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STN-50-530 Palo Verde Nuclear Station, Unit 3, Arizona Publi 05000530
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SUBJECT: Application for amends to Licenses NPF-41,NPF-51 & NPF-74, changing TS Sections 3/4.3.1,3/4.4.2 & 3/4.7.1.2.

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WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

161-03587-WFC/JST

November 13, 1990

Docket Nos. STN 50-528/529/530

Document Control Desk
U. S. Nuclear Regulatory Commission
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Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Proposed Technical Specification Amendment to Sections 3/4.3.1, 3/4.4.2, 3/4.7.1, and 3/4.7.1.2
File: 90-056-026

This letter respectfully requests an Amendment to the PVNGS Units 1, 2, and 3 Technical Specifications Sections 3/4.3.1, 3/4.4.2, 3/4.7.1, and 3/4.7.1.2. The proposed changes would increase the allowable setpoint tolerances for the Main Steam Safety Valves (MSSVs) and Pressurizer Safety Valves (PSVs), reduce the minimum required auxiliary feedwater (AFW) flow, and reduce the high pressurizer pressure trip (HPPT) response time.

Arizona Public Service has performed a safety evaluation of the proposed amendment using the ABB-Combustion Engineering CESEC III code demonstrating compliance with the acceptance criteria of the Standard Review Plan. These analyses have been reviewed and concurred with by ABB-Combustion Engineering. The Unit 3, Cycle 3, reload analysis report and all future reload analysis reports will utilize the proposed amendment values as input. To provide adequate time for preparation of future reload analysis reports, approval of this amendment is requested by March 1, 1991.

Enclosed with this request are the following attachments:

Attachment 1 - Safety Evaluation of Proposed Amendment
Attachment 2 - Basis for No Significant Hazards Consideration
Attachment 3 - Environmental Impact Consideration Determination
Attachment 4 - Marked-up Technical Specification change pages

Pursuant to 10 CFR 50.91(b)(1) a copy of this request has been forwarded to the Arizona Radiation Regulatory Agency.

A001
1/1

If there are any questions concerning this request, please contact Michael E. Powell of my staff at (602) 340-4985.

Sincerely,

WFCmcw

WFC/JST/jle

Attachments

cc: J. B. Martin (all w/attachments)
D. H. Coe
A. C. Gehr
A. H. Gutterman
C. E. Tedford

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, W. F. Conway, represent that I am Executive Vice President - Nuclear, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true and correct.

W F Conway
W. F. Conway

Sworn To Before Me This 13 Day Of November, 1990.

Linda Spell
Notary Public

My Commission Expires

June 5, 1992

ATTACHMENT 1

SAFETY EVALUATION

Attachment 1

Safety Evaluation

Table of Contents

Attachment 1

Safety Evaluation

1.0 INTRODUCTION

Three Technical Specification changes are being proposed as a result of difficulties encountered in meeting surveillance requirements for Auxiliary Feedwater (AFW) flow, and Pressurizer Safety Valve (PSV) and Main Steam Safety Valve (MSSV) lift setpoint tolerances. A fourth Technical Specification change, reduction of the High Pressurizer Pressure Trip (HPPT) response time, was necessary to ensure that the peak pressures during postulated accident scenarios which include the proposed changes do not violate the safety limits. A safety evaluation of these proposed changes is included in this document.

During the safety valve surveillance testing, the PSV/MSSV setpoint tolerance limit (± 1 %) was found to be too restrictive. These limits have been exceeded several times, necessitating the issuance of several Licensee Event Reports (LERs) (Reference 1). The inability of surveillance testing to consistently meet the (±1 %) setpoint tolerance limit is an industry wide problem, as indicated by a request for a similar technical specification change by Pacific Gas and Electric Company (Reference 2) and a previously approved increase in safety valve tolerance for Portland General Electric. The inability to meet this specification also results in increased man-rem due to additional testing and maintenance of the safety valves. Since testing of safety valves normally occurs during refueling outages multiple test failures and subsequent valve rework has the potential of impacting restart schedules resulting in significant economic impact.

This safety evaluation demonstrates that using the following parameters does not significantly impact Updated Final Safety Analysis Report (UFSAR) Chapter 6, Chapter 15, and natural circulation cooldown events:

Attachment 1

Safety Evaluation

a) Reduction of the High Pressurizer Pressure Trip (HPPT) response time from the currently docketed 1.15 seconds to 0.5 seconds,

b) Reduction of the minimum AFW flow rate from 750 gpm to 650 gpm, and

c) Changing the PSV tolerance[1] from the current (±1 %) to (+3 %, -1 %) and MSSV tolerance from (±1 %) to (± 3 %)

Surveillance test acceptance criteria based on these changes will ensure that achievable and measurable criteria are available that fully preserve the safety analysis assumptions.

1.1 Description and Purpose of Proposed Changes

The proposed Technical Specification changes affect Technical Specifications 3/4.3.1, 3/4.4.2, 3/4.7.1 and 3/4.7.1.2 for High Pressurizer Pressure Trip (HPPT) response time, auxiliary feedwater flow, and the PSV and MSSV tolerances.

1.1.1 High Pressurizer Pressure Trip Response Time

In Technical Specification 3/4.3.1, Table 3.3-2, the High Pressurizer Pressure Trip response time for trip generation is reduced to 0.5 seconds from 1.15 seconds. APS has reviewed the surveillance history for HPPT response time and found it to be consistently been between 200 ms.(0.2 sec) and 275 ms.(0.275 sec). A reduction to 0.5 seconds is high enough above the historical data to allow for any changes in response time due to equipment aging.

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[1] While it is desirable to change the PSV tolerance also from ±1% to ±3%, thus retaining symmetry, a -3% tolerance is not possible at this time for the reasons discussed in Section 1.1.3.

The purpose of this change is to provide margin in the plant pressure response to allow for increased MSSV and PSV safety valve tolerances.

1.1.2 Auxiliary Feedwater Requirements

Technical Specification 3/4.7.1.2 affects "Auxiliary Feedwater System." The proposed change will accomplish three objectives.

First, it will revise the auxiliary feedwater pump surveillance requirement acceptance criteria of 4.7.1.2.c from 750 gpm to 650 gpm, to allow additional margin between actual pump performance and safety analysis assumptions.

Second, it will clarify the pressure criteria of Technical Specification 4.7.1.2.c so that 1270 psia is clearly understood to correspond to a steam generator test pressure.

Third, it will update the bases section of Technical Specification 3/4.7.1.2 to reflect the changes made above. It should be noted that the surveillance requirement acceptance criteria of Technical Specification 4.7.1.2.c as revised will continue to be safety analysis values. Surveillance test procedures and test acceptance criteria will be adjusted as necessary to ensure these safety analysis assumptions are preserved, while conservatively accounting for surveillance test parameter measurement uncertainties.

1.1.3 PSV and MSSV Tolerance Change

The proposed setpoint tolerance change is different for PSVs vs. MSSVs, as follows:

Attachment 1

Safety Evaluation

a) Technical Specification 3/4.4.2.1 and 3/4.4.2.2 for PSV lift settings is changed from 2500 psia ±1 % to 2500 psia -1/+3 %. The highest and lowest pressure at which the PSVs are presently assumed to open are 2525 psia and 2475 psia respectively using the ±1 % tolerance. Chapter 15 safety analyses use a High Pressurizer Pressure Trip (HPPT) setpoint of 2450 psia for normal environment and 2475 psia for steam environment. Note that the actual plant set point is 2388 psia. The difference accounts for various instrument uncertainties. Thus, retaining the PSV setpoint at 2500 psia with - 1 % tolerance is necessary to ensure that the reactor is tripped before the PSVs open.

b) Technical Specification 3/4.7.1.1, Table 3.7-1: the lift settings are being changed from a ±1 % tolerance to ±3 % tolerance. The constraint discussed for PSVs does not exist for MSSVs in lowering the setpoint tolerance to - 3 % from the current - 1 %.

The purpose of these changes is to more accurately model safety valve performance in safety analysis and prevent future surveillance test failures which could adversely impact refueling schedules.

2.0 IMPACT ON SAFETY ANALYSIS

All of the UFSAR (Reference 4) Chapter 15 analyses, several design basis accidents from FSAR Chapter 6, as well as the natural circulation cooldown were evaluated to determine the impact of reducing the HPPT response time from 1.15 seconds to 0.5 seconds, reducing the Auxiliary Feedwater (AFW) delivered flow rate to 650 gpm, and changing the PSV and MSSV setpoint tolerances. This evaluation is summarized in Section 2.1. In most cases, a detailed evaluation was not required. In these instances, Section 2.1 provides a justification for the impact of the proposed changes to HPPT response time, reduction in AFW flow rate and PSV and MSSV setpoint tolerances on the event. In cases where a detailed evaluation was required (i.e., instances where the reduction in HPPT response time, or AFW flow rate, or PSV/MSSV setpoint tolerance changes had an adverse impact on event results), the results of the evaluations are presented in further detail. Five events required detailed evaluation:

a) Loss of Condenser Vacuum (LOCV)
b) Main Feedwater Line Break (MFLB)
c) Steam Generator Tube Rupture (SGTR)
d) Reactor Coolant Pump (RCP) Shaft Seizure
e) Loss of Coolant Accident (LOCA)

The evaluations of these events are summarized in Sections 2.2, 2.3, 2.4, 2.5, and 2.6 respectively. Additionally, an evaluation of the natural (long term) circulation cooldown is provided in Section 3.

Attachment 1

Safety Evaluation

2.1 Summary of Evaluations

Table 2.1-1 summarizes the evaluation of the impact of reducing HPPT response time, decreasing the AFW delivered flow and changing the setpoint tolerance limits for the PSVs and MSSVs on the PVNGS Safety Analysis.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 1 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Decrease in Feed Water Temperature + Loss Of Power (LOP)	15.1.1	Bounded by LOCV.	Inadvertent Opening of Steam Generator Atmospheric Dump Valve (IOSGADV) event is more limiting than this event. The additional single failure of LOP is similar to (IOSGADV + LOP) event. IOSGADV event itself is bounded by LOCV event. AFW is not actuated.
Increase in Feedwater Flow + LOP	15.1.2	Bounded by LOCV.	IOSGADV event is more limiting than this event. The additional single failure of LOP is similar to (IOSGADV + LOP) event. IOSGADV event itself is bounded by the LOCV event. AFW is not actuated.
Increase in Main Steam Flow + LOP	15.1.3	Bounded by LOCV.	This is identical to IOSGADV event. Additional single failure is similar to (IOSGADV+LOP) event. IOSGADV event itself is bounded by the LOCV event. AFW is not actuated.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 2 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
IOSGADV Case 1	15.1.4	Bounded by LOCV.	Minimum Departure from Nucleate Boiling (MDNBR) equal to Specified Acceptable Fuel Design Limits (SAFDL) occurs at 30 secs. PSVs do not open. MSSVs open after 1850 secs when manual trip is initiated. A secondary peak of about 1300 psia is reached at about 1860 secs. Turbine is tripped on Reactor trip (manually operated) at 30 minutes. MSSVs and Atmospheric Dump Valves (ADVs) remove decay heat. DNBR starts to increase after manual trip. No fuel failure occurs during the transient. Peak secondary pressure is bounded by the LOCV event. AFW is actuated but is important only for post accident cooldown as discussed in Section 3.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 3 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
IOSGADV + LOP Case 2	15.1.4	No impact from PSV tolerance change. MSSVs open after reactor trip and MDNBR. Bounded by LOCV.	Since the PSV setpoint is not challenged during the transient, the PSV tolerance change has no impact. Low DNBR trip occurs at 45.6 secs. MDNBR occurs at 46.1 secs and then starts increasing (refer to Fig 15.1.4.2.15 of Chapter 15 of the UFSAR). MSSVs open at 52 secs. A delayed opening time due to increased MSSV setpoint tolerance will not have any impact on the percentage of failed fuel. Use of -3 % tolerance would result in an earlier opening of MSSVs by about 2 secs i.e. after 50 secs. Therefore MDNBR is not affected by this change. AFW used for post accident cooldown, as discussed in Section 3.
Steam Line Break w/LOP & w/o LOP	15.1.5	Results improve.	These are all cooldown accidents and they do not challenge either the PSV or MSSV pressure limits. Lower AFW reduces cooldown, and associated positive reactivity addition.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 4 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Loss of external load (LOL)	15.2.1	Bounded by LOCV.	Results of LOL are less limiting than the LOCV event. AFW used for post accident cool-down as discussed in Section 3.
Turbine Trip	15.2.2	Bounded by LOCV.	Results of Turbine Trip are less limiting than the LOCV event. AFW is not actuated.
Turbine Trip w/LOP	15.2.2	Bounded by LOCV.	Results of Turbine Trip w/LOP are the same as the LOCV event. AFW used for post accident cool-down, as discussed in Section 3.
LOCV	15.2.3	Case was ana-lyzed. Peak pressures are within Standard Review Plan (SRP) limits.	Peak RCS pressure occurs prior to AFW actuation. See Section 2.2 for details. AFW used for post accident cooldown, as dis-cussed in Section 3.
MSIV Closure	15.2.4	Bounded by LOCV.	The LOCV event is more limiting. AFW is actuated well after peak RCS pressure is reached.
Steam Pressure Regulator Fail-ure	15.2.5	None.	Event not applicable to PWRs.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 5 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Loss of AC Power (LOP)	15.2.6	Bounded by LOCV.	Results of LOP are identical to LOF (15.3.1) and bounded by the LOCV event. AFW is actuated well after MDNBR and peak RCS pressure are reached.
Loss of Normal Feed Flow	15.2.7	Bounded by LOCV.	Maximum RCS pressure is less than the LOCV event. AFW is actuated well after Reactor trip and peak RCS pressure.
Feedwater Line Break	15.2.8	Case was analyzed. Peak pressures are within SRP limits.	See Section 2.3 for details.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 6 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Loss of Reactor Coolant Flow + LOP	15.3.1	Bounded by LOCV.	Per Sections 15.3.1 of the UFSAR, Reactor coolant pump shaft speed trip occurs at 0.6 secs and MDNBR greater than or equal to SAFDL occurs at 2.2 secs. The Max RCS and Secondary pressures reached are 2576 and 1338 psia at 5.3 and 11.7 secs, respectively. These are much lower than the 2742 and 1353 psia reached at 8.6 and 14.0 secs for the LOCV event. This event is bounded by the LOCV event. AFW is actuated well after peak RCS pressure and MDNBR are reached.
Flow Controller Malfunction	15.3.2	None.	Event not applicable to PWRs
Single Reactor Coolant Pump Seizure + LOP	15.3.3	Bounded by LOCV	Low DNBR trip occurs at 0.76 secs. Maximum RCS pressure reached for this event is 2387 psia at 4.2 secs and does not challenge the PSV setpoint. MDNBR occurs at about 1.3 secs. So the turn around in MDNBR is not affected by safety valve setpoints. See Section 2.5 for AFW impact.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 7 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Single Reactor Coolant Pump Sheared Shaft	15.3.4	Bounded by LOCV.	This event is bounded by RCP Seizure + LOP.
Low Power Control Element Assembly (CEA) withdrawal	15.4.1	None.	PSV and MSSV setpoints are not challenged. AFW is not actuated.
Full Power CEA withdrawal	15.4.2	Bounded by LOCV.	Low DNBR trip occurs at 9.5 secs. The Max RCS pressure reached is 2363 psia at 12.3 secs, this is less than the LOCV event peak pressure, and does not challenge the PSV setpoint. MDNBR occurs at 11.0 secs and before the max RCS pressure. Secondary peak of ~1355 psia occurs at 18 secs after reactor trip and therefore will not impact MDNBR. AFW is used for post accident cooldown, as discussed in Section 3.
CEA Assembly Drop	15.4.3	None.	PSV and MSSV setpoints are not challenged. AFW is not actuated.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 8 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Startup of Inactive Reactor Coolant Pump (RCP)	15.4.4	None.	AFW is not actuated.
Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	15.4.5	None.	Not applicable to PWR.
Inadvertent Deboration	15.4.6	None.	PSV and MSSV setpoints are not challenged. AFW is not actuated.
Inadvertent Fuel Misload	15.4.7	None.	UFSAR analysis is still valid since neither PSVs nor MSSVs are challenged. AFW is not actuated.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 9 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
CEA Ejection	15.4.8	Less than allowed by Service Limit C defined in ASME code for faulted conditions.	Trip occurs due to Variable Over Power Trip (VOPT), highest RCS pressure is 2757 psia. PSVs open at 2525 psia. Using an additional 50 psia increase in PSV setpoint (corresponds to + 3 % tolerance) would result in a peak of less than the 3000 psia allowed by Service Limit C as defined in ASME code. Max secondary pressure reached is 1348 psia at 4.8 secs, this is lower than 1353 psia reached at 14.0 secs for the LOCV event. AFW is used for post accident cooldown, as discussed in Section 3.
Inadvertent Emergency Core Cooling System (ECCS) operation	15.5.1	None.	PSV and MSSV setpoints are not challenged. AFW is not actuated.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 10 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Pressurizer Level Control System (PLCS) malfunction with LOP	15.5.2	Bounded by LOCV.	HPPT setpoint of 2450 psia occurs at 1250.7 secs. LOP also occurs at 1250.7 secs. Pressurizer and secondary peak pressures reached for this event are 2561 psia and 1298 psia at 1253 and 1270 secs, respectively. For the LOCV event, the primary pressure has a much steeper rise to its peak value and the initial pressure is iterated to result in the maximum peak pressure. Thus the LOCV event results in worse peak RCS and SG pressures. These are 2742 and 1353 psia at 8.6 secs & 14.0 secs, respectively for the UFSAR LOCV event. AFW is used for post accident cooldown, as discussed in Section 3.
Inadvertent Opening of a Pressurizer / Relief Safety Valve	6.3.3 (15.6.1)	None.	PSV and MSSV setpoints are not challenged. AFW is used for post accident cooldown, as discussed in Section 3.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 11 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Double Ended Break of Letdown Line Outside Containment	15.6.2	None.	This is not a pressurization event and thus the PSVs and MSSVs are not required to operate. AFW is not actuated.
Steam Generator Tube Rupture	15.6.3.1	None.	PSV setpoint is not challenged since primary pressure is decreasing for the event. Max Secondary Pressure of 1283 psi is reached at 1210 secs and is much less than 1353 psia reached for the LOCV event. AFW is not actuated.
Steam Generator Tube Rupture + LOP	15.6.3.2	Impact bounded by SGTR + LOP + Stuck ADV.	PSV setpoint is not challenged since primary pressure is decreasing for the event. The Max Secondary peak pressure of 1310 psia at 1205 secs is well below the 1353 psia reached for the LOCV event. AFW is not actuated.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 12 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Steam Generator Tube Rupture + Stuck ADV	15.6.3.2	Analyzed in Section 2.4	PSV setpoint is not challenged since primary pressure is decreasing for the event. The maximum secondary pressure of 1310 psia is reached at 56 secs and is less than 1353 psia reached for the LOCV event. Refer to Section 2.4 for details on radiological releases. AFW is not actuated.
Radiological Consequences of MSLB Outside Containment (BWR)	15.6.4	None.	Event not applicable to PWR.
Large Break Loss Of Coolant Accident (LOCA)	15.6.5	None.	See Section 2.6 for details.
Small Break LOCA	15.6.5	Bounded by Feedwater Line Break + LOP results.	See Section 2.6 for details.
Containment Design Basis Accident	6.2.1	None.	Increase in safety valve setpoint tolerances, lower AFW flow, and lower HPPT response time do not affect the mass/energy release to containment.

Table 2.1-1
Summary of the Effects of Decreasing AFW Delivered Flow and HPPT Response Time, and Changing the PSV/MSSV Setpoint Tolerances on PVNGS UFSAR Transients (Sheet 13 of 13)

EVENT	UFSAR SECTION	IMPACT OF CHANGES	JUSTIFICATION
Natural Circulation Cooldown		Results show that 650 gpm AFW flow is sufficient to remove decay heat and cool down the plant to cold shut down.	See Section 3.

2.2 Loss of Condenser Vacuum

2.2.1 Purpose

The loss of condenser vacuum event was reanalyzed to verify that the reduced HPPT response time, PSV and MSSV tolerance changes, and AFW flow reduction do not result in a violation of design limits. A LOCV may occur due to the failure of the circulation water system to supply cooling water, failure of the condenser evacuation system to remove the non-condensible gases, or excessive in-leakage of air through the turbine glands. The turbine is conservatively assumed to trip immediately coincident with the cause for the LOCV.

In automatic mode, the Steam Bypass Control System (SBCS) and the Reactor Power Cutback System (RPCS) would function to reduce the steam generator and RCS pressure increases during a turbine trip. These systems would mitigate the consequences of a loss of condenser vacuum. However, in this analysis both the SBCS and RPCS are conservatively assumed to be in a manual mode, and credit is not taken for their operation. This is in line with the requirement that those items which are not safety grade should not be credited in the analysis. An automatic reactor trip is assumed to occur on high pressurizer pressure.

Closure of the turbine stop valves and coastdown of the main feedwater pumps causes the primary and secondary pressures and temperatures to increase rapidly. The pressure increase is limited by the pressurizer and main steam safety valves. The operator may cool the NSSS by using manual operation of the auxiliary feedwater system and the atmospheric dump valves anytime after the reactor trip occurs.

The assumptions made in this LOCV analysis are similar to the ones made in the Updated FSAR. Three additional assumptions, each supported by either tests or analyses, have been made to limit the RCS peak pressure increase. These assumptions are summarized below:

1) The High Pressurizer Pressure Trip (HPPT) response time was changed to 0.5 secs from 1.15 secs. Surveillance tests for all the three units have shown that this trip response time is consistently less than 0.3 seconds. An assumed response time of 0.5 seconds is therefore conservative.

2) The surge line friction form loss factor was reduced to 3.0 from 3.9 to reflect actual PVNGS design. This change was analytically justified in a calculation.

3) In previous analyses, the pressurizer safety valves were assumed to open to 70 % of the nominal area opening at the setpoint. In this analysis, the safety valves are assumed to open to 100 % (modeled in the CESEC code as .99 of the nominal area opening) at the setpoint pressure. This operation of primary safety valves is justified based on the test data presented in ABB-Combustion Engineering Topical Report CEN-227 "Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants" (Reference 3). This report was docketed for Palo Verde in Supplement 8 of the Safety Evaluation Report.

2.2.2 Results

A number of cases were run using CESEC III to make sure the maximum primary pressure peak was obtained. Ordinarily, the reactor trip as a result of LOCV could occur either due to the

HPPT set point or Low Steam Generator Level (LSGL) trip set point being reached. In this analysis, the LSGL is disabled and the trip occurs due to a HPPT for LOCV. The initial steam generator mass is adjusted such that the LSGL trip set point also occurs at the same time as the HPPT set point.

Table 2.2-1 presents the peak pressure and the sequence of events for the most limiting case. The maximum RCS pressure of 2740.9 psia occurs at 9.15 secs while the secondary pressure of 1369.6 psia peaks at 10.6 secs. Both of these peak pressures are below the acceptance criteria of 110% of system design pressure (2750 psia and 1375 psia, respectively).

Table 2.2-1

Sequence of Events for LOCV at Full Power

Time (secs)	Event	Setpoint or Value
0.0	Loss of Condenser Vacuum - Loss of all normal Feedwater & Turbine Load	
7.2	MSSVs Begin to Open, psia	1302
7.75	High Pressurizer Pressure Trip Occurs, psia	2450
8.25	Pressurizer Safety Valves Begin to Open	2575
8.25	Trip Breakers Open 0.5 secs after trip	--
8.59	CEAs Begin to Fall into the Core	--
9.15	Maximum RCS Pressure Occurs, psia	2740.86
10.55	Maximum Secondary Pressure Occurs, psia	1369.6

2.2.3 Conclusion:

The maximum RCS and secondary pressures for a LOCV accident scenario are less than 110 % of the system design pressures as required by the Standard Review Plan for incidents of moderate frequency. Since DNBR increases during the event, there will be no fuel failure and therefore no radiological releases to the atmosphere.

2.3 Feedwater System Pipe Breaks

The feedwater system pipe break events currently documented in the UFSAR were reanalyzed to verify that the RCS and SG overpressurization limits are not exceeded, and that long-term RCS heat removal for the feedwater line break event is not adversely impacted, as a result of the proposed changes to the Technical Specifications. The impact of increased pressurizer safety valve (PSV) and main steam safety valve (MSSV) setpoint tolerances on RCS and SG overpressurization in the large and small feedwater line break limiting cases is addressed in Section 2.3.1. The combined impact of reduced auxiliary feedwater flow and the increased PSV/MSSV setpoint tolerances on long-term RCS heat removal following the feedwater line break limiting case is evaluated in Section 2.3.2.

2.3.1 Overpressurization

2.3.1.1 Purpose

The purpose of this analysis is to verify that the overpressurization limits for large and small feedwater system pipe breaks are not exceeded if the pressurizer safety valve (PSV) and main steam safety valve (MSSV) upper setpoint

tolerances are increased from +1 to +3 %. The decrease in auxiliary feedwater flow from 750 to 650 gpm does not affect the RCS and SG peak pressures, since the auxiliary feedwater flow is not initiated until after the peak pressures occur.

2.3.1.2 Discussion

The limiting cases for large and small feedwater line break events presented in the UFSAR are re-analyzed over the interval bounding peak RCS and SG peak pressures (0 to 50 seconds) to evaluate the impact of increasing the upper setpoint tolerance for the PSVs and MSSVs from +1 % to +3 %. The assumptions made in the UFSAR analysis are maintained, except for changes to the HPPT response time, the PSV setpoint area fraction, and the pressurizer surge line loss coefficients as documented in Section 2. The methods of analysis are consistent with those described in the UFSAR.

2.3.1.3 Large Feedwater Line Break - Limiting Case

Results

The large feedwater line break limiting case is a 0.2 square foot feedwater line break with a loss of offsite power at trip. The dynamic behavior of the important NSSS parameters is essentially the same as that shown in UFSAR Appendix 15E, "Analysis Methods for Feedwater Line Breaks," for the first 50 seconds of the loss of feedwater inventory limiting case as depicted in UFSAR Table 15E-2 and Figures 15E-13 through 15E-31. This UFSAR case is the large feedwater line break limiting case currently represented in the UFSAR, with a peak RCS pressure of 2843 psia, and a SG peak pressure of 1318 psia.

The corresponding peak pressures for this analysis are 2816 and 1358 psia, respectively. The RCS peak pressure is bounded by the current UFSAR analysis because the increase in RCS peak pressure that would be due solely to the increased PSV setpoint tolerance is offset by the other changes noted in section 2.3.1.2. The SG peak pressure of 1358 psia exceeds the value in the current UFSAR analysis, but is significantly less than the Standard Review Plan SG peak pressure limiting criteria of 1500 psia (120 % of design pressure).

Conclusion

This analysis demonstrates that the overpressurization limits for the large feedwater line break limiting case will not be challenged by the proposed changes. The minimum DNBR remains above the Specified Acceptable Fuel Design Limits (SAFDL) because this is a pressurization event. This indicates that no fuel cladding failure would be expected to occur.

2.3.1.4 Small Feedwater Line Break - Limiting Case

Results

The small feedwater line break limiting case is a feedwater line break of 0.2 square foot with a limiting single failure and offsite power available. The limiting single failure for this event is a failure of two RCPs to fast transfer following turbine trip. This single failure was selected because it provides the most adverse RCS peak pressure for this event. The analysis was also performed without the single failure to provide additional evaluation of the SG peak pressure. The dynamic behavior of the important NSSS parameters is essentially the same as that shown in UFSAR Appendix 15E, "Analysis Methods for Feedwater Line Breaks," for the first 50 seconds of the small loss of feedwater inventory events depicted in UFSAR Table 15E-4 and Figures 15E-

32 through 15E-38. This UFSAR case is the small feedwater line break limiting case with a peak RCS pressure of 2712 psia, and a peak SG pressure of 1342 psia.

The RCS peak pressure for this analysis is 2668 psia, and is bounded by the current UFSAR analysis. The corresponding SG peak pressure is 1365 psia. A slightly higher SG peak pressure of 1369 psia was obtained by reanalyzing the event without the single failure noted above. Hence, the bounding RCS and SG pressures for this event are 2668 and 1369 psia, respectively.

The RCS peak pressure for the small feedwater line break limiting case is bounded by the current UFSAR analysis because the increase in RCS peak pressure that would be due solely to the increased PSV setpoint tolerance is offset by the other changes noted in section 2.3.1.2. The SG peak pressure of 1369 psia exceeds the value in the UFSAR analysis, but is less than the Standard Review Plan, Section 15.2.8, SG peak pressure limiting criteria of 1375 psia (110 % of design pressure).

Conclusion

This analysis demonstrates that the overpressurization limits for the small feedwater line break limiting case will not be challenged by the proposed changes.

2.3.2 Long Term RCS Heat Removal

2.3.2.1 Purpose

The purpose of this analysis is to verify that long term RCS heat removal during the limiting Chapter 15 transient, the feedwater line break, will not be adversely impacted by the

proposed technical specification changes.

2.3.2.2 Discussion

The limiting case for the feedwater line break event presented in the UFSAR was re-analyzed to evaluate the impact of the increased PSV/MSSV setpoint tolerances and reduced auxiliary feedwater flow on long term RCS heat removal. The assumptions made in the UFSAR are maintained. Following actuation of auxiliary feedwater on low steam generator water level, auxiliary feedwater is delivered to the steam generators throughout the transient by a single auxiliary feedwater pump. The auxiliary feedwater flow to the ruptured steam generator is lost through the feedwater line break. The auxiliary feedwater flow to the intact steam generator is diverted to the ruptured steam generator at a rate which is dependent on the pressure difference between the steam generators. This continues until the ruptured steam generator is identified and isolated. This flow diversion is specifically accounted for in the evaluation.

The CESEC III code and input was modified to define the auxiliary feedwater flow to the steam generators as a function of PVNGS specific piping resistances and head curves. A reduced auxiliary feedwater flow was then defined, in which the delivered flow from a single auxiliary feedwater pump is reduced from 750 to 650 gpm for a steam generator pressure of 1270 psia. This allows crediting of the greater auxiliary feedwater flow which is delivered to the steam generators at pressures below 1270 psia, while reducing the auxiliary feedwater flow to steam generators at pressures above 1270 psia. Hence, the increased steam generator pressures due to the higher MSSV setpoint tolerance result in a correspondingly lower auxiliary feedwater flow to the steam generators.

The analysis assumes the loss of offsite power occurs coincident with the turbine-generator trip. The physics data generated for PVNGS Unit 1 Cycle 3 applies. The input parameters and initial conditions are the same as those used in the UFSAR analysis, except that:

1) The PSV and MSSV opening pressures are increased by +3 % to reflect the proposed change.
2) The PSV setpoint area fraction is increased from 0.7 to 0.99, as described in Section 2.2.
3) The pressurizer surge line form loss coefficients are reduced from 3.9 to 3.0, as described in Section 2.2.
4) The auxiliary feedwater flows following isolation of the ruptured steam generator are calculated to explicitly model auxiliary feedwater delivered flows for a wider range of steam generator pressures.

Results

The dynamic behavior of the important NSSS parameters following the 0.2 ft^2 feedwater line break with a loss of offsite power is presented in Figures 2.3-1 through 2.3-8. The sequence of events provided in Table 2.3-1 summarizes the important results of this event.

A 0.2 ft^2 rupture in the main feedwater line is assumed to instantaneously terminate main feedwater flow to both steam generators and establish critical flow from the steam generator nearest the break. This causes a decrease in steam generator inventory as shown in Figure 2.2-8.

At 27.6 seconds, reactor trip conditions are reached on high pressurizer pressure[2]. An auxiliary feedwater actuation signal is generated at that time. The reactor trip breakers open 1.15 seconds later (The reduction in HPPT response time from 1.15 to 0.5 seconds is not credited in this event). At 35.71 seconds, the ruptured steam generator dries out. By 73.75 seconds, auxiliary feedwater reaches the intact steam generator as shown in Figure 2.3-7.

Both steam generator pressures decrease gradually until the main steam isolation signal is generated at 224.25 seconds. As shown in Figure 2.3-6, it is at this point that the pressure of the two steam generators begins to deviate significantly. Between this time and the isolation of auxiliary feedwater to the ruptured steam generator, auxiliary feedwater to the intact steam generator is assumed to be 0.0 gpm. At 231.9 seconds, the intact steam generator dries out and remains empty until auxiliary feedwater is reinitiated to it at 275.0 seconds. During this time when both steam generators are dried out, there is a slight increase in the RCS temperature, pressure, and pressurizer water volume as shown in Figures 2.3-2, 2.3-4, and 2.3-5, until auxiliary feedwater is reestablished.

After isolation of the ruptured steam generator, the intact steam generator's pressure gradually increases to the main steam safety valve setpoint. This causes the RCS temperatures, pressurizer level, and pressurizer pressure to gradually increase. By 600 seconds, the main steam and pressurizer safety valves are maintaining primary and secondary pressure. By 900 seconds, the plant is in a stable condition with the pressurizer liquid

[2] No credit is taken for an earlier trip signal on low steam generator level in the ruptured steam generator.

inventory less than 1300 ft^3, compared to 1200 ft^3 in the current UFSAR analysis, and the intact steam generator liquid inventory is at approximately 35,800 lb_m compared to 51,000 lb_m for the current UFSAR analysis.

2.3.2.3 Conclusions

This analysis demonstrates that long-term RCS removal will be maintained during the feedwater line break limiting case, and is not adversely impacted by the proposed changes to the technical specifications.

Table 2.3-1

Sequence of Events for a 0.2 ft^2 Feedwater Line Break With the Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump (Sheet 1 of 2)

Time (sec)	Event	Setpoint or Value
0.0	Rupture in the main feedwater line, ft^2.	0.2
0.0	Complete loss of feedwater to both steam generators.	--
0.0	Initial steam generator break flow, lb_m/sec.	1987
27.60	Pressurizer pressure reaches reactor trip analysis setpoint, psia.	2475
27.60	Steam generator water level reaches low level reactor trip and auxiliary feedwater actuation signal analysis setpoint, ruptured generator heat transfer degradation begins[1].	--
27.60	High pressurizer pressure trip and auxiliary feedwater actuation signals generated.	--
28.75	Reactor trip breakers open.	--
31.73	Maximum RCS pressure (uncorrected for reactor coolant pump and elevation heads), psia.	2673 [2]

[1] Heat transfer degradation begins when the SG liquid volume equals 40,000 lb_m.

[2] Not limiting - see section 2.3.1 for overpressurization limiting cases.

Table 2.3-1

Sequence of Events for a 0.2 ft^2 Feedwater Line Break With the Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump (Sheet 2 of 2)

Time (sec)	Event	Setpoint or Value
35.71	Ruptured steam generator dries out[3].	--
73.75	Auxiliary feedwater initiated to intact steam generator.	--
224.25	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia.	820
224.25	MSIS generated.	--
230.15	MSIVs completely closed.	--
231.9	Intact steam generator dries out.	--
255.0	Difference between steam generator pressures reaches analysis setpoint for lockout of AFW to ruptured steam generator, psid.	325
255.0	Signal to isolate AFW from ruptured steam generator generated.	--
275.0	Isolation of steam generator completed, auxiliary feedwater reestablished to intact steam generator.	--
900	Pressurizer water volume at end of simulation, ft^3.	1272.3
900	Intact steam generator liquid inventory at end of simulation, lb_m.	35,878

[3] Dryout occurs when the SG liquid volume equals 5,000 lbm.

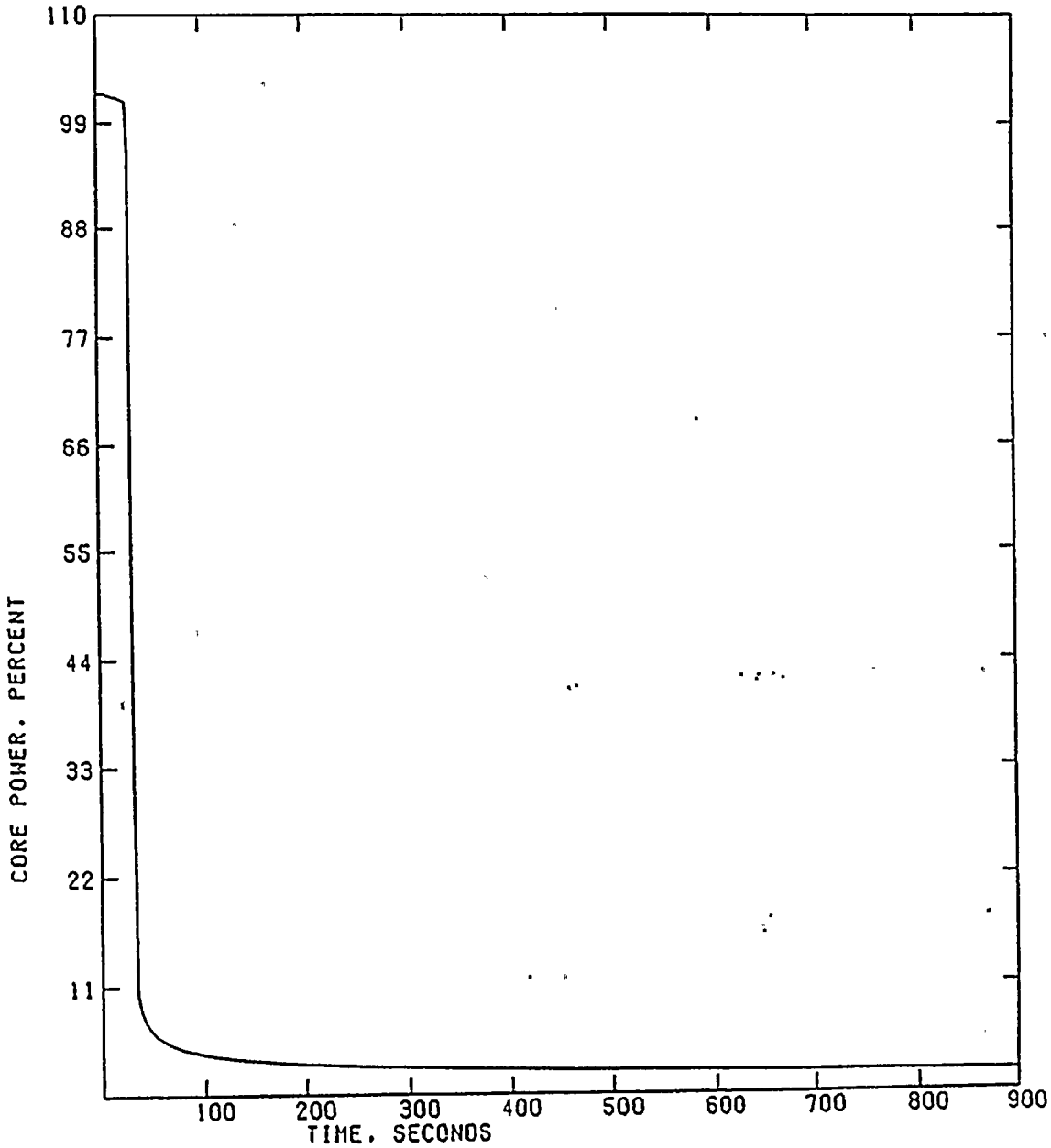

Figure 2.3-1: Core Power Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump

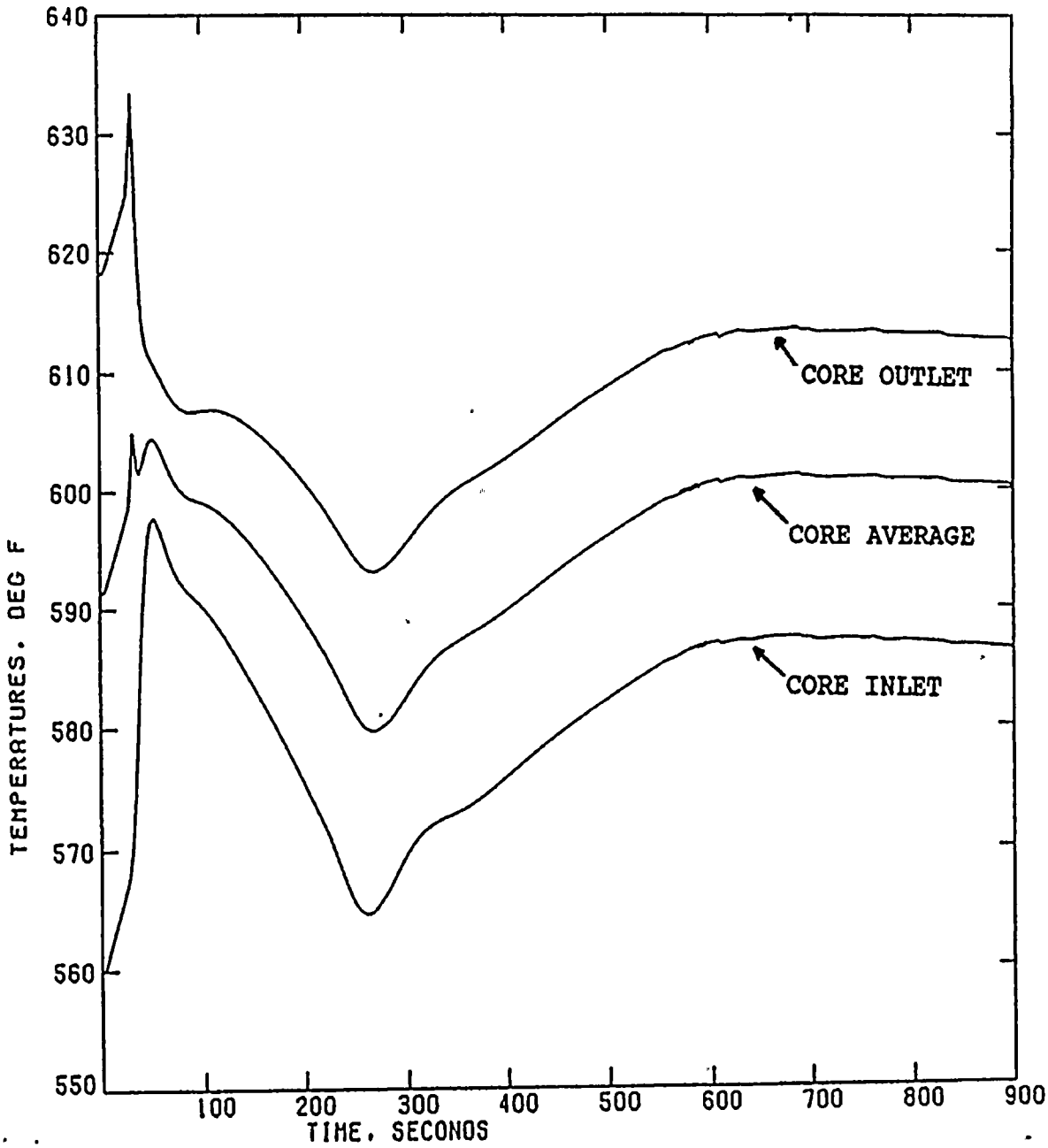

Figure 2.3-2: Core Coolant Temperatures Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One AFW

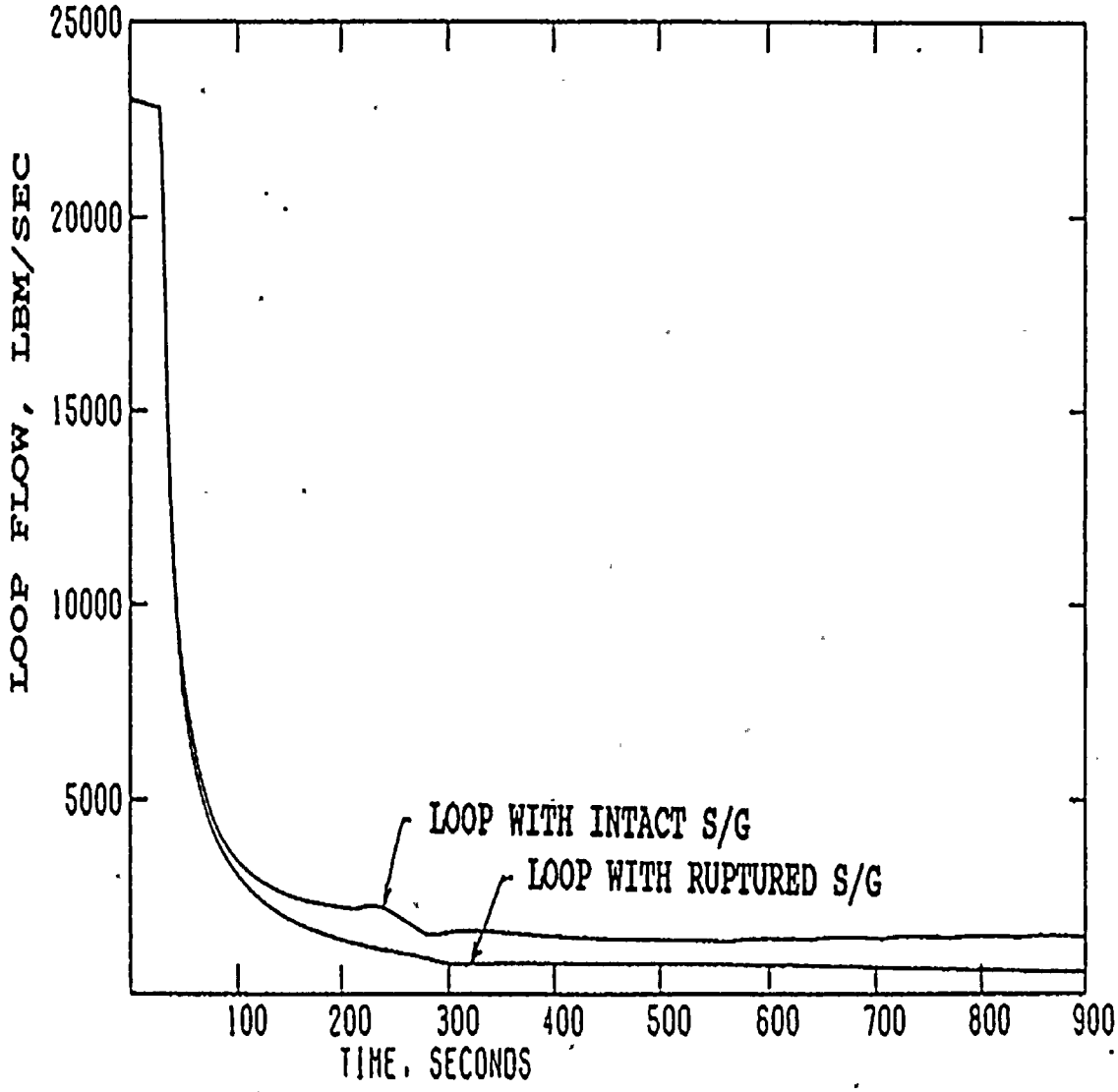

Figure 2.3-3: RCS Loop Flows Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump

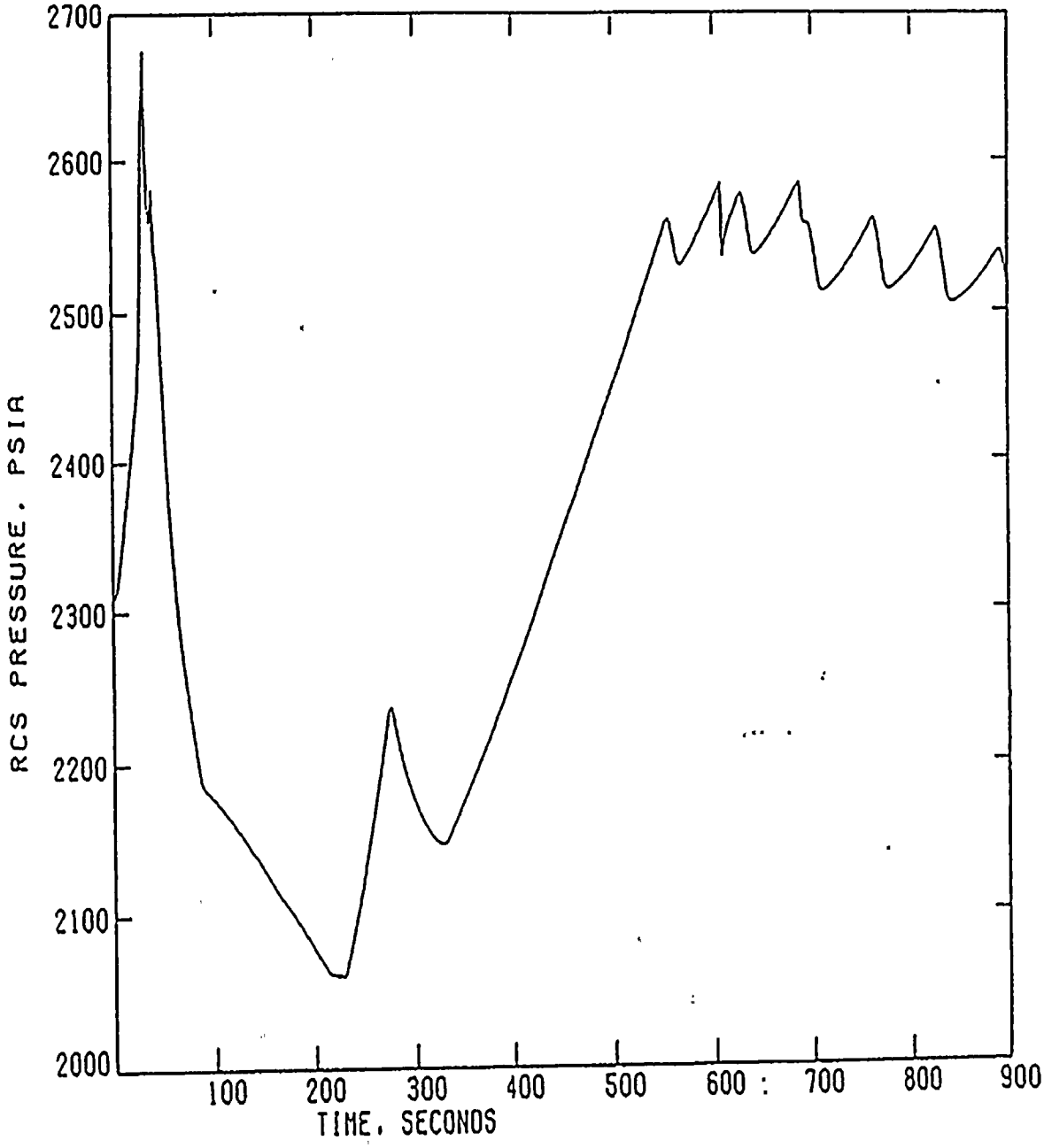

Figure 2.3-4: RCS Pressure Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump

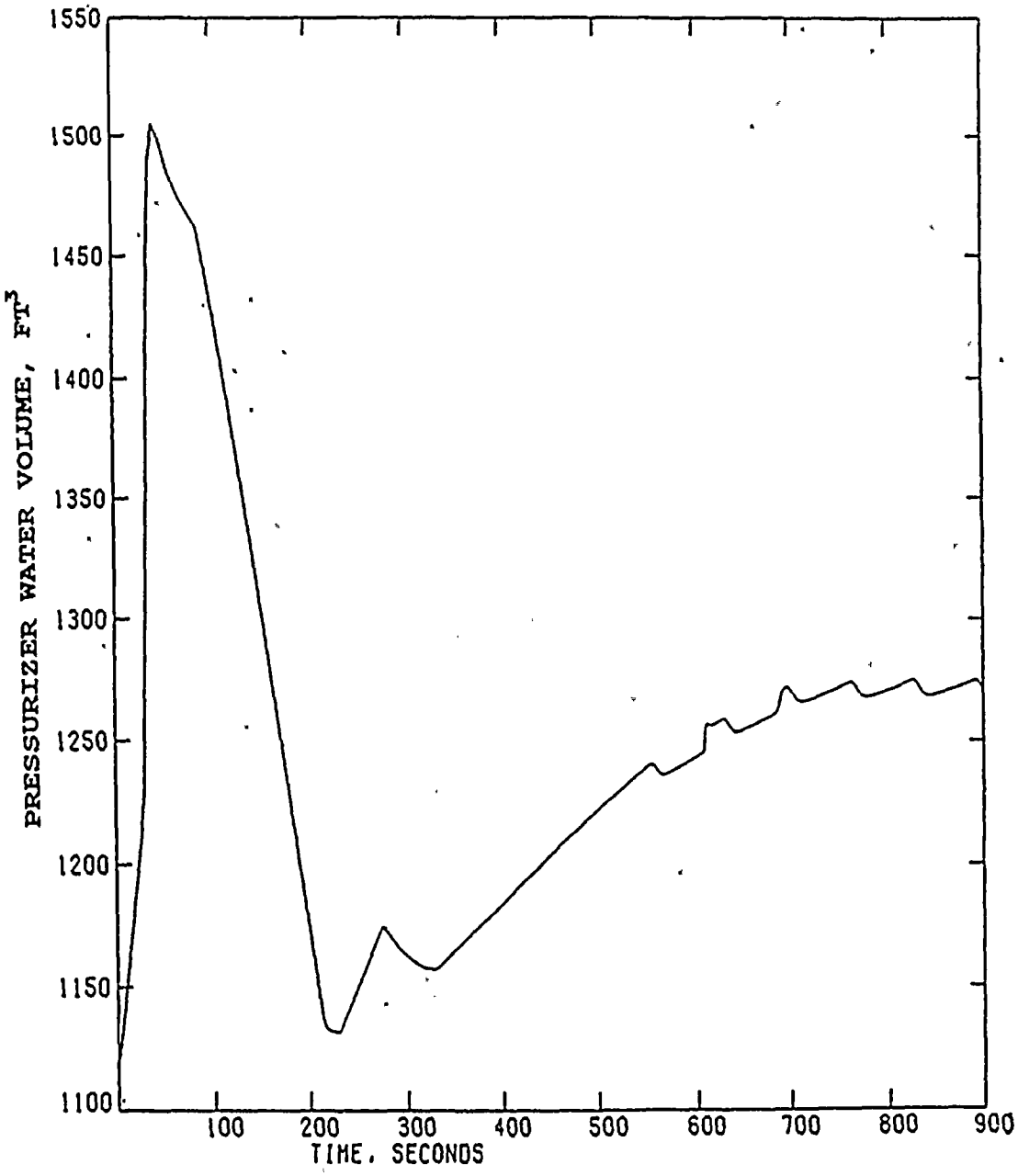

Figure 2.3-5: Pressurizer Water Volume Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One AFW

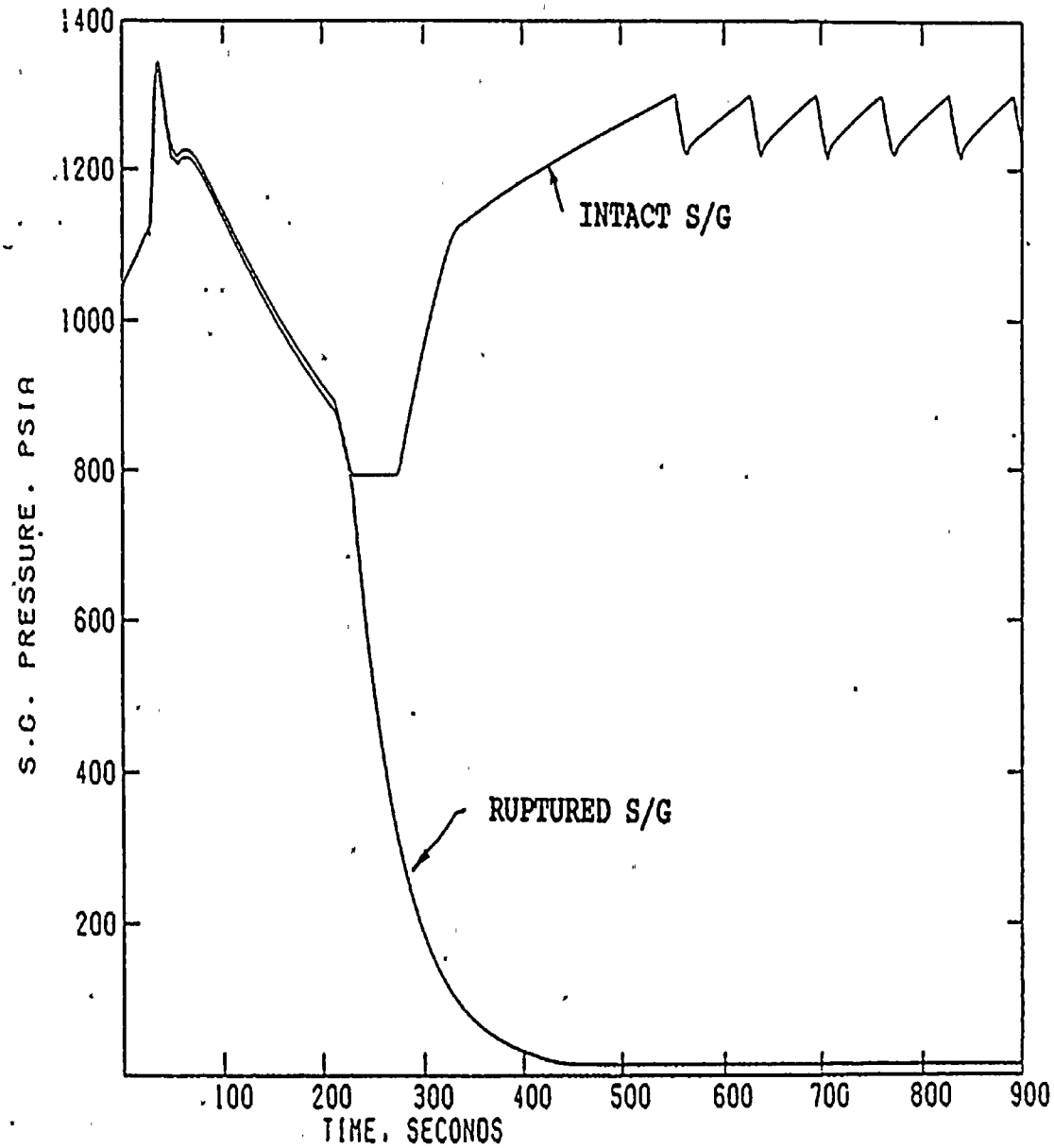

Figure 2.3-6: Steam Generator Pressure Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One AFW

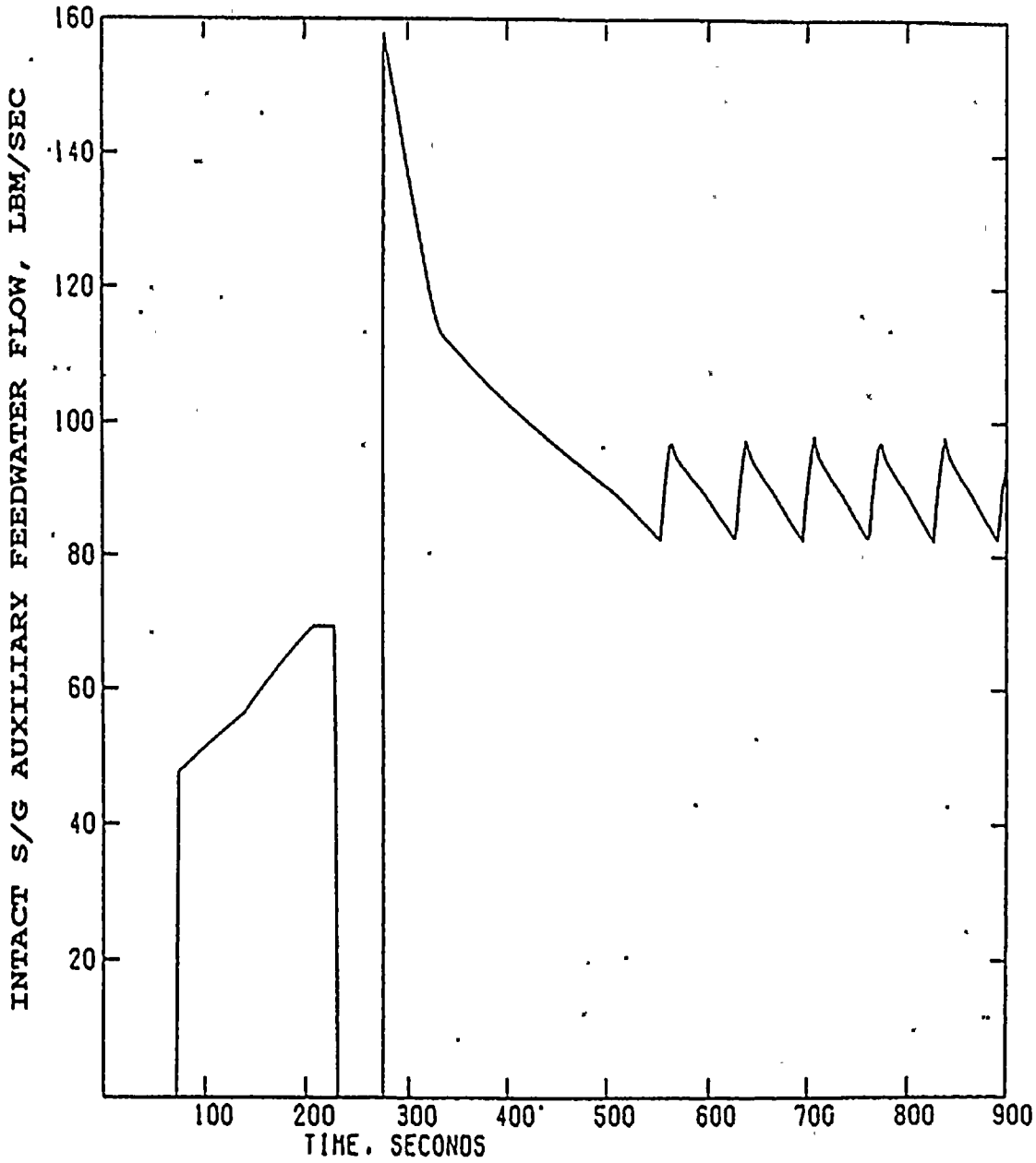

Figure 2.3-7: Intact Steam Generator Auxiliary Feedwater Flow Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One Auxiliary Feedwater Pump

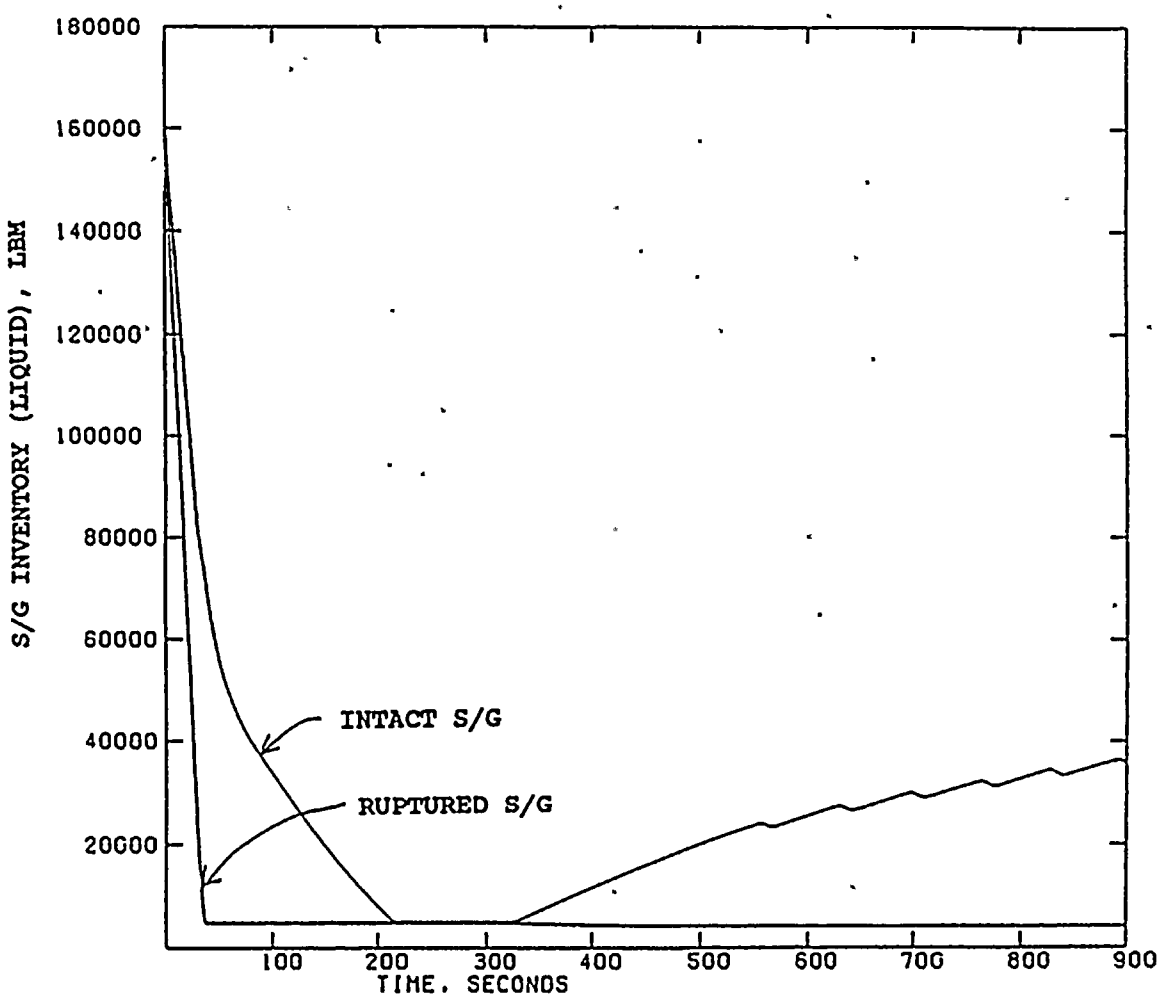

Figure 2.3-8: Steam Generator Liquid Inventory Vs. Time for 0.2 ft^2 Feedwater Line Break with Loss of Offsite Power and the Single Failure of One AFW

2.4 Steam Generator Tube Rupture Events

Two separate analyses, each justifying the acceptability of the proposed changes in Technical Specification have been performed. In the first analysis, the reduction in AFW flow rate from 750 gpm to 650 gpm is considered while in the second, the steam generator analysis for a +3 % tolerance has been performed with the AFW flow rate set at 750 gpm. A review of these analyses is presented below. Even though an SGTR analysis that incorporated all four of the proposed changes to the Technical Specifications has not been performed, there is enough conservatism in the analyses to establish the acceptability of the new Technical Specifications for AFW, PSVs and MSSVs, as the following discussion illustrates.

For the Steam Generator Tube Rupture (SGTR) event with loss of offsite power and a stuck open atmospheric dump valve (ADV) described in the UFSAR Section 15.6.3.2, the reduction in auxiliary feedwater (AFW) flow from 750 gpm to 650 gpm can potentially impact the duration of the affected steam generator tube uncovery. The offsite radiological releases for a SGTR is strongly dependent on the partitioning of the primary to secondary leakage. When the steam generator tubes remain covered, the maximum partition coefficient in the UFSAR analysis is less than 0.1 (i.e., less than 10 % of the leakage goes out through the ADV without any decontamination). However, in the UFSAR SGTR analysis, the partition coefficient is conservatively assumed to be 1 when the tubes remain uncovered. Therefore, an increased duration of uncovery could potentially increase the offsite radiological releases. A conservative calculation was performed to qualify the extended tube uncovery period and the resulting increase in offsite dose. This calculation showed an increase in the duration of the tube uncovery from about 14.8 minutes to approximately 18.3 minutes

(less than 24 percent). The calculation was based on a steam generator mass balance which accounted for primary to secondary leakage, ADV flow, and AFW flow.

Since a significant portion of the offsite radiological releases occur during the period of tube uncovery, it was conservatively postulated that offsite doses will also increase in proportion to the increase in the duration of the tube uncovery. For the most limiting SGTR scenario (pre-existing iodine spike case), this results in an increase of the 2 hour thyroid dose from 200 rem to less than 248 rem. This value is considered to be a conservative upper bound which still provides an adequate margin to the 10 CFR Part 100 guideline of 300 rem. Thus, the decrease in AFW flow from 750 gpm to 650 gpm would result in offsite doses which are still acceptable based on 10 CFR Part 100 guidelines.

The increase in PSV and MSSV setpoint tolerances from the current values to their new values resulted in a 5 % increase above the currently reported 2 hour site boundary doses for the SGTR analysis with loss of offsite power and stuck open ADV. This Technical Specification change alone would mean an increase of the 2-hour thyroid dose from less than the 200 rem to about 210 rem.

Making a conservative assumption that the impact of the two Technical Specifications is cumulative, the total 2-hour dose would increase to 1.24 * 1.05 * 200 = 260 rem. This value is considered to be an upper bound which still provides an adequate margin to the 10 CFR Part 100 guideline of 300 rem.

2.5 RCP Shaft Seizure with Loss of Offsite Power

RCP shaft seizure is examined for dose consequences. In order to bound the potential release to atmosphere via the unaffected steam generator the following heuristic argument is used. The unaffected steam generator is the only source of additional steam release (and thus additional dose) because at 1800 seconds into the event, the ADV of the affected steam generator is assumed to stick open and all the mass inside the steam generator is assumed to be released to the atmosphere. This release is already included in the 2-hour dose calculation. Hence the increased steam release due to lower AFW in the affected steam generator, prior to 1800 seconds, need not be considered.

The relationship between the decrease in AFW flow and the increased steam release was previously expressed heuristically as shown in Equation (1) below:

$$(Wg,2 - Wg,1)\ DT = (hf-hfw) * (Wfw,1 - Wfw,2)\ DT / hfg \quad (1)$$

where:

'Wg,1' and 'Wg,2' represent the two different steam flow rates due to the above AFW flow rates, 'DT' represents the time period for which the AFW flows to the steam generator are maintained, 'hfw' represents the feedwater enthalpy, and 'hf' and 'hfg' have their usual enthalpy definitions.

From UFSAR Table 15.3.3-1 and Figure 15.3.3-10, the AFW flow to the unaffected steam generator begins at 263 seconds and stops at 1460 seconds. The feedwater enthalpy, hfw, during this time is 88 BTU/lbm. Since the steam generator is 1200 psia, hfg = 612 BTU/lbm and hf = 571.7 BTU/lbm based on steam tables.

Using these values in Equation (1), the additional amount of steam released by the unaffected steam generator is:

(571.7-88.0) BTU/lbm*(750-650) gpm*(1460-263) secs*
[(0.1337 ft3/gallons)/(0.0162 ft3/lbm)/
(612 BTU/lbm)/(60 secs/min)]

= 13,013 lbm

UFSAR Table 15.3.3-1 previously reported the total steam release to the atmosphere for this event as 1,128,293 lbm. Based on this total release, this reduction in AFW delivered flow rate causes less than a 1.2 % increase in the total amount of steam released to the atmosphere.

The increase in 2-hour thyroid inhalation dose at the exclusion area boundary due to the increase in steam released to the atmosphere was determined to be less than 0.5 rem. This dose increase is negligible compared to the 191 rem reported for this event in the FSAR and to the 300 rem acceptance criteria for this event.

As a result of the proposed PSV/MSSV tolerance change, the steam release to the atmosphere during the initial phase when the MSSVs cycle will be different from the one presented in the UFSAR. Note that the UFSAR analysis assumes that all the liquid/steam in the steam generator with the stuck open ADV escapes to the atmosphere is still valid. This means that the dose release from this steam generator will not be impacted as a result of changes in the MSSV set point. For the intact steam generator, however, less than 5

% of the total flow released to the atmosphere (half of 120,398 lbm out of a total of 1,128,293 lbm) is released during the MSSV cycle phase.

Thus the change in the tolerance will only affect this 5 % of the total flow only and there is no significant change in the activity release to the atmosphere.

2.6 Loss Of Coolant Accident (LOCA) Evaluation

An engineering evaluation for both small break LOCA and large break LOCA was done by Combustion Engineering to support the PSV and MSSV Technical Specification changes. In the LOCA analysis there is a loss in primary system pressure. Therefore, for both large and small break LOCAs, the PSVs are not challenged. Likewise, in the licensing analyses for the large break LOCA, heat transfer from the primary to the secondary side in the steam generators is not credited. Therefore, the large break licensing analysis is not impacted by the change in tolerance to the MSSVs. However, for the small break LOCA, heat transfer via the steam generators is credited.

An evaluation of the limiting small break LOCA licensing analysis for Palo Verde Units 1, 2 and 3 was performed to ascertain the impact of the Technical Specification changes. This evaluation of the licensing calculations concluded that the existing small break LOCA calculations conservatively bound the tolerance changes.

3 NATURAL CIRCULATION COOLDOWN

During natural circulation cooldown, both primary and secondary pressure decrease with time from their nominal pressure and reach pressures well below the set point pressures of the PSVs and MSSVs

shortly into the cooldown mode. The Safeties are not required to operate and neither is the HPPT. Therefore, these three Technical Specification changes do not impact the natural circulation cooldown. The effect of reducing the auxiliary feedwater flow from 750 gpm to 650 gpm is estimated to have no impact on plant natural circulation cooldown based on Branch Technical Position RSB 5-1. This is because the auxiliary feedwater (AFW) rate of 650 gpm per pump is more than adequate to remove the decay heat as well as primary and secondary stored energy for plant cooldown without significantly changing the steam generator inventory.

For the System 80 design, the analysis of BTP RSB 5-1 natural circulation cooldown is presented in Reference 5. For this analysis, the largest AFW flow rate is required during the first hour hot standby period. This is estimated to be about 650 gpm total to ensure that all of the decay heat is removed while replenishing steam generator inventory. An AFW flow rate of 650 gpm per pump is therefore more than adequate to maintain hot standby conditions. After the first hour of the cooldown, the AFW flow rate requirement is about 310 gpm to accomplish heat removal and cooldown. The RCS cooldown rate is not impacted since significantly more AFW flow is available than required. Thus, there is no adverse impact due to the reduction in AFW flow rate (750 gpm to 650 gpm) on plant natural circulation cooldown per BTP RSB 5-1. The requirement on Condensate Storage Tank capacity would not be affected since RCS cooldown to shutdown cooling entry conditions would be accomplished within the same time period and using similar AFW flow rates as identified for the case in Reference 5.

4 REFERENCES

1. PVNGS, Docket Nos. STN-50-528/529/530 Licensee Event Reports (LER), 528-88-014, 528-89-007, 528-89-010, 529-88-014, and 529-89-002.

2. Docket No. 50-275, OL-DFR-80, Docket No.50-323, OL-DFR-82, Diablo Canyon Units 1 & 2, Licensee Amendment Requests 89-11, Revision of Technical Specification 3.4.2.1, 3.4.2.2, Table 3.7-2 and Associated Bases to Inverse Setpoint Tolerances for Safety Valves.

3. CEN-227, " Summary Report on Operability of Pressurizer Safety Valves in C-E Designed Plants", Nuclear Power System Division, December 1982.

4. PVNGS UPDATED FSAR for Licensing for Unit 1, Unit 2 and Unit 3: NRC docket No. STN 50-528/529/530.

5. Docket No. STN 50-470F, LD-83-074, "Natural Circulation Cooldown Re-Analysis for CESSAR-F," CE letter, A.E. Scherer (CE) to D.G. Eisenhut (NRC) dated August 12, 1983.

ATTACHMENT 2

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

Attachment 2

Basis for No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulations, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability of consequences of an accident previously evaluated: or (2) create the possibility of a new or different kind of accident from any accident previously evaluated or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes, namely:

a) reduction of High Pressurizer Pressure Trip (HPPT) response time,
b) reduction of the minimum acceptable flow rate for the auxiliary feedwater flow pump surveillance requirement, and
c) changing of the lift setpoint tolerances for the Pressurizer Safety Valves (PSVs), the Main Steam Safety Valves (MSSVs),

do not involve any modification to the system or the manner in which it is operated. Therefore, there is no increase in the probability of any accident previously evaluated.

Additionally, all accidents previously analyzed in Chapter 6 and 15 of the PVNGS UFSAR have been reviewed to address the impact of this change on accident consequences. In a number of the events previously analyzed, the above changes do not play a significant

role that could be adversely impacted by the above changes. In all those cases where the above changes have any impact, either a new analysis was performed or a review of the previous analyses was performed to provide justification for acceptability of the events. Specifically the following six analysis/reviews were performed:

1. Loss of condenser vacuum.
2. Feedwater system piping breaks.
3. Steam generator tube rupture.
4. RCP shaft seizure.
5. Loss of coolant accident.
6. Natural circulation cooldown.

Loss of Condenser Vacuum (LOCV) Event

The Loss of Condenser Vacuum (LOCV) event has been reanalyzed using higher setpoint tolerances of + 3% instead of + 1% for both PSVs and MSSVs. The proposed HPPT response time of 0.5 sec instead of 1.15 sec was also used in this analysis. The reduction in HPPT response time is relied on for an earlier trip once the HPPT set point is reached. The HPPT response time of 0.5 sec, although less than 1.15 sec that was used earlier is substantially greater than the less than 0.3 second values that have been observed during the surveillance tests for all the three units. It is also high enough to accommodate any changes in the response time as the equipment ages.

Increasing the tolerance of the PSVs and MSSVs delays the opening of the PSVs and MSSVs. There is no physical modification for either of these types of valves, nor are there changes in the manner in which they are operated. For this deterministic analysis using CESEC III, the PSVs/MSSVs are assumed to open at the set point plus

3% tolerance level. The peak pressures of 2740.9 psia for the RCS and 1369.6 psia for the secondary side even with these conservative and deterministic analysis is lower than the Standard Review Plan acceptance criteria. The four PSVs are in parallel, as are the twenty MSSVs (i.e., the respective valves see same pressures) and it is more likely that some of the safety valves will open at pressures below those used in the analyses. This will further reduce the peak pressures that have been obtained for this event.

AFW is not initiated during the early part of the transient and is required only for long term heat removal. Longterm cooling is adequately covered by 650 gpm as discussed below.

Feedwater Line Break (FLB) Events

The Feedwater Line Break (FLB) events described in the UFSAR, Section 15.2.8, were reanalyzed by incorporating the proposed changes into the FLB analyses for RCS and SG overpressurization (UFSAR Appendix 15E), and into the FLB analysis for long term RCS heat removal (UFSAR 15.2.8).

The FLB analyses assume that the PSVs and MSSVs open at pressures equal to their respective setpoints, plus a 3 % tolerance. Additionally, the FLB analyses for RCS and SG overpressurization included the following changes:

1) the proposed HPPT response time of 0.5 seconds,
2) PSV setpoint area fractions increased from 0.7 to 0.99.[1]

[1] Justification provided in CEN-227, "Summary Report on Operability of Pressurizer Safety Valves in C-E Designed Plants," Nuclear Power System Division, December 1982.

3) Surge line form loss coefficients reduced from 3.9 to 3.0, to reflect actual PVNGS design. This change is analytically justified in a calculation.

The proposed reduction in AFW flow does not impact RCS and SG peak pressures, since the pressure peaks occur before actuation of the AFW.

The increase in the PSV setpoint tolerance from +1 % to +3 % would, by itself, increase the RCS peak pressure for these events. The pressure increasing effect of the setpoint tolerance change is offset, however, by the additional, mitigating changes noted above, such that the RCS peak pressures for the large and small FLB events (2816 and 2668 psia, respectively) are bounded by the corresponding UFSAR results. The SG peak pressures for the large and small FLB events (1358 and 1369 psia, respectively) exceed the corresponding UFSAR values (1318 and 1342 psia), but are bounded by the SRP overpressurization limiting criteria of 1500 and 1375 psia.

In addition to the setpoint tolerance change, the reanalysis of the FLB event for long term RCS heat removal (UFSAR 15.2.8) included items (2) and (3), above, and the proposed reduction in AFW flow from 750 to 650 gpm. At 900 seconds, the liquid inventory in the intact SG is 30,800 lbm (adjusted for dry out conditions), compared to 51,000 in the current UFSAR analysis, and RCS heat removal is being maintained by the MSSVs. Additionally, a steady-state condition is achieved in the NSSS, with core temperatures subcooled and stable, RCS pressures below the PSV setpoint, and a pressurizer liquid inventory of approximately 1500 ft^3. These results demonstrate that the reduced AFW flow is sufficient to

satisfy the long term RCS heat removal requirements with the proposed amendments.

Steam Generator Tube Rupture (SGTR) Event

The Steam Generator Tube Rupture event, with loss of offsite power as described in the UFSAR Section 15.6.3.2, has been reevaluated for both the safety valve set point tolerance change and the proposed reduction in auxiliary feedwater flow rate. In this scenario, the main concern is the radiological release. With respect to the SGTR, the reduction in auxiliary feedwater flow delivery rate alone results in an increased period of time during which the tubes are uncovered minimizing the Iodine partitioning of released activity. This can potentially increase the 2-hour thyroid inhalation dose at the exclusion area boundary by 24 % (200 rem to 248 rem). The impact of the PSV and MSSV set point tolerance increase showed an increase of 5 % above the 200 rem dose rate that is currently on the docket for PVNGS units. Thus due to the increase of tolerance limits alone, the 2-hour thyroid dose increase would change from 200 rem to 210 rem. Making a conservative assumption that the dose rate for the changes are cumulative, the total 2-hour dose would increase from 200 rem to 260 rem.

Thus the 2-hour dose rates conservatively estimated at 260 rem, still provides adequate margin to the SRP criteria.

RCP Shaft Seizure

With respect to the RCP shaft seizure event, the UFSAR analysis demonstrated that limited fuel failure would occur resulting in

fission product release. Maximum RCS pressure reached for this event is 2387 psia. This is less than the lowest possible opening of PSV of 2475 psia (corresponds to 2500 - 1 % psia) which has not changed as a result of the proposed Technical Specification changes. MDNBR occurs at 2 sec and is not affected by the safety setpoint. Thus no additional fuel failure occurs as a result of the proposed amendment.

Assuming the maximum allowable steam generator tube leakage rate of 1 gpm, the increased steam release caused by a reduction in auxiliary feedwater flow has been conservatively calculated to result in a 0.5 rem increase in the 2-hour thyroid inhalation dose at the exclusion area boundary. This increase is insignificant compared to the 191 rem dose reported in the UFSAR and the 300 rem acceptance criteria of the SRP.

Loss of Coolant Accident

An engineering evaluation for both small break LOCA and large break LOCA was done to support the proposed PSV and MSSV Technical Specification changes. In the LOCA analysis there is a loss in primary system pressure. Therefore, for both large and small break LOCAs, the PSVs are not challenged. The analyses for the large break LOCA does not credit heat transfer from the primary to the secondary side of the steam generators. Therefore, the large break analysis is not impacted by a change in the MSSVs tolerance. However, for the small break LOCA, heat transfer from the RCS to the steam generators is credited. An evaluation of the limiting small break LOCA analysis for was performed to ascertain the impact

of the proposed Technical Specification changes. This evaluation concluded that the existing small break LOCA calculations conservatively bound the tolerance changes.

Natural Circulation Cooldown

An evaluation of long term post-accident cooling concluded that the natural circulation cooling capability of the plant will not be affected by this change. Specifically, the maximum total auxiliary feedwater flow rate required during the event previously analyzed in accordance with Branch Technical Position RSB 5-1 is 650 gpm during the first hour of the four hour hot standby period. A reduction in pump capacity to 650 gpm, therefore, will not affect the ability of the auxiliary feedwater system to maintain the plant in a hot standby condition. In all aspects of the scenario, therefore, the BTP RSB 5-1 analysis of record remains bounding.

Based on these analyses, the amendment request does not (1) involve a significant increase in the probability or consequences of the previously evaluated FLB analyses.

(2) The proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since no changes in plant equipment or operating procedures have been made or proposed, no additional accidents of a new or different kind will be created.

(3) The proposed amendment would not involve a significant reduction in margin of safety.

Attachment 2

Basis for No Significant Hazards Consideration

The combined effects of reduced AFW flow rate, reduced HPP trip response time, and changes in PSV and MSSV set point tolerances do not involve a significant reduction in the margin of safety. The results of the analysis presented in Safety Evaluation demonstrated only a minor decrease in the margin to some of the acceptance criteria. The RCS and steam generator integrity were maintained for all affected events. Fuel integrity was demonstrated for the proposed changes by maintaining SAFDL in those analyses for which fuel failure did not occur and by not increasing the number of failed fuel pins for those events where fuel damage was a consideration. All the analyses performed demonstrated acceptability when compared to the acceptance criteria of the Standard Review Plan.

Additionally, the NRC staff has provided examples of license amendments that are considered not likely to involve significant hazards considerations. This license amendment request is most like example (vi) "a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan," 51 FR 7759 (March 6, 1986). In this case, the increase in radiological dose consequences of the steam generator tube rupture and RCP shaft seizure events clearly remains within the 300 rem acceptance criteria of the Standard Review Plan.

Based on the above, it has been determined that the amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident

from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

ATTACHMENT 3

ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

Attachment 3

Environmental Impact Consideration Determination

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Units 1, 2, and 3 in accordance with this change, would not:

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

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

As discussed in Attachements 1 and 2 of this amendment, no significant reduction in safety and no new accidents are introduced by this change. This amendment does not change the the installed plant equipment, effluents, or operating power levels, and has no environmental impact.