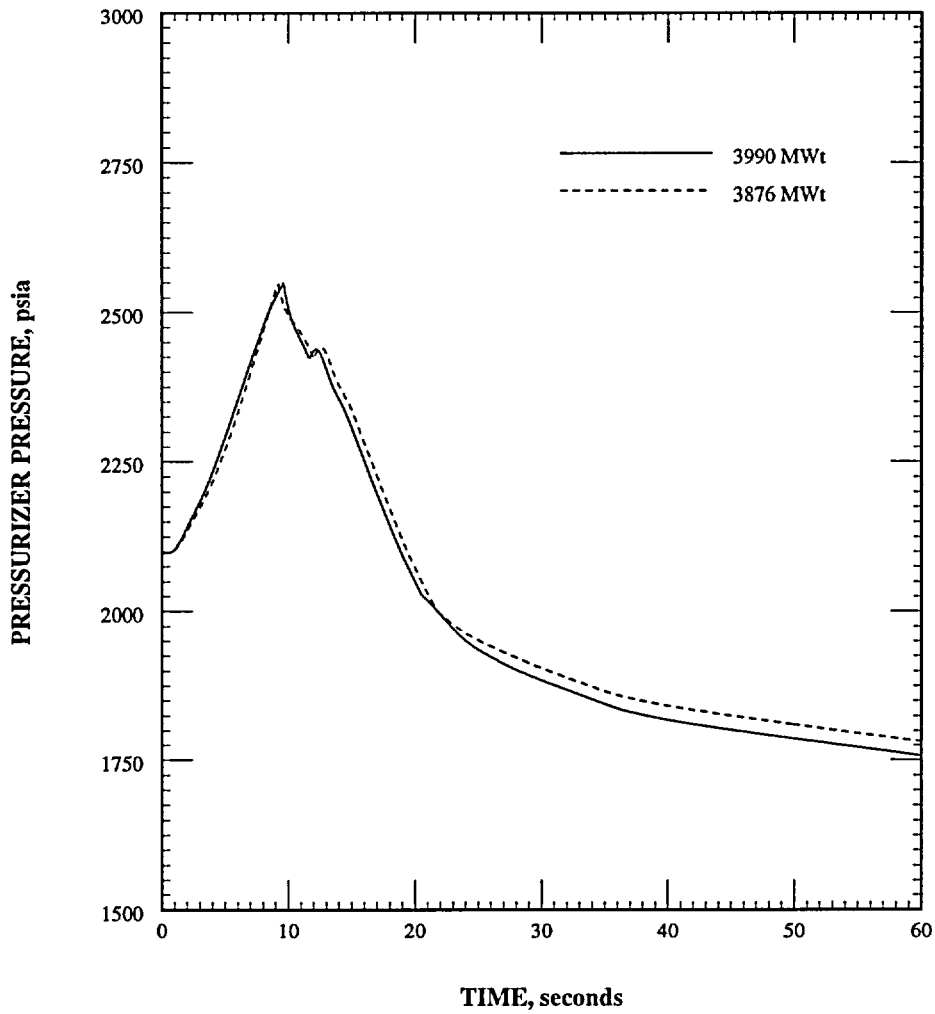


Figure 6.3-87
LOCV Secondary Peak Pressure Case-SG Pressure vs. Time

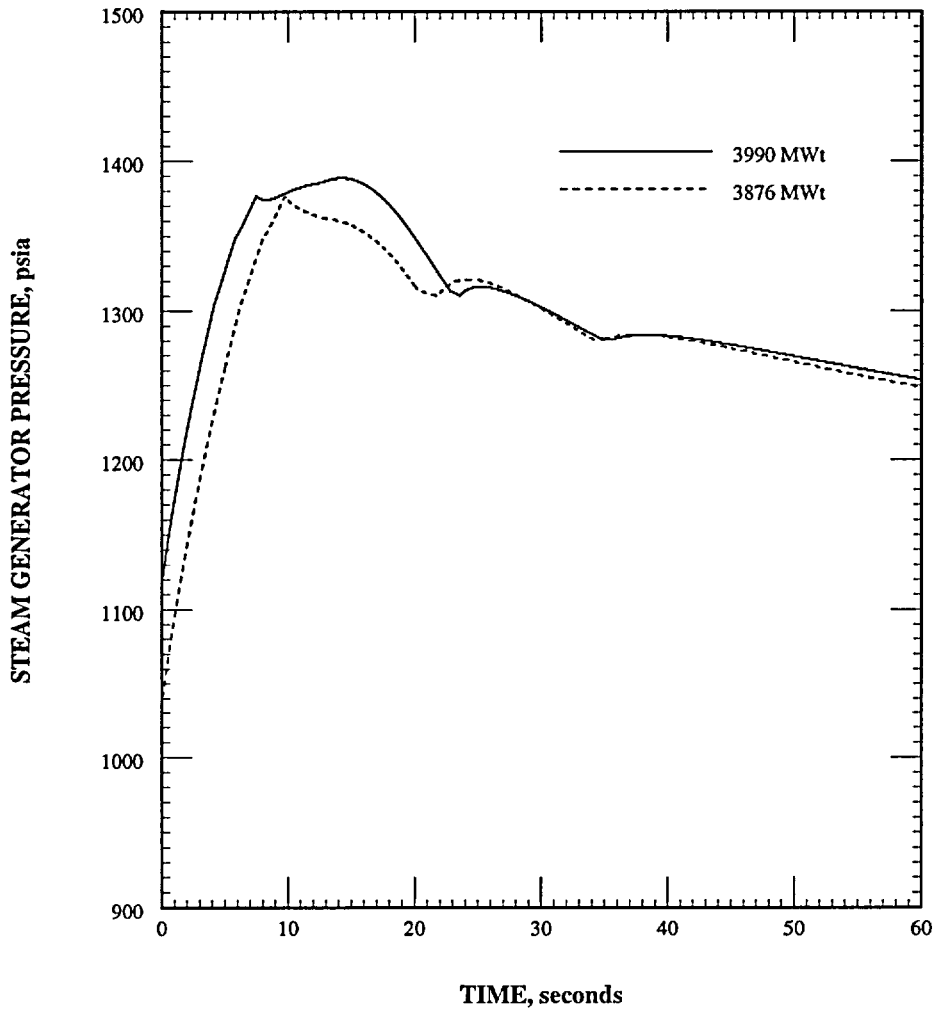


Table 6.3-29
Sequence of Events for FWLB with LOP Primary Peak Pressure Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	FWLB and complete LOFW to both SGs, break size (ft^2).	0.21	0.14
24.41	27.90	Pressurizer pressure reaches trip setpoint (psia).	2450	2450
24.41	27.90	HPPT signal generated.	---	---
24.59	28.11	Dryout of affected SG ($lb_m$of liquid inventory), AFAS generated in affected SG.	< 5000	< 5000
24.91	28.40	Reactor trip breakers open	---	---
24.91	28.40	Turbine trip occurs	---	---
24.91	28.40	LOP occurs	---	---
25.51	29.00	Scram CEAs begin falling.	---	---
26.11	29.55	PSVs open (psia).	2550	2550
30.88	30.11	MSSVs bank 1 open on unaffected SG (psia)	1303	1303
31.30	30.32	MSSVs bank 1 open on affected SG (psia)	1303	1303
27.92	31.43	Maximum RCS pressure (psia).	2749	2778
---	33.78	MSSVs bank 2 open on unaffected SG (psia).	---	1344
33.95	33.88	Peak secondary pressure occurs.	1317	1354
32.11	35.78	PSVs close (psia).	2422	2422
35.29	38.92	Maximum liquid volume of pressurizer (ft^3).	1000	1012
---	44.93	MSSVs bank 2 close on unaffected SG (psia).	---	1277
46.01	52.79	MSSVs bank 1 close on affected SG (psia).	1238	1238
47.09	55.52	MSSVs bank 1 close on unaffected SG (psia).	1238	1238
< 1800	< 1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the long-term cooling FWLB event.	---	---
1800	1800	Operator initiates the cooldown (min).	30	30

Figure 6.3-97
FWLB with LOP Primary Peak Pressure Case-RCS Pressure vs. Time

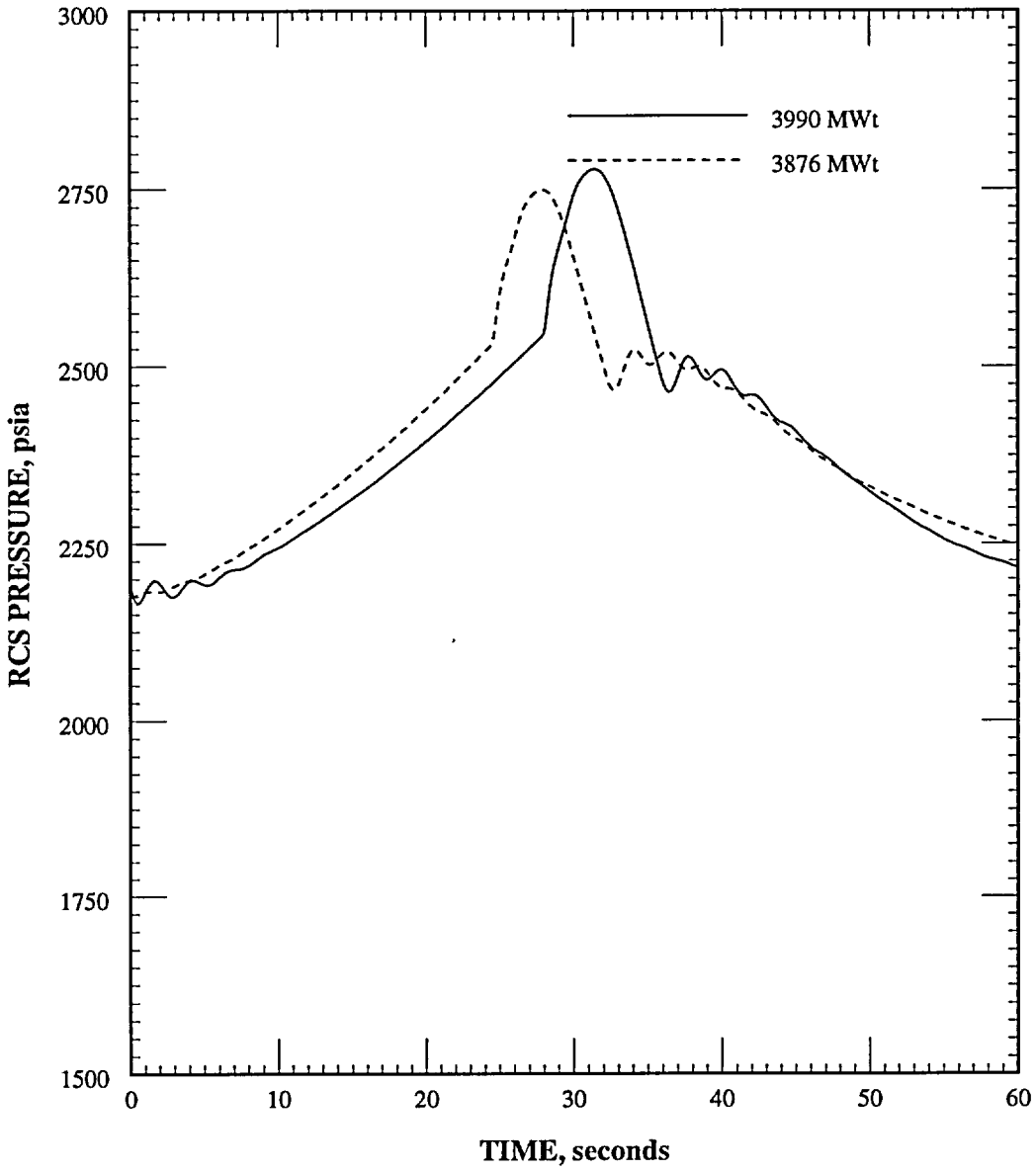


Figure 6.3-98
FWLB with LOP Primary Peak Pressure Case-Pressurizer Pressure vs. Time

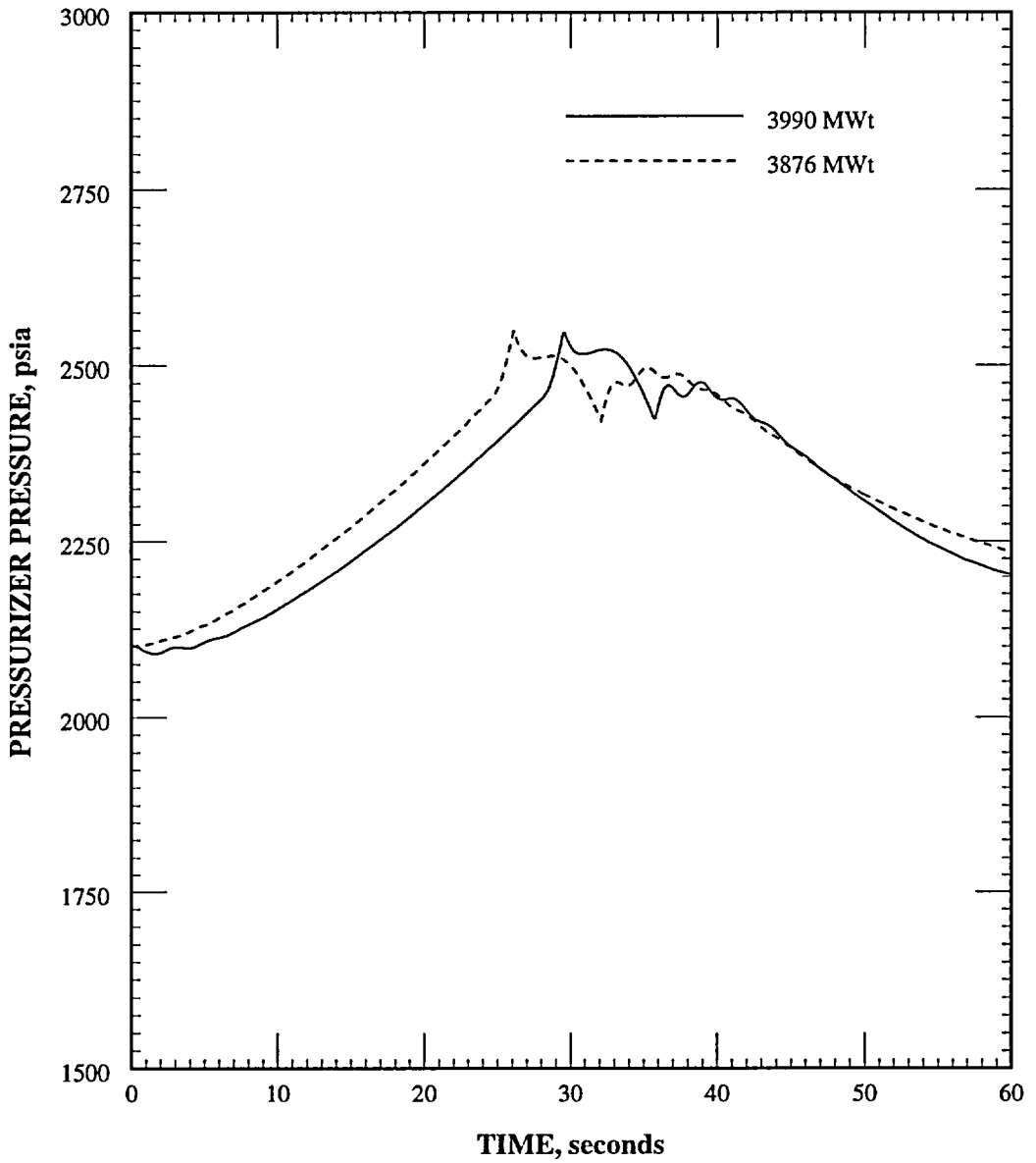


Table 6.3-32
Sequence of Events for SFWLB Primary Peak Pressure Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	FWLB and complete LOFW to both SGs, break size (ft^2).	0.20	0.20
18.53	18.14	Pressurizer pressure reaches trip setpoint (psia).	2450	2450
18.53	18.14	HPPT signal generated.	---	---
19.03	18.64	Reactor Trip breakers open.	---	---
19.03	18.64	Turbine Trip occurs	---	---
19.63	19.24	Scram CEAs begin falling.	---	---
21.24	20.81	PSVs open (psia).	2550	2550
22.54	21.04	MSSVs bank 1 open on unaffected SG (psia).	1303	1303
22.56	21.04	MSSVs bank 1 open on affected SG (psia).	1303	1303
21.87	21.48	Maximum RCS pressure (psia).	2684	2706
24.60	22.68	MSSVs bank 2 open on unaffected SG (psia).	1344	1344
24.84	22.78	MSSVs bank 2 open on affected SG (psia).	1344	1344
24.62	24.51	PSVs close (psia).	2422	2422
25.72	25.64	Peak secondary pressure occurs.	1353	1366
27.36	26.46	Dryout of affected SG (lb_m of liquid inventory), AFAS generated in affected SG.	< 5000	< 5000
30.31	28.00	Maximum liquid volume of pressurizer (ft^3).	754	720
31.05	32.14	MSSVs bank 2 close on affected SG (psia).	1277	1277
31.48	32.63	MSSVs bank 2 close on unaffected SG (psia).	1277	1277
38.33	40.08	MSSVs bank 1 close on affected SG (psia).	1238	1238
40.75	42.74	MSSVs bank 1 close on unaffected SG (psia).	1238	1238
< 1800	< 1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the long-term cooling FWLB event.	---	---
1800	1800	Operator initiates the cooldown (min).	30	30

Figure 6.3-127
SFWLB Primary Peak Pressure Case-RCS Pressure vs. Time

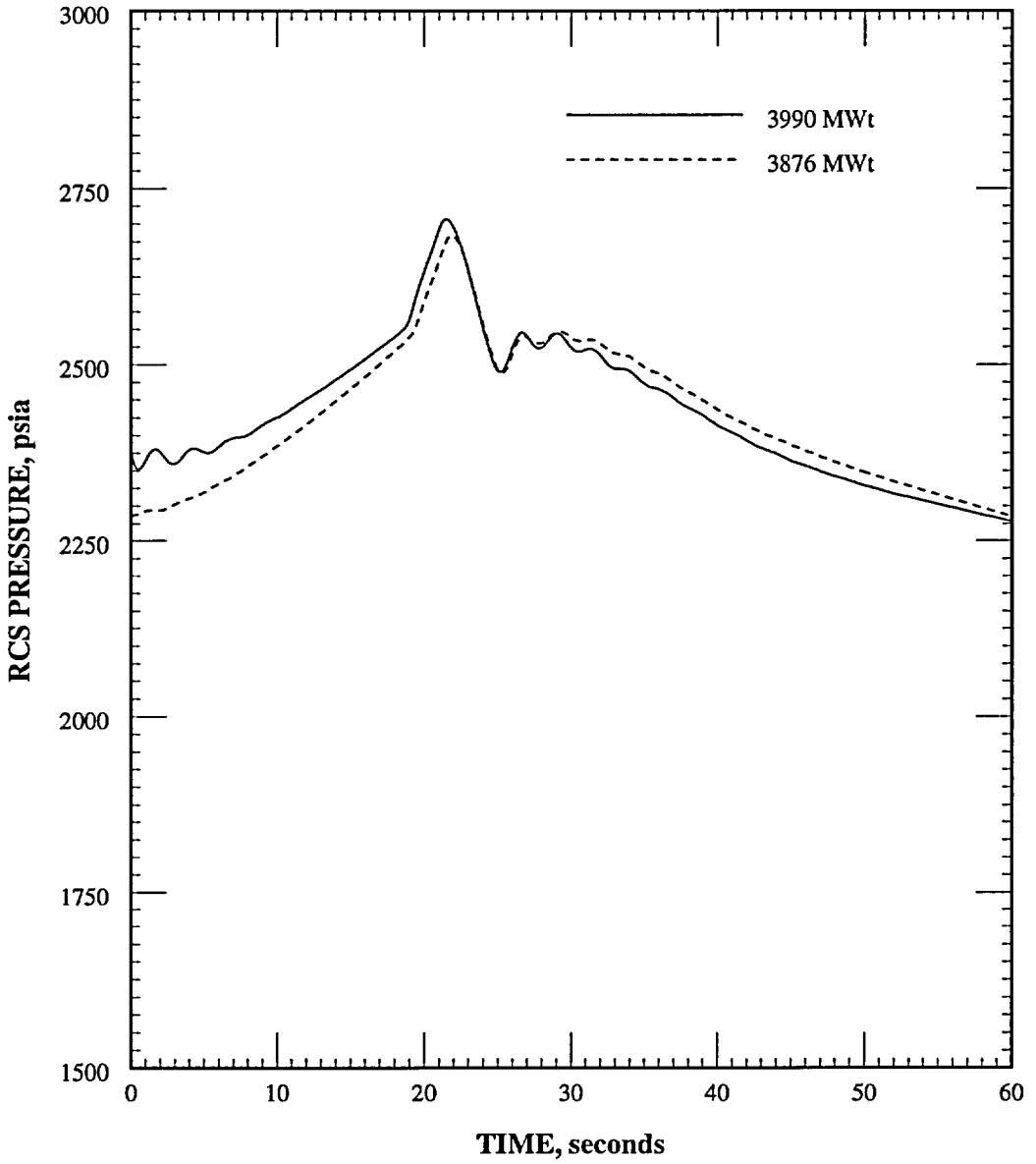


Figure 6.3-128
SFWLB Primary Peak Pressure Case-Pressurizer Pressure vs. Time

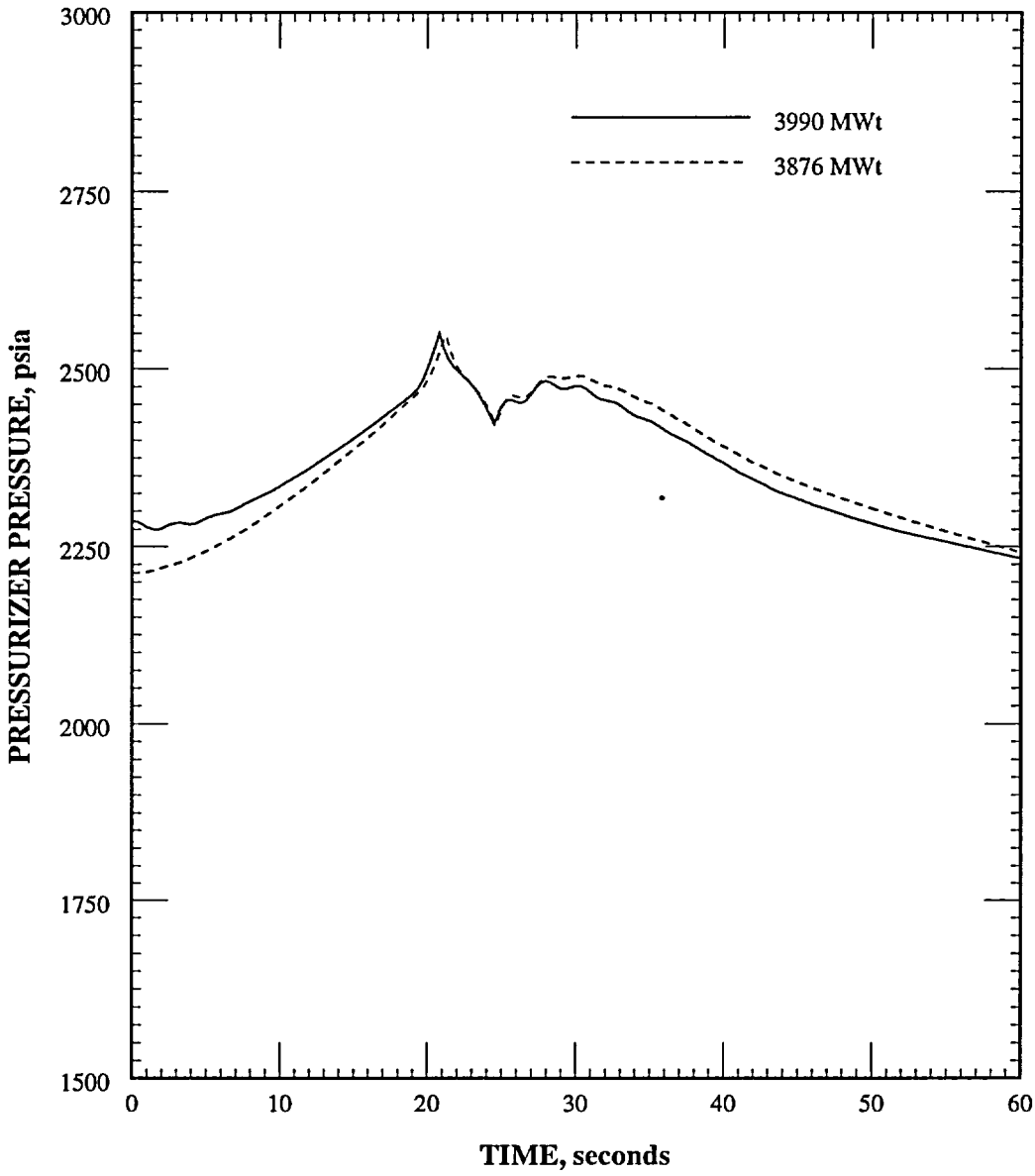


Table 6.3-34
Sequence of Events for the LOF Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOP occurs.	---	---
0.00	0.00	Turbine Trip.	---	---
0.00	0.00	DG starting signal, RCP coast down, main FW is lost.	---	---
0.60	0.60	Low RCP shaft speed trip condition	0.95	0.95
0.90	0.90	Reactor trip occurs	---	---
1.50	1.50	Scram CEAs begin falling.	---	---
2.75	2.75	Minimum DNBR.	> SAFDL	> SAFDL
4.50	4.30	PSVs open (psia) (1st occurrence, valves cycle several times).	2550	2550
4.80	4.70	Maximum RCS pressure (psia).	2644	2654
30.1	28.7	MSSVs open (psia) (1st occurrence, valves cycle several times).	1303	1303
30.0	28.6	Maximum SG pressure (psia).	1304	1304
134.4	174.8	PSVs close (psia) (last occurrence, valves cycle several times).	2476	2487
1066.1	1079.5	MSSVs close (psia) (last occurrence, valves cycle several times).	1279	1279
1800	1800	Operator initiates cooldown (min).	30	30

Figure 6.3-141
Total Loss of RCS Flow-System-Pressure vs. Time

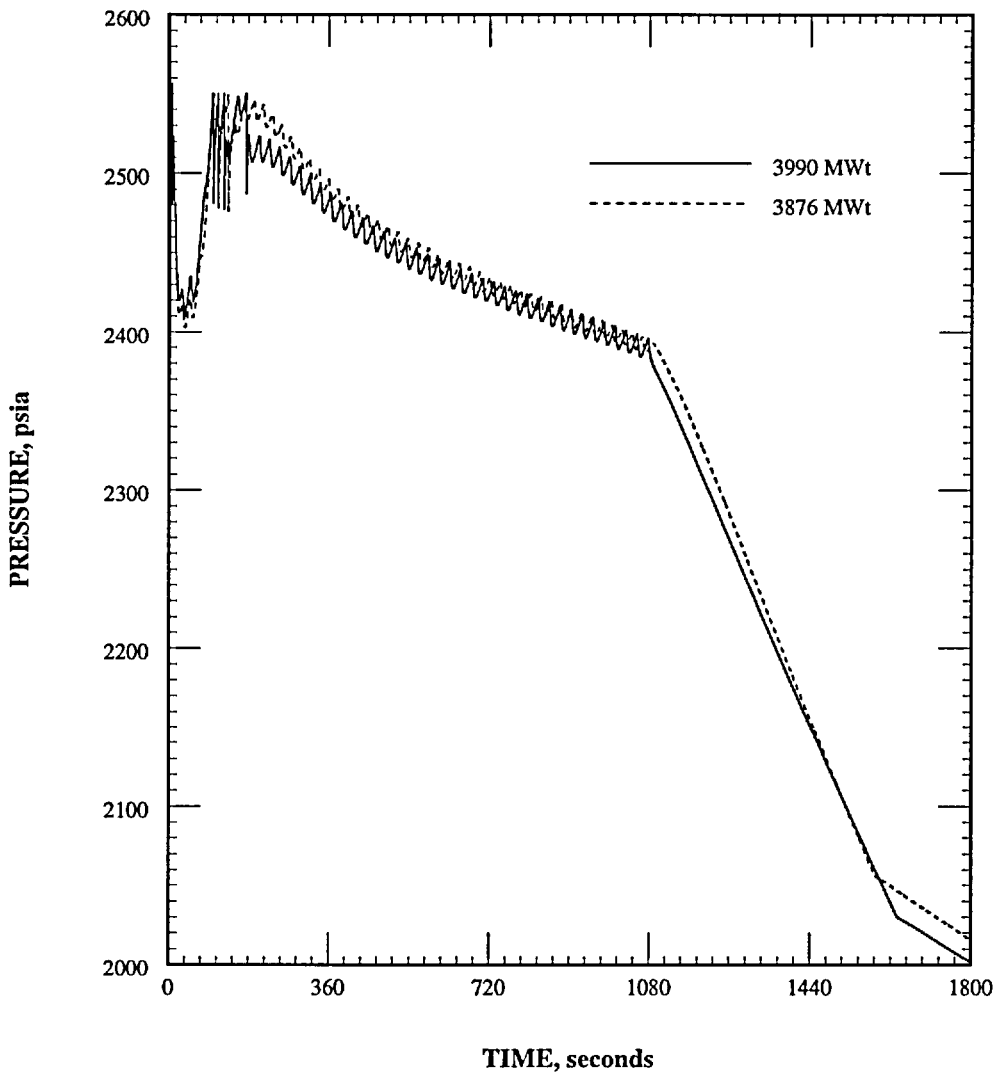


Table 6.3-46
Sequence of Events for the CEA Ejection Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	Mechanical failure of CEDM causes CEA to eject.	---	---
0.05	0.05	CEA fully ejected.	---	---
0.07	0.07	Maximum core power (% of rated).	149.5	149.5
19.7	19.6	Pressurizer pressure reaches trip setpoint (psia).	2450	2450
20.45	20.35	Turbine trip occurs	---	---
20.45	20.35	HPPT reactor trip, turbine trip, main FW (MFW) trip.	---	---
21.05	20.95	Scram CEAs begin falling.	---	---
21.74	21.62	PSVs open (psia).	2550	2550
22.22	22.16	Maximum RCS pressure (psia).	2682	2702
24.56	24.71	PSVs closed (psia).	2422	2422
25.34	24.81	MSSV bank 1 open.	1303	1303
27.95	27.89	Maximum SG pressure (psia).	1348	1349
32.8	30.7	SG level at AFAS setpoint (%WR).	20	20
< 1800	< 1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the CEAW at power event.	---	---
1800	1800	Operator initiates cooldown (min).	30	30

Figure 6.3-188
CEA Ejection Primary Peak Pressure Case-RCS Pressure vs. Time

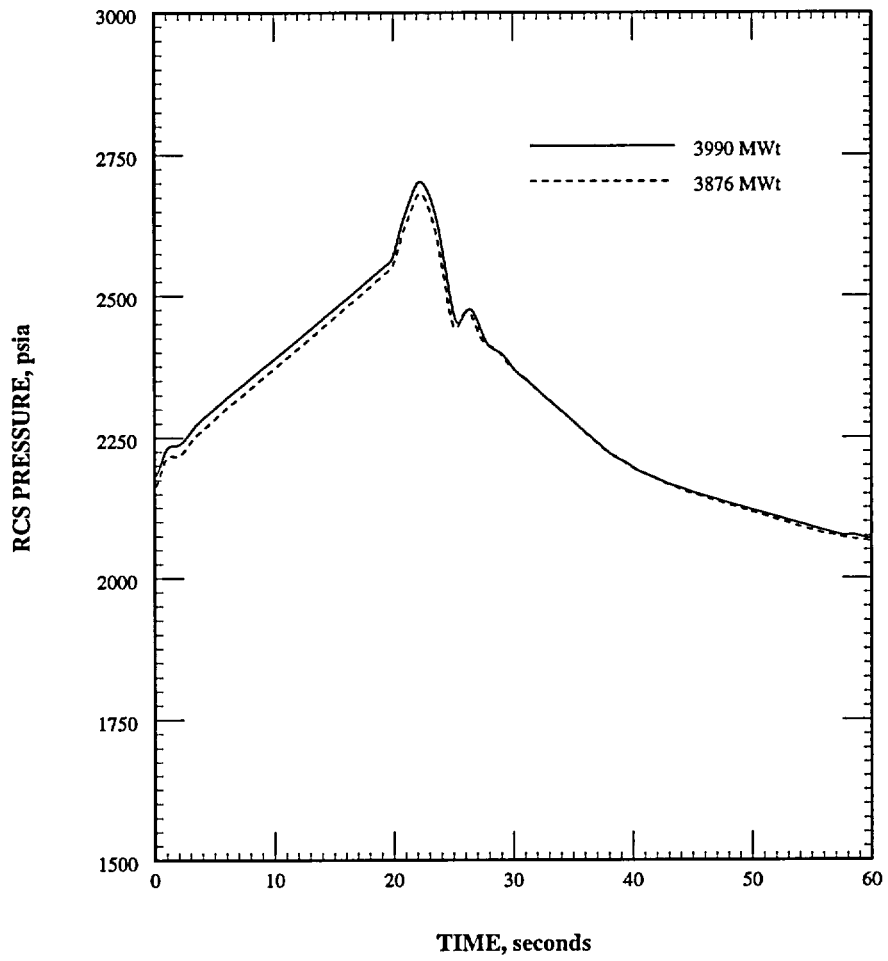


Figure 6.3-189
CEA Ejection Primary Peak Pressure Case-Pressurizer Pressure vs. Time