APS

David Mauldin
Vice President
Nuclear Engineering
and Support

Palo Verde Nuclear
Generating Station

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

Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

10 CFR 50.90

102-04866-CDM/TNW/RAB
November 21, 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 No. 102-04847- CDM/TNW/RAB, dated October 11, 2002, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request"

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Unit 2, Docket No. STN 50-529
Supplement to Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations

In Reference 1, Arizona Public Service Company (APS) submitted a license amendment request to support steam generator replacement and uprated power operations for Unit 2. 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. These issues were reported to the NRC in Reference 2. One of these issues identified concerns with the analysis for the Feedwater Line Break (FWLB) with Loss of Offsite Power (LOP) – Long Term Cooling Event, and APS stated that a description of the issue and results of the new analysis would be provided to the NRC by November 22, 2002.

During the work to support CENTS implementation for the PVNGS units, APS questioned the existing methodology for FWLB – Long Term Cooling Event regarding the assumptions made for the response of several Engineered Safety Features Actuation Systems (ESFAS) during the transient. It was determined that certain

A001

assumptions would be changed and the postulated event was reanalyzed. In accordance with 10 CFR 50.59, APS has determined that the new analysis for FWLB with LOP – Long Term Cooling Event requires NRC approval prior to implementation. Therefore, APS hereby supplements Reference 1 by providing a description of the issue and the results of the new analysis. Attachment 2 provides the description of the issue and Enclosure 1 provides replacement pages for Attachment 6 to Reference 1.

The No Significant Hazards Consideration provided in Reference 1 remains valid for the original submittal, including this supplement.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed supplement. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10CFR 50.91(b)(1).

Since this supplement must be reviewed and approved by the NRC Staff in conjunction with the review and approval of the amendment request submitted in Reference 1, APS understands that the staff will need more time for its review. Therefore, APS requests that the NRC approve this supplement and the amendment requested in Reference 1 by March 31, 2003.

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

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

Sincerely,

David Mauldin

CDM/TNW/RAB/kg

Attachments:

1. Notarized Affidavit
2. Issue Description: Feedwater Line Break with Loss of Offsite Power – Long Term Cooling Event

Enclosure 1:

Power Uprate Licensing Report Replacement Pages

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

Attachment 1

Notarized Affidavit

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

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

David Mauldin

David Mauldin

Sworn To Before Me This 21st Day Of November, 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

Issue Description:
Feedwater Line Break with Loss of Offsite Power –
Long Term Cooling Event

1.0 Summary

In the process of converting existing safety analyses from CESEC to CENTS, several issues were identified regarding assumptions for the existing analysis for the postulated Feedwater Line Break (FWLB) with Loss of Offsite Power (LOP) and Single Failure Long-Term Cooling Event. This event has been reanalyzed to address these issues, and the reanalysis includes several changes in input parameters and elements of the methodology. The revised analysis verifies that the acceptance criteria related to long-term heat removal and Pressurizer Safety Valve (PSV) qualification continue to be satisfied for operation at 3990 MWt, as well as at the current licensed power level of 3876 MWt.

Several changes were made to the input parameters and elements of the methodology for this event. Of these changes, one analytical change involves an element of the methodology that constitutes a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2). This change, which requires NRC staff review and approval, is as follows:

- For the purpose of establishing the initial Reactor Coolant System (RCS) cold leg temperature for the analysis, it is assumed that the plant is operated on program T_{avg}, and the Pressurizer Level Control System (PLCS) is in the automatic mode of operation at the beginning of the event.

In this mode of operation, there is a correlation between pressurizer water level and RCS loop average temperature (T_{avg}), as determined by the PLCS program. As T_{avg} increases, pressurizer level will increase from a programmed minimum level to a programmed maximum level. Therefore, for the revised analysis, the initial pressurizer level was conservatively set to the Technical Specification maximum level, and T_{avg} was set to its maximum programmed setpoint.

Note that, since the plant is assumed to be operating on program T_{avg}, the initial RCS cold leg temperature is no longer assumed to be at the minimum allowed by the Technical Specifications. The assumed initial cold leg temperature is now the cold leg temperature that is associated with program T_{avg} at hot full power (HFP). Any reduction in actual cold leg temperature will result in a reduction in actual T_{avg}, and a corresponding reduction in pressurizer level.

All other analytical changes were screened and evaluated in accordance with 10 CFR 50.59, and were determined to not require NRC staff review and approval. These changes involve minor corrections to input parameters, removal of discretionary conservatism from input parameters, and new or revised elements of the methodology. The changes to the input parameters were determined to be not “adverse” as defined in NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation, and therefore screened out. The new or revised elements of the methodology yield analytical results that are conservative with respect to, or essentially the same as, previously approved methods. Therefore, these additional changes do not require NRC staff review and approval.

The revised FWLB with LOP and Single-Failure Long-Term Cooling Event analysis is described in greater detail below. Enclosure 1 provides replacement pages to Attachment 6 of Reference 1.

2.0 Introduction

This submittal describes the reasons for, and the results of, the revised postulated FWLB with LOP and Single Failure Long-Term Cooling Event analysis.

The FWLB with LOP and Single Failure Event is analyzed to address UFSAR Chapter 15, Accident Analyses, licensing basis acceptance criteria for peak RCS and main steam system pressures, fuel integrity, and radiological dose. APS has traditionally analyzed the FWLB with LOP and Single Failure Long-Term Cooling Event as the most limiting event for demonstrating long-term Auxiliary Feedwater (AFW) capacity for removal of decay and sensible heat (UFSAR Chapter 10, Steam and Power Conversion System), and PSV adequacy for overpressure protection (UFSAR Chapter 5, Reactor Coolant System and Connected Systems). Also, the long-term PSV operating conditions are compared with the results of PSV qualification tests, as required by Item II.D.1 of NUREG-0737, "The Three Mile Island (TMI) Action Plan" (UFSAR Chapter 18, TMI-2 Lessons Learned Implementation Report).

The licensing basis safety analysis methodology for this event was originally developed to ensure conservative analytical estimates for short-term RCS peak pressure. The event analyses were subsequently modified to address long-term RCS heat removal and PSV operability. The revised analyses used analytical inputs and assumptions that were chosen to exacerbate the RCS heat load and pressurizer water level to provide a conservative determination of AFW capacity and PSV operating conditions.

APS engineers questioned whether the FWLB with LOP and Single Failure Long-Term Cooling Event analysis conservatively modeled the Nuclear Steam Supply System (NSSS) response to the postulated event. Specifically, it was noted that the existing licensing basis analysis did not include the effects of containment pressure which, for a FWLB inside the containment, would initiate several Engineered Safety Features Actuation Sytems (ESFAS). These effects are the receipt of a Safety Injection Actuation Signal (SIAS), Main Steam Isolation Signal (MSIS), Containment Isolation Actuation Signal (CIAS), and High Containment Pressure Trip on high containment pressure. Of these effects, the receipt of a CIAS has no affect on the analysis, and the High Containment Pressure Trip is conservatively not credited in the analysis. The receipt of SIAS and MSIS may have an adverse effect on long-term RCS heat removal and PSV operability. The effect of the receipt of the SIAS and MSIS is described below.

The SIAS, in combination with a LOP following a turbine trip, would cause the restart of a charging pump as described in UFSAR Table 8.3-3, "Diesel Generator Load Sequencing". Restarting a charging pump results in continuous inventory addition to the RCS for the duration of the event. The additional inventory addition results in more adverse effects with respect to long-term PSV operability criteria for this event. Note that although all three charging pumps will have a permissive to start, only the "always running" charging pump will start since pressurizer level will be well above the PLCS setpoint.

Although the existing method of evaluation models a MSIS, it is delayed until low steam generator pressure initiates the MSIS since an earlier MSIS was previously deemed to result in more benign consequences. However, the APS review determined that an earlier MSIS may be more adverse depending on the location and the size of the break. For FWLBs inside containment, an earlier MSIS would occur on high containment pressure and would reduce blowdown through the break and heat removal from the intact steam generator via the main steam header downstream of the Main Steam Isolation Valves.

Based on the above discussion, it was concluded that incorporation of these adverse effects of high containment pressure into the long-term analysis could further aggravate the long-term RCS heat load and PSV operating conditions, and would be conservative with respect to the current licensing basis Analysis of Record (AOR) for the FWLB with LOP and Single Failure Long-Term Cooling Event. These effects, however, have no adverse impact on the RCS peak pressure, fuel integrity, and radiological dose consequence analyses for the FWLB with LOP and Single Failure Event.

For the reasons stated above, the FWLB with LOP and Single Failure Long-Term Cooling Event described in Attachment 6 of Reference 1 was reanalyzed. The revised analysis retains the following principal conservative assumptions documented in NUREG-0852, "Combustion Engineering Standard Safety Analysis Report (CESSAR) Safety Evaluation Report," Supplement 2, Appendix G, dated September 1983:

- delaying the reactor trip until a High Pressurizer Pressure Trip (HPPT) condition occurs, which is coincident with emptying of the faulted steam generator, i.e., no reactor trip upon a low SG level until all of the liquid inventory within the affected SG is depleted,
- retaining the original blowdown characteristics of the break, i.e., the break flow and enthalpy are maintained at zero quality until all of the liquid inventory is depleted,
- retaining the SG heat transfer models, i.e., at the time when all liquid inventory is depleted from the affected SG, the heat transfer coefficient is set to zero as a step function,
- delaying the AFAS until liquid mass in the affected SG is depleted, and
- not taking credit for any operator action for the first 30 minutes of the transient.

The postulated event sequence includes the single failure of an AFW pump, consistent with the current licensing basis. The analysis has been revised, however, to model a high containment pressure signal coinciding with the HPPT, which initiates a MSIS and quickly isolates the intact steam generator from the break. Additionally, the analysis has been revised to model a SIAS due to high containment pressure. Although RCS pressure remains high enough to preclude the injection of borated water into the RCS, the SIAS, in combination with the LOP following the turbine trip, results in the automatic restart of one charging pump.

The reanalysis also includes several changes to input parameters and elements of the methodology, which are described below in Section 3.0. One change in an element of the methodology, which involves the selection of the initial RCS cold leg temperature for the analysis, constitutes a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2), and therefore requires NRC staff review and approval.

3.0 Evaluation

As described in Section 2.0 above, the current NRC-approved analysis for the postulated FWLB with LOP and Single Failure Long-Term Cooling Event does not include the effects of high containment pressure which, for a FWLB inside containment, would initiate several Engineered Safety Features Actuation Systems (ESFAS), including MSIS and SIAS. If these effects are incorporated into the analytical methodology, they may result in more adverse consequences with respect to long-term AFW capacity and PSV operating conditions. However, they would not adversely affect the short-term RCS peak pressure, fuel integrity, and radiological dose assessment analyses for FWLB. Therefore, only the long-term cooling analysis was revised to model the anticipated ESFAS actuations that may occur following a FWLB inside containment. Additionally, other changes were made with respect to input parameters and elements of methodology. These changes are described in Sections 3.1, 3.2 and 3.3 below.

The change described in Section 3.1 constitutes a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2), and therefore requires NRC staff review and approval. All other analytical changes described in Sections 3.2 and 3.3 involve minor corrections to input parameters, removal of discretionary conservatism from input parameters, and new or revised elements of the methodology. The changes to the input parameters were determined to be not "adverse" as defined in NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation, and therefore screened out. The new or revised elements of the methodology yield analytical results that are conservative with respect to, or essentially the same as, previously approved methods, and therefore, do not require NRC staff review and approval. The following sections describe the changes made in the revised analysis.

3.1 Change in Initial PLCS Mode (Requires NRC Approval)

In accordance with PVNGS design and licensing bases, safety analyses typically model control systems in the mode of operation -- manual or automatic -- that results in the most severe consequences for each analyzed event. This general analysis methodology was noted by the NRC staff in Section 7.7 of NUREG-0852, "Combustion Engineering Standard Safety Analysis Report (CESSAR) Safety Evaluation Report," dated November 1981, as well as Section 7.7 of NUREG-0857, "PVNGS Safety Evaluation Report," dated November 1981, and Section 6.3.0.2 of Attachment 6 of Reference 1. Previous FWLB with LOP and Single Failure Long-Term Cooling Event analyses have modeled the PLCS in the manual mode of operation. In that mode, the initial conditions were established by the assumption that plant operators would utilize control room indications to control both pressurizer water level and RCS cold leg temperature at their Technical Specification (TS) limits. Maximum TS initial pressurizer water level and minimum TS initial cold leg temperature (along with associated instrument uncertainties) were utilized in the previous analyses to ensure a highly conservative prediction of RCS coolant swell effects during the predicted heatup following the FWLB. This, in turn, exaggerated pressurizer water level and the potential for moisture carryover into the PSVs.

For the purpose of establishing the initial RCS cold leg temperature and pressurizer level for the reanalysis of this event, however, it is assumed that the plant is operated on program T_{avg} and the PLCS is in its automatic mode of operation. In this mode of operation, there is a correlation between pressurizer water level and T_{avg}, as determined by the PLCS program. Specifically, for a nominal T_{avg} of 583°F at hot full power (HFP), the PLCS would regulate charging and letdown to maintain pressurizer water level at a maximum of 52.6% of calibrated span. If T_{avg} were to increase above 583°F, the PLCS would continue to maintain 52.6% pressurizer water level. If T_{avg} were to decrease below 583°F, the PLCS would regulate the pressurizer water level in a linear fashion, to a minimum program setpoint of 33% at an RCS loop average temperature of 569°F.

For analytical purposes, the initial pressurizer water level utilized in the reanalysis was established at 59% of calibrated span (i.e., the Technical Specification upper limit of 56% indicated level, plus 3% to account for control room instrument uncertainty). This assumption remains the same as for the previous FWLB with LOP and Single Failure Long-Term Cooling Event analysis, and is conservative with respect to the PLCS automatic control program setpoint (56% vs. 52.6%). This assumption ensures that the predicted pressurizer water level transient response is conservative with respect to the anticipated response of the physical plant.

The assumption of operating the plant on program T_{avg} establishes the initial RCS cold leg temperature for this event. At an initial HFP T_{avg} of 583°F, the initial minimum RCS cold leg temperature corresponds to 555°F for the 3990 MWt Rated Thermal Power (RTP) case, and 552°F for the 3876 MWt RTP case. These temperatures include 2°F for instrument uncertainties. For comparative purposes, it is noted that, had the Technical Specification minimum cold leg temperature been utilized in the reanalysis, the initial RCS cold leg temperature would have been set to 548°F (i.e., the Technical Specification minimum temperature of 550°F, minus 2°F instrument uncertainty) for both power level cases. Therefore, modeling the event with the PLCS in automatic rather than manual, and assuming that the plant is operated on program T_{avg}, resulted in a 7°F (4°F for 3876 MWt RTP case) increase in the assumed initial RCS cold leg temperature. Although the change in initial cold leg temperature amounts to only a few degrees Fahrenheit, it has the effect of reducing the water mass in the RCS, which somewhat mitigates the pressurizer level swell that occurs during system heatup following the FWLB.

Because this change is contingent upon operating the plant on program T_{avg} and modeling the PLCS in the automatic mode of operation, it is outside the constraints and limitations described in Section 7.7 of the CESSAR and PVNGS SERs as noted in the first paragraph to this section. Therefore, this change in assumed PLCS operating mode and operating the plant on program T_{avg} is classified as a change in an element of the methodology, which constitutes a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2). This change, therefore, requires NRC staff review and approval.

The following additional points should, however, be noted with respect to this proposed change in an element of methodology:

- The PLCS is normally maintained in the automatic mode of operation in accordance with station operating procedures. The manual mode.of operation is selected only when necessary to support certain activities (e.g., instrumentation and control maintenance) or in response to control system malfunctions. Station operating experience shows that the manual mode of operation is selected approximately 2 or 3 times during a typical 18-month fuel cycle.

- The likelihood of the initiating event occurring outside the assumed bounds of the analysis can be estimated from the likelihood of the initiating event and the probability of the PLCS system being concurrently in manual mode. The frequency associated with a FWLB for PVNGS is 1.9e-04 per year. The probability of a consequential LOP is 1.9e-03. Thus the frequency for this initiating event is on the order of 4.0e-07 per year. The probability that a PVNGS unit would be operating with PLCS in manual mode of operation may be estimated by assuming each occurrence noted above lasts for three days, which is a conservatively long time assumption to effect the repair of a control system important to plant operation. This leads to a probability of 2.0e-02 that the PLCS is in manual. Therefore, the frequency of this event occurring with PLCS in the manual mode of operation is on the order of 1.0e-08 per year. Regardless of the PLCS operating mode, the frequency of this event is less than 1.0e-06 per year.

- Modeling the PLCS in the automatic mode of operation serves primarily to establish more realistic initial conditions for the event. The PLCS serves little to mitigate the NSSS transient response, because an assumed LOP following a turbine trip (i.e., at about 30 to 35 seconds into the event sequence) effectively disables the normal functions of PLCS. For example, letdown from the RCS, which would otherwise serve to reduce the transient pressurizer water level response, is isolated following the LOP.

3.2 Changes to Include Containment Pressure Effects

The UFSAR currently states that, although a FWLB may result in a reactor trip on high containment pressure, this trip is not credited. This statement, which was originally reviewed by the NRC staff on the CESSAR docket, suggests that a containment pressure trip prior to the credited high pressurizer pressure trip would result in more benign consequences. However, the UFSAR FWLB safety analyses also do not include other effects associated with a high containment pressure condition, including the actuation of MSIS and SIAS. Exclusion of these actuations may be attributed to the fact that the FWLB analyses were originally performed to evaluate RCS peak pressure, and these actuations have no significant impact on the acceptance criteria for that event. When FWLB analyses were later performed to assess long-term cooling for AFW capacity and PSV operability, the licensing basis safety analysis methodology was not revised to include the effects of containment pressurization.

APS personnel determined that consideration of containment pressurization could yield a more adverse transient response for the FWLB with LOP and Single Failure Long-Term Cooling Event case. The analysis was therefore revised to include MSIS and SIAS actuation on high containment pressure. This constitutes a change in an element of the methodology to the analysis. However, because this change would yield results that are conservative with respect to the current licensing basis methodology, it does not constitute a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2). Therefore, this change does not require NRC staff review and approval.

The following paragraphs describe the analytical changes that were made to account for containment pressurization effects:

- Timing of MSIS: UFSAR Section 15.2.8.3.2 currently states that an early MSIS would make the event consequences more benign because more inventory would be retained in the secondary system. The previous FWLB with LOP and Single Failure Long-Term Cooling Event analyses, therefore, assumed a delayed MSIS on low steam generator pressure. A MSIS on low steam generator pressure is expected to occur later than a MSIS on high containment pressure for a FWLB inside containment. Therefore, the previous analyses used MSIS on low steam generator pressure, which was predicted to occur as late as 1 to 2 minutes following the reactor trip.

The validity of this assumption is dependent on the location and the size of the break. During the post-trip period and prior to receipt of the MSIS, the intact steam generator blows down through the break via the main steam header downstream of the Main Steam Isolation Valves, and therefore provides about 1 to 2 minutes of cooling by the secondary system in the existing Analysis of Record (AOR). As stated previously, APS has since determined that an earlier MSIS would reduce this cooling effect and could further aggravate the RCS heat load and PSV operating conditions in the long-term. Therefore, assuming an early MSIS on high containment pressure would be conservative with respect to the current licensing basis AOR.

The revised analysis assumes that a MSIS occurs on high containment pressure, and the high containment pressure condition is assumed to occur simultaneously with emptying of the faulted steam generator and HPPT. The post-trip cooldown period previously afforded by the intact steam generator is now reduced to the few seconds that it takes for the Main Steam Isolation Valves to close. The inclusion of an earlier MSIS in the analytical model has no adverse effect on peak RCS pressure, because peak RCS pressure, which is effectively limited by the HPPT and the PSVs, occurs immediately after the reactor trip and prior to isolation of the intact SG.

- Charging Pump Restart: The current AOR assumes that the charging pumps stop following the LOP, and do not operate throughout the transient. This assumption would be valid if a LOP occurs without a SIAS. However, if a SIAS occurs and a LOP exists, the charging pumps would automatically be sequenced onto the emergency diesel generators when certain conditions are met. This feature is provided as additional protection against a possible Loss of Coolant Accident (LOCA) and LOP.

  If a LOP and SIAS occurs, one or more charging pumps will restart as allowed by the Balance Of Plant (BOP)-ESFAS sequencer and as demanded by the PLCS, depending upon the pressurizer level. Restarting a charging pump results in continuous inventory addition to the RCS for the duration of the event. The additional inventory addition results in more adverse effects with respect to the long-term PSV operability criteria for this event.

  The revised analysis assumes that a SIAS occurs on high containment pressure, at the same time as the HPPT and emptying of the faulted steam generator. Then, following the turbine trip and LOP, the "always running" charging pump is automatically restarted to add inventory to the RCS. The "normally running" and "standby" charging pumps do not restart because of the high pressurizer level condition at the time of the LOP would prevent the PLCS from demanding those pumps. The reanalysis does not take credit for operator action to stop the charging pump during the first 30 minutes of the transient, even though plant operators are trained to do so if the pressurizer continues to fill.

3.3 Other Changes

In addition to the changes described in Sections 3.1 and 3.2, the new analysis includes several other changes that differ from the inputs and assumptions of the Analysis of Record (AOR) presented in Attachment 6 of Reference 1. These changes involve minor corrections to input parameters, removal of discretionary conservatism from input parameters, and new or revised elements of the methodology. The changes to the input parameters were determined to be not "adverse" as defined in NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation, and therefore screened out. The new or revised elements of the methodology yield analytical results that are conservative with respect to, or essentially the same as, previously approved methods, and therefore, do not require NRC staff review and approval. Although these changes do not require NRC review and approval, they are included here for information. The following paragraphs describe the changes made in the new analysis.

- Use of Pressurizer Water Volume to Verify Acceptance Criteria: NUREG-0852, "Combustion Engineering Standard Safety Analysis Report (CESSAR) Safety Evaluation Report," Supplement 2, Appendix G, dated September 1983, documented the NRC's approval of the FWLB analysis methodology. In Section 15.3.2 of that SER, the NRC staff concluded that adequacy of the AFW design for long-term RCS heat removal capability was demonstrated, and noted that the pressurizer did not fill solid during the event. Subsequently, in Section 5.2.2 of NUREG-0857, "PVNGS Safety Evaluation Report," Supplement 8, dated May 1985, the NRC staff concluded that a FWLB with LOP analysis was acceptable to approve design changes that resulted in increasing PSV minimum blowdown, noting that the pressurizer level remained well enough below the bottom of the PSV inlet nozzles to prevent moisture carryover through the PSVs. Therefore, within these SERs, the Staff defined two expectations regarding pressurizer level: (1) the pressurizer does not go water solid, and, (2) the level in the pressurizer remains well enough below the PSV inlet nozzles to prevent moisture carryover through the PSVs. As a result of these defined expectations, the revised analysis verifies that the pressurizer level remains sufficiently below the PSV inlet nozzles to prevent moisture carryover through the PSVs when the PSVs are open, and the pressurizer does not go water solid when the PSVs are closed during the transient.

  NUREG-0857, Supplement 8 also noted that the bottoms of the PSV inlet nozzles are located at 99.4% of the pressurizer level instrumentation calibrated span. It should be noted, however, that the CENTS code models the pressurizer and surge line as a segmented node, comprised of three cylindrical sections. Therefore, when water rises into the bottom of the pressurizer's hemispherical upper head during a transient simulation, minor adjustments must be made to translate the predicted water level (in the model's cylindrical upper head) with the expected water level in the physical plant (in the pressurizer's hemispherical upper head). This is accomplished by equating water volumes between the model and the actual plant. For the FWLB with LOP and Single Failure Long-

Term Cooling reanalysis, it was determined that a pressurizer water level of 99.4% would equate to a pressurizer water volume of 1738 cubic feet, based on the physical configuration of the PVNGS pressurizer. The revised analysis uses this volume as the acceptance criterion at the times when PSVs are open. The acceptance criterion at the times when PSVs are closed remains as the pressurizer not going water solid, which corresponds to 1813 cubic feet based on the physical configuration of the PVNGS pressurizer.

The revised FWLB with LOP and Single Failure Long-Term Cooling Event analysis demonstrated that the pressurizer did not go solid throughout the transient and that RCS pressure control was maintained. Therefore, the adequacy of AFW capacity for long-term heat removal has been demonstrated. Additionally, the pressurizer water volume remained well enough below the PSV inlet nozzles at the times that the PSVs were predicted to open to prevent moisture carryover through the PSVs. Therefore, the PSVs will only pass saturated steam when they open, which is consistent with the PSV qualification tests.

- <u>PSV Flow Area:</u> The PSV correction factor error described in Reference 2 has been corrected in the revised analysis.

- <u>Time Delay between Turbine Trip and LOP:</u> UFSAR Section 15.0.4.2 states that a LOP may occur following a turbine trip due to grid destabilization following the loss of the generating unit. However, based on plant characteristics and grid stability studies, the minimum time that a LOP may occur following a turbine trip has been conservatively calculated to be 3 seconds. The AOR described in Attachment 6 of Reference 1 included discretionary conservatism in that it did not account for this time delay. The revised analysis, however, removed the discretionary conservatism and includes the 3-second delay to more accurately reflect plant response.

  The 3-second delay has been utilized in other PVNGS licensing basis safety analyses, including analyses of Steam Generator Tube Rupture (SGTR), Reactor Coolant Pump (RCP) Sheared Shaft, and RCP Rotor Seizure Events. In addition, the NRC staff has reviewed and approved the use of the 3-second delay in various safety analyses, as documented in CESSAR safety evaluations (NUREG-0852, "Combustion Engineering Standard Safety Analysis Report (CESSAR) Safety Evaluation Report," Supplement 1, dated March 1983, and Supplement 2, dated September 1983), and in NRC letter to 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)", dated May 23,1996.

- <u>PSV Blowdown:</u> The FWLB with LOP Long-Term Cooling Event analysis described in Attachment 6 of Reference 1 used a conservative PSV blowdown of 20%, which bounds the Electric Power Research Institute (EPRI) PSV qualification test results for a steam-water transition test. The reanalysis, however, utilizes a PSV blowdown of 14.2%, which is the maximum blowdown observed in the EPRI tests for saturated steam only conditions. This value is justified on the basis that the revised analysis predicts only saturated steam passing through the PSVs, and because the predicted peak pressure and rate of pressure increase are bounded by the EPRI test conditions.

The EPRI tests are documented in topical report CEN-227, "Summary Report on the Operability of Pressurizer Safety Valves in C-E Designed Plants," dated December 1982, which was previously submitted for NRC review in accordance with Item II.D.1 of NUREG-0737, "The Three Mile Island (TMI) Action Plan". By letter dated April 25, 1988, the NRC staff notified APS that the staff's contractor, Idaho National Engineering Laboratory (INEL), had reviewed the test report, and that the NRC staff endorsed the contractor's findings. The staff also concluded that APS had provided an acceptable response to this TMI Action Plan item, and reconfirmed that the PSVs met the requirements of General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary", GDC 15, "Reactor Coolant System Design", and GDC 30, "Quality of Reactor Coolant Pressure Boundary". In the INEL's Technical Evaluation Report for CEN-227, the contractor noted that ". . . the Dresser 31709NA safety valves at PVNGS Units 1, 2, and 3 are required to operate with steam inlet conditions only. The EPRI test program tested the Dresser 31709NA valve for the required range of conditions. The four applicable steam tests all showed stable performance of the safety valve with it opening at <1.5% over the set pressure and closing with 9.0 to 14.2% blowdown. . . .". Therefore, based on empirical test results, the maximum observed blowdown value is selected for the revised analysis.

- Initial Steam Generator Water Level: Revision 3 of NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition", states that the most adverse conditions within a permitted operating band should be used as initial conditions for safety analyses, where a permitted operating band is defined as the permitted fluctuations in a given parameter and associated uncertainties. The Technical Specifications, however, do not specify the permitted fluctuations for steam generator water level in Mode 1, Power Operations. Therefore, consistent with the guidance provided in ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", the initial conditions at the time of postulated fault initiation are derived from the full range of expected normal operating conditions as specified in plant procedures. Derivation of initial conditions takes into consideration systems under manual control, alarm setpoints, required manual actions, and protective overrides. On this basis, the revised analysis utilized the lower operational limit of 30% narrow range steam generator level, which is the lowest indicated level allowed by plant operating procedures when steam generator blowdown is in service. For analytical purposes, this value was reduced by 5% to account for instrument uncertainty. Based on this change, the initial steam generator water level used in the reanalysis is about 3 feet higher than that which was used in the analysis described in Attachment 6 of Reference 1.

- Perfect Mixing During Natural Circulation Conditions: Section 7.2.5 of the NRC-approved CENTS code technical manual, CE-NPD-282-P-A, "Technical Manual for the CENTS Code", dated March 17, 1994, describes how CENTS calculates the core-exit/hot-legs enthalpy tilt that results from asymmetric operation of the steam generators. The technical manual also states that two sets of mixing factors in the reactor vessel mixing model are used, one for low flow conditions and one for high flow conditions. Response to NRC Request for Additional Information No. 13, associated with CE-NPD 282-P-A, states that these factors are derived from experimental data.

  The FWLB with LOP Long-Term Cooling Event analysis described in Attachment 6 of Reference 1 conservatively utilized mixing factors which are representative of forced flow conditions. However, for this event, the RCPs would coast down to a stop following the LOP, resulting in natural circulation flow conditions. For natural circulation conditions, experimental data shows that perfect mixing would occur in the reactor vessel. That is, during natural circulation the coolant that exits the vessel and passes into the hot legs would have essentially the same temperature in both RCS loops, even if the coolant temperatures between the cold legs differed. The NRC staff has independently reviewed the validity of perfect mixing under natural circulation conditions, as evidenced by the staff's safety evaluation dated February 11, 1999, associated with Westinghouse topical report WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses".

  The revised analysis therefore utilizes mixing factors that are representative of perfect mixing when natural circulation conditions are established in the reactor vessel following the LOP.

- Auxiliary Feedwater Flow: The FWLB with LOP Long-Term Cooling Event analysis described in Attachment 6 of Reference 1 conservatively used a constant minimum AFW flow for the duration of the event. Previous FWLB AORs, however, used a bounding AFW flow curve as a function of steam generator pressure. The revised analysis utilizes this pressure-dependent AFW flow curve, consistent with previous practice.

Note that as a result of the changes in initial conditions and assumptions, the limiting break size was re-determined by using the existing method, i.e., the parametric studies.

4.0 Conclusion

The postulated Feedwater Line Break (FWLB) with Loss of Offsite Power (LOP) and Single Failure Long-Term Cooling Event has been reanalyzed to address questions that were raised during conversion of the analyses to the CENTS code, The revised analysis includes several changes in input parameters and elements of the methodology, and verifies that acceptance criteria related to long-term cooling and PSV qualification are satisfied for operation at 3990 MWt, as well as at the current licensed power level of 3876 MWt.

One analytical change involves an element of methodology that constitutes a "departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses" as defined in 10 CFR 50.59(a)(2). This change, which requires NRC staff review and approval, is as follows:

- For the purpose of establishing the initial Reactor Coolant System (RCS) cold leg temperature for the analysis, it is assumed that the plant is operated on program T_{avg}, and the Pressurizer Level Control System (PLCS) is in the automatic mode of operation at the beginning of the event.

As discussed above, all other changes to input parameters, removal of discretionary conservatism from input parameters, and new or revised elements of methodology do not require NRC approval. The description of these changes is provided for completeness and is for information only. These changes, with the exception of PSV flow area, which was addressed in Reference 2, do not adversely affect the short-term RCS peak pressure. The fuel integrity, and radiological dose assessment analyses for FWLB are also not impacted by these changes. The revised analysis retains the original conservative licensing basis methodology as described in Section 2.0.

Enclosure 1 provides replacement pages for Attachment 6 of Reference 1.

Enclosure 1

Power Uprate Licensing Report
Replacement Pages

Power Uprate Licensing Report
Replacement Pages

Remove Pages:	**Insert Pages:**
Page xx	Page xx
Page xxviii	Page xxviii
Page 6-40	Page 6-40
--------------	Page 6-40a
Page 6-185	Page 6-185
Page 6-186	Page 6-186
Page 6-187	Page 6-187
Page 6-188	Page 6-188
Page 6-190	Page 6-190
Page 6-191	Page 6-191
Page 6-193	Page 6-193
Page 6-212	Page 6-212
Page 6-213	Page 6-213
Page 6-214	Page 6-214
Page 6-215	Page 6-215
Page 6-216	Page 6-216
Page 6-217	Page 6-217
Page 6-218	Page 6-218
Page 6-219	Page 6-219
Page 6-220	Page 6-220
Page 6-221	Page 6-221
Page 6-222	Page 6-222
Page 6-223	Page 6-223
Page 6-224	Page 6-224

Figure 6.3-97 FWLB with LOP Primary Peak Pressure Case - RCS Pressure vs. Time....6-199

Figure 6.3-98 FWLB with LOP Primary Peak Pressure Case - Pressurizer Pressure vs. Time....6-200

Figure 6.3-99 FWLB with LOP Primary Peak Pressure Case - Pressurizer Water Volume vs. Time....6-201

Figure 6.3-100 FWLB with LOP Primary Peak Pressure Case - SG Pressures vs. Time....6-202

Figure 6.3-101 FWLB with LOP Primary Peak Pressure Case - SG Levels vs. Time....6-203

Figure 6.3-102 FWLB with LOP Primary Peak Pressure Case - SG Liquid Inventories vs. Time....6-204

Figure 6.3-103 FWLB with LOP Primary Peak Pressure Case - RCS Loop Flows vs. Time....6-205

Figure 6.3-104 FWLB with LOP Primary Peak Pressure Case - SG Steam Flows vs. Time....6-206

Figure 6.3-105 FWLB with LOP Primary Peak Pressure Case - Break Flow vs. Time....6-207

Figure 6.3-106 FWLB with LOP Primary Peak Pressure Case - Break Enthalpy vs. Time....6-208

Figure 6.3-107 FWLB with LOP Primary Peak Pressure Case - PSV Flow vs. Time....6-209

Figure 6.3-108 FWLB with LOP Primary Peak Pressure Case - Surge Flow vs. Time....6-210

Figure 6.3-109 FWLB with LOP Primary Peak Pressure Case - Minimum DNBR vs. Time....6-211

Figure 6.3-110 FWLB with LOP Long-Term Cooling Case - Core Power vs. Time....6-212

Figure 6.3-111 FWLB with LOP Long-Term Cooling Case – Unaffected Loop RCS Temperatures vs. Time....6-213

Figure 6.3-112 FWLB with LOP Long-Term Cooling Case - RCS Pressure vs. Time....6-214

- Post-Trip Main Steam Line Break (MSLB) employs a more detailed reactivity calculation including moderator density feedback in the hot channel as described in Section 6.3.1.5.5.1.
- Single RCP Sheared Shaft with LOP assumes the operators refill the affected SG as described in Section 6.3.3.4.1.
- Dose calculations assume a decontamination factor (DF) of 100 (partition factor of 0.01) for the unaffected SG as described in Section 6.4.1.1.1.
- The FWLB -Long Term Cooling event assumes that the plant is operated on program Tavg, and the Pressurizer Level Control System (PLCS) is in the automatic mode at the beginning of the event.

The results of the NSSS analyses and evaluations demonstrate that PVNGS Unit 2 can operate acceptably at the increased rated thermal power and that applicable licensing criteria and requirements are satisfied.

The effect of this license amendment request represents minimal safety significance and minimal impact on the health and safety of the general public.

2. decay heat of eleven-fission product groups, as defined by the user.

The minimum Departure from Nucleate Boiling Ratio (DNBR) values and Departure from Nucleate Boiling (DNB) thermal margin requirements were determined using the CETOP-D code (Reference 6-18). The minimum DNBR values for events that include a loss of RCS flow were determined using a more detailed open channel thermal hydraulics code, TORC (Reference 6-19).

The STRIKIN-II code (Reference 6-5) was employed to simulate fuel and cladding integrity for CEA ejection events.

The HERMITE code (Reference 6-20) was employed to simulate the core response to loss of RCS flow and single RCP sheared shaft events. The RCS flow coastdown experienced following a LOP, single RCP sheared shaft, and single RCP seized rotor events was analyzed using the COAST code (Reference 6-21).

The following methods/assumption changes have been applied to Non-LOCA transient analysis as discussed in Section 6.3 and Section 6.4:

- More realistic Inadvertent Opening of an Atmospheric Dump Valve (ADV) (IOSGADV) with a Loss of Power (LOP) event analyzed separately from Limiting Anticipated Operational Occurrence (AOO) with single failure (i.e., Loss of Flow (LOF) from Specified Acceptable Fuel Design Limit (SAFDL) as described in Section 6.3.1.4.1.
- Post-Trip Main Steam Line Break (MSLB) employs a more detailed reactivity calculation including moderator density feedback in the hot channel as described in Section 6.3.1.5.3.1.
- Single RCP Sheared Shaft with LOP assumes the operators refill the affected SG as described in Section 6.3.3.4.1.
- Dose calculations assume a decontamination factor (DF) of 100 (partition factor of 0.01) for the unaffected SG as described in Section 6.4.0.
- The FWLB -Long Term Cooling event assumes that the plant is operated on progran Tavg, and the Pressurizer Level Control System (PLCS) is in the automatic mode of operation at the beginning of the event.

<u>Section 6.3.0.2</u> <u>Initial Conditions</u>

The range of initial conditions evaluated in the non-LOCA transient analyses is listed in Table 6.3-2. The analytical range includes instrument uncertainties that were applied to extend the operating limits.

In accordance with the SRP (Reference 6-6), the transient analyses employ the most limiting combination of core characteristics (i.e., Doppler, MTC, power distribution, etc.). In some instances, this has been achieved by combining the most adverse value of each parameter, regardless of burnup. Other analyses used burnup consistent sets of physics parameters, with the most adverse time in cycle combination being reported. A set of bounding core physics parameters was utilized in the transient analyses. These physics parameters are verified for future core loading patterns following the reload design process in accordance with PVNGS procedures. Refer to the specific event section for a more detailed list of the core physics parameters for any given transient.

Depending on the enthalpy of the reverse flow released through the break and the affected SG's heat transfer characteristics, the reverse flow may induce either an RCS heatup or cooldown. However, excessive heat removal through the break is not considered in this analysis, because the cooldown potential is less than that of MSLB events. Therefore, the FWLB is analyzed as a heatup event.

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

Section 6.3.2.8.1.1 Identification of Event and Causes

As described in UFSAR Section 15.2.8, FWLB event is initiated by a break downstream of the check valves. Assuming inoperability of the FW system and low enthalpy liquid discharge through the break, the event can be described as follows:

The termination of the main FW to both SGs and discharge of exiting SG liquid inventory through the break causes increasing SG temperatures and decreasing levels. This leads to decreasing heat removal by the secondary system. The result is a heatup and pressurization of the RCS. The heatup and pressurization becomes more severe as the affected SG experiences a further reduction in its heat transfer capability due to insufficient liquid inventory. This initial sequence of events culminates with a reactor trip on high pressurizer pressure and opening of the PSVs. In an actual transient, a low SG level trip or a high containment pressure trip may occur much earlier than the HPPT making the consequences less adverse. RCS heatup may continue after the trip due to a total loss of heat transfer in the affected SG and reduced heat transfer in the unaffected SG.

A LOP causes a loss of forced RCS flow, turbine load, pressurizer pressure and level control, and SBCS, making the consequences of this event more severe. Consideration of LOP and single failures are addressed in Section 6.3.2.8.1.3.1.

During the transient, opening of the MSSVs after turbine trip on reactor trip provides additional cooling by the secondary system, and eventually, decreasing core power reduces the heat load to the SGs. An AFAS is actuated by low SG level in the affected SG, and AFW that is supplied to both SGs results in increasing cooldown of the RCS. Reduction in secondary system pressure or increase in containment pressure cause MSIS to isolate the affected SG. Following the MSIV closure, the pressure difference between the SGs increases and eventually the AFW is fully diverted to the unaffected SG due to AFW lockout, restoring the unaffected SG liquid level and long-term cooling of RCS.

An operator may cool the NSSS by using manual operation of the AFW system and the ADVs anytime after the trip occurs. However, no credit is taken for the operator action for the first 30 minutes.

Section 6.3.2.8.1.2 Acceptance Criteria

FWLB with LOP is the most severe "limiting fault" event that results in an unplanned decrease in secondary system heat removal. Due to the low probability of occurrence, this event is subject to ASME Boiler and Pressure Vessel Code Service C limits for pressurization of primary and secondary systems. As defined in the ASME Code and SRP Section 15.2.8, the specific acceptance criteria are:

a. Pressure in the RCS and main steam system should be maintained below 120% of the design.
b. The potential for core damage should be evaluated on the basis that it is acceptable if the minimum DNBR remains above SAFDL. If the DNBR falls below SAFDL value, fuel damage should be assumed unless it can be shown, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically unaffected with no loss of core cooling capability.
c. Any activity release must be such that the calculated doses at the site boundary are well within the guidelines of 10 CFR Part 100.

In addition, AFW system should be available and capable to supply adequate water flow to the unaffected SG during the accident and subsequent shutdown.

Section 6.3.2.8.1.3 Description of Analysis

The NSSS response to a FWLB with LOP is simulated using the CENTS code. Several assumptions, which conservatively model the break discharge flow and enthalpy, and the affected SG level and heat transfer characteristics are made.

Initial and transient DNBR is calculated using the CETOP-D code which uses the CE-1 CHF correlation.

Two cases are analyzed for a FWLB event with LOP:

1. the maximum RCS pressure case and
2. long-term cooling case for AFW capacity.

In the first case, the input parameters and initial conditions are selected to maximize the RCS pressure, and demonstrate that the peak RCS pressure remains within 120% of the design pressure. Inputs to the second case were selected to maximize the pressurizer volume to demonstrate that the cooling by the AFW system is provided so that RCS heatup and pressurization is controlled without the pressurizer being filled beyond the PSV nozzle elevation at the times when the PSVs open, and without the pressurizer going water solid at all times.

Section 6.3.2.8.1.3.1 Transient Simulation

The system is initialized at 102% power using the most limiting initial parameters. At time equal zero, the limiting size break is simulated to occur downstream of the check valves. Blowdown of the SG nearest the FWLB, is modeled assuming frictionless critical flow as calculated by the Henry-Fauske correlation. Although the enthalpy of the blowdown physically depends on the location of the break, it is conservatively assumed that saturated liquid is discharged until no liquid remains, at which time saturated steam discharge is assumed.

A LOFW is simulated by a rapid ramp down of main FW flow to zero (in 0.1 seconds). The total loss of feed flow and discharge from the break yields a reduction of the SG water inventory, pressurization of the secondary side, and a resulting heatup and pressurization of the primary side. No credit is taken for a low water level trip condition in the affected SG until the SG is depleted of liquid. This conservatively delays the reactor trip prolonging the RCS heatup and overpressurization.

Further reduction in the SG inventory decreases the primary-to-secondary heat transfer due to heat transfer degradation in the affected SG.

Reactor trip occurs on HPPT. Since a LSGLT is assumed to occur when the affected SG is depleted, the most adverse condition is when HPPT occurs at the same time as low-level trip. Therefore, the initial conditions were selected to result in coinciding HPPT with affected SG dryout. AFAS is generated on low SG level in the affected SG. For conservatism, it is delayed until the affected SG is depleted of liquid.

A turbine trip occurs on reactor trip. A three second time delay between the turbine trip and the LOP is assumed as described in UFSAR Section 15.0.2.4. Following the combination of HPPT, SG dryout, and turbine trip, lifting of the PSVs and MSSVs provides decay heat removal.

The peak pressure transient is continued until the primary and secondary pressures and temperatures are decreasing and stabilizing.

The long-term cooling case continued for the first 30 minutes. In the long-term simulation, AFW is initiated to SGs after a conservative delay time that accounts for the start of AFW pumps and delivery of the flow. A reduced AFW flow is assumed to evaluate a single failure of one AFW pump. AFW flow is supplied to both SGs.

Eventually, cooldown of secondary system by the AFW, opening of MSSVs, and flow through the break results in decreasing secondary system pressure. When the secondary system reaches the main steam isolation pressure, a MSIS is generated and MSIVs close, isolating the affected SG. Depending on the break size and location, an earlier MSIS may be generated on high containment pressure resulting in consequences that are more adverse. A MSIS occurring at the time of trip is found to be the most limiting for isolation of the affected SG for the limiting break size. Upon isolation of the affected SG, the pressure difference between the SGs increase, and when the difference reaches to the AFW lockout setpoint, the total available AFW flow is diverted to the unaffected SG.

Cycling of PSVs and MSSVs, with the AFW flow provides adequate energy removal from RCS and secondary systems. When the cooling capability balances and exceeds the decay heat addition, the RCS pressure and pressurizer level begin to decrease. After 30 minutes, the operator may take actions to resume plant cooldown by opening the ADVs.

An active single failure was also considered in the analysis. Considering the peak pressure criteria, the only mechanisms for mitigation of the RCS and main steam system overpressurization are the PSVs, RCS flow, and MSSVs. There are no credible failures that can degrade the PSVs or MSSVs. A decrease in RCS-to-SG heat transfer due to RCS flow coastdown is caused by a LOP. If the LOP occurs prior to the HPPT, the RCP coastdown results in an almost immediate reactor trip, generated by the CPC on RCP speed, making the event consequences less severe. A LOP resulting from turbine trip has an effect that is more adverse. Following the turbine trip and the LOP, there is no credible single failure to make the FWLB with LOP event peak pressure consequences more adverse.

For the long-term cooling, the mechanisms to mitigate the primary and secondary heatup and pressurization are the PSVs, MSSVs, RCS flow, and the AFW capacity. Again, there is no credible single failure that can degrade the PSV and MSSV capacity, and the degradation of the RCS flow is the same as the peak pressure consideration. For the long-term cooling for FWLB, the only single failure that can degrade the AFW capacity is the failure of one of the AFW pumps to start that will result in reduced heat removal capacity by the AFW. Therefore, FWLB event with LOP for long-term cooling is analyzed with failure of one AFW pump as an active single failure.

Section 6.3.2.8.1.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-27 and Table 6.3-28 contain the initial conditions used for the peak primary pressure and long-term cooling events, respectively. In addition, the most limiting break size, heat transfer degradation, and time of trip are determined by investigation of their effects on peak primary pressure.

The following assumptions are made in this analysis:

1. In accordance with Section 6.3.0.2, the initial conditions for the process variables were varied within the ranges of steady state operational configurations. These included the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. Conservative break flow and enthalpy were used, i.e., discharge of saturated liquid until SG is dry.
3. Heat transfer degradation in the affected SG is delayed until the liquid inventories are depleted and then an instantaneous loss of heat transfer is assumed.
4. The key parameters are initialized such that a reactor trip occurs from a high pressurizer pressure signal simultaneously with depletion of SG liquid mass.
5. The AFAS is delayed until liquid mass inventory in the affected SG is depleted.
6. Only the HPPT is enabled. Although a low SG level trip may occur earlier than the HPPT, no credit is taken for this trip until liquid mass in the affected SG is depleted.
7. There is no operator action for the first 30 minutes of the event.
8. MSIS and SIAS are initiated on high containment pressure at the time of reactor trip.

Table 6.3-28
Parameters Used for FWLB with LOP Long-Term Cooling Event

Parameter	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	552	555
Initial pressurizer pressure (psia)	2100	2100
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	23.9	23.9
Initial SG level (ft)	34.0	35.8
MTC ($\Delta\rho$/°F)	0.0E-04	0.0E-04
FTC	least negative	least negative
Kinetics	maximum β	maximum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.0	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	500	500
Plugged SG tubes (% of tubes/SG)	asymmetric 9 | 23	asymmetric 0 | 10
PSV tolerance	-1%	-1%
PSV blowdown	14.2%	14.2%
MSSV tolerance	+3%	+3%
MSSV blowdown	5%	5%
Single failure	one AFW pump	one AFW pump
LOP	yes	yes
FWLB area (ft^2)	0.24	0.23

Section 6.3.2.8.1.5 Results

Table 6.3-29 and Table 6.3-30 present a sequence of events which occur following the FWLB with LOP until operator action is initiated for the primary peak pressure and long-term cooling cases, respectively. FWLB with LOP analyses are performed separately for primary peak pressure and long-term cooling criteria since the selection of worst parameters for these events are not mutually conservative. These sequences of events are representative for 3990 MW_t, and 3876 MW_t units.

The behaviors of NSSS parameters following the FWLB with a LOP resulting from turbine trip and a single failure are presented in Figure 6.3-93 to Figure 6.3-122.

Examination of the sequence of events during the transient reveals a similar NSSS response between the existing plant configuration (3876 MW_t) and PUR (3990 MW_t). The 3990 MW_t plant configuration experiences slightly higher peak primary pressures due to the higher initial core power level and RCS temperature.

Primary Peak Pressure Case

Figure 6.3-93 through Figure 6.3-109 shows the response for the FWLB with LOP event under most adverse transient conditions that maximize the RCS peak pressure.

The sudden reduction of primary-to-secondary heat transfer caused by decrease in SG inventory and LOFW leads to RCS and secondary system temperature and pressure increase. The rapid heatup of the RCS results in a reactor trip on high pressurizer pressure coinciding with affected SG dryout. A turbine trip followed by LOP causes further increase in pressure and temperature in both primary and secondary systems. The PSVs and MSSVs open providing cooldown and maintaining primary pressure well below 120% of the design value. MSSVs provide adequate pressure relief so that the main steam system pressure is limited to opening pressures of the MSSVs. Thus, the secondary system pressure remains well below 120% of the design pressure.

No significant change occurs in the minimum DNBR value during the initial RCS heatup and pressurization. The DNBR value starts to decrease following the combined reactor trip and LOP, but quickly turns around and remains above the SAFDL.

Long-Term Cooling Case

Figure 6.3-110 through Figure 6.3-122 show the response for the FWLB with LOP and failure of one of AFW pumps to start event under most adverse initial and transient conditions that minimize the heat removal by the secondary system, and maximize the pressurizer level.

The sudden reduction of primary-to-secondary heat transfer caused by decreasing SG inventory and total LOFW leads to RCS and secondary system temperature and pressure increase. The rapid heatup of the RCS results in a reactor trip on high pressurizer pressure coinciding with affected SG dryout. An AFAS is generated at the time of affected SG dryout delivering one-pump AFW flow after time of delay to both SGs. Cooldown by AFW results in depressurization of the secondary system to the MSIS pressure, isolating the affected SG. Depending on the break size, an earlier MSIS may occur on high containment pressure. Following the MSIS, the pressure difference between the SGs increases, and eventually AFW lockout occurs, diverting full available AFW flow to unaffected SG. AFW addition and the cycling of PSVs and MSSVs provide adequate cooling to remove the decay heat until operator action is taken after the first 30 minutes.

Radiological consequences for this event are presented in Section 6.4.2.1.

Table 6.3-30
Sequence of Events for FWLB with LOP Long-Term Cooling 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)	0.24	0.23
26.09	29.38	AFAS generated in unaffected SG	10% WR	10% WR
26.77	29.94	Pressurizer pressure reaches trip setpoint (psia)	2450	2450
26.77	29.94	HPPT signal generated		
26.77	29.94	SIAS/CIAS/MSIS signal generated		
26.79	29.96	PSVs open (psia)	2450	2450
26.98	30.17	Dryout of affected SG, AFAS generated in affected SG	<5000	<5000
27.27	30.44	Reactor trip breakers open		
27.27	30.44	Turbine trip occurs		
27.47	30.69	Maximum RCS pressure (psia)	2551	2562
27.87	31.04	Scram CEAs begin falling		
30.27	33.44	LOP occurs		
34.64	35.55	MSSVs bank 1 open (psia) (1)	1303	1303
32.39	35.56	Main Steam Isolation Valves close	---	---
37.78	37.19	MSSVs bank 2 open (psia)	1344	1344
34.89	38.26	PSVs close (psia)	2102	2102
---	38.26	Pressurizer Volume (ft^3)	---	1644
36.18	41.51	AFW Lockout (psid)	270	270
37.78	41.89	Maximum SG Pressure (psia)	1357	1372
70.28	73.45	One charging pump restarts (gpm)	44	44
73.00	76.20	AFW initiated to SG #2 (one pump, gpm)	650	650
50.26	58.06	MSSVs bank 2 close (psia)	1277	1277
78.40	85.92	MSSVs bank 1 close (psia)	1238	1238
487.0	---	PSVs open (psia)	2450	---
488.6	---	PSVs close (psia)	2102	---
488.6	---	Pressurizer Volume (ft^3)	1695	---
1792	1800	Maximum liquid volume of pressurizer (ft^3)	1736	1701
1800	1800	Operators initiate plant cooldown (min)	30	30

Note: (1) MSSVs cycle between 500 sec and 1800 sec, approximately every 100 seconds.

Figure 6.3-110
FWLB with LOP Long-Term Cooling Case – Core Power vs. Time

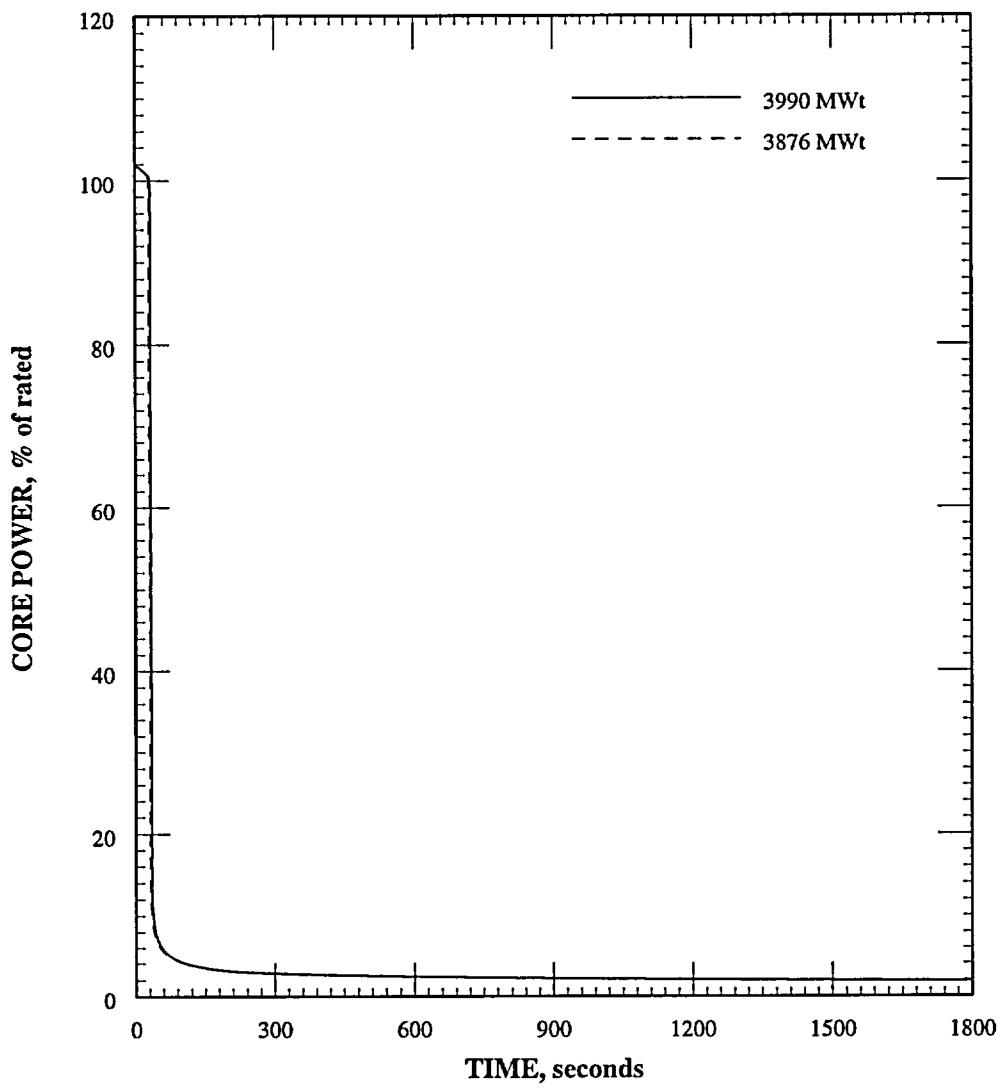

Figure 6.3-111
FWLB with LOP Long-Term Cooling – Unaffected Loop RCS Temperature vs. Time

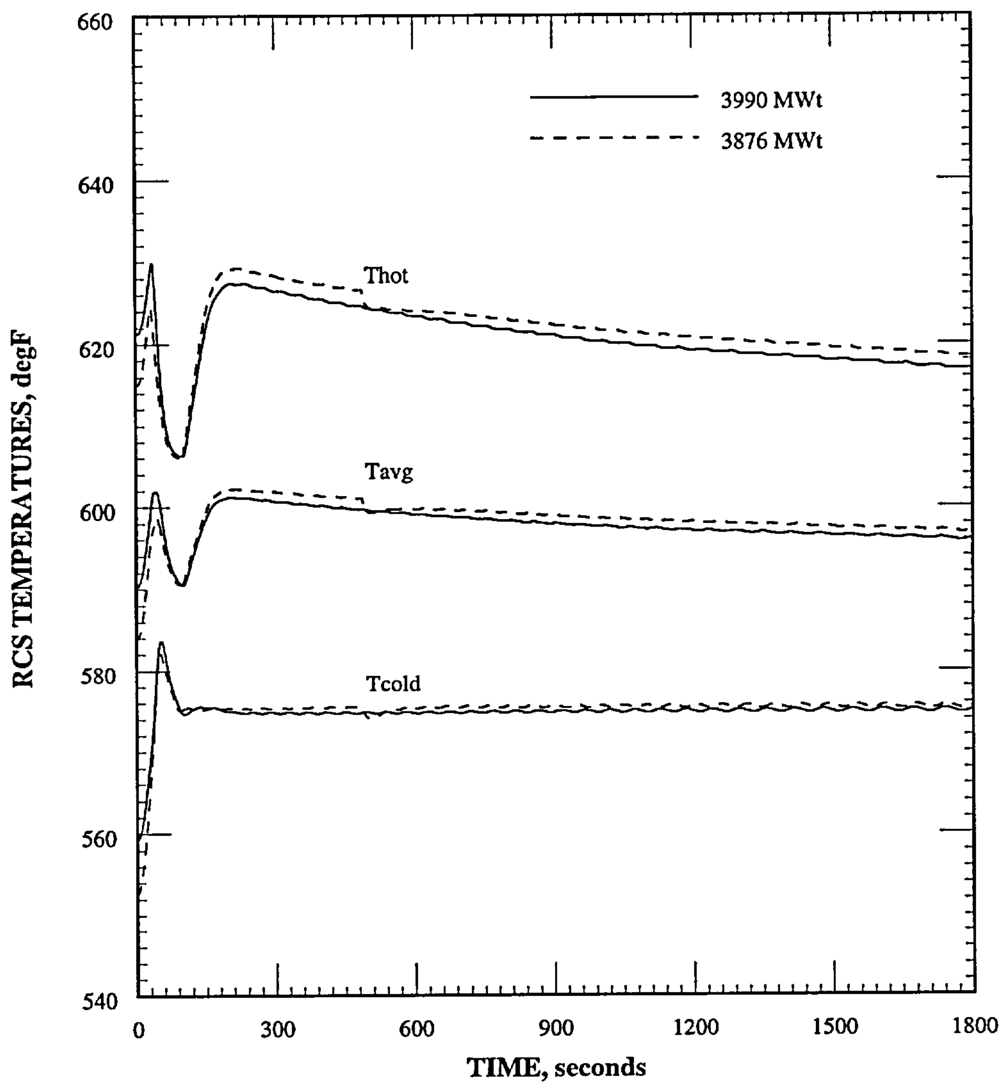

Figure 6.3-112
FWLB with LOP Long-Term Cooling Case – RCS Pressure vs. Time

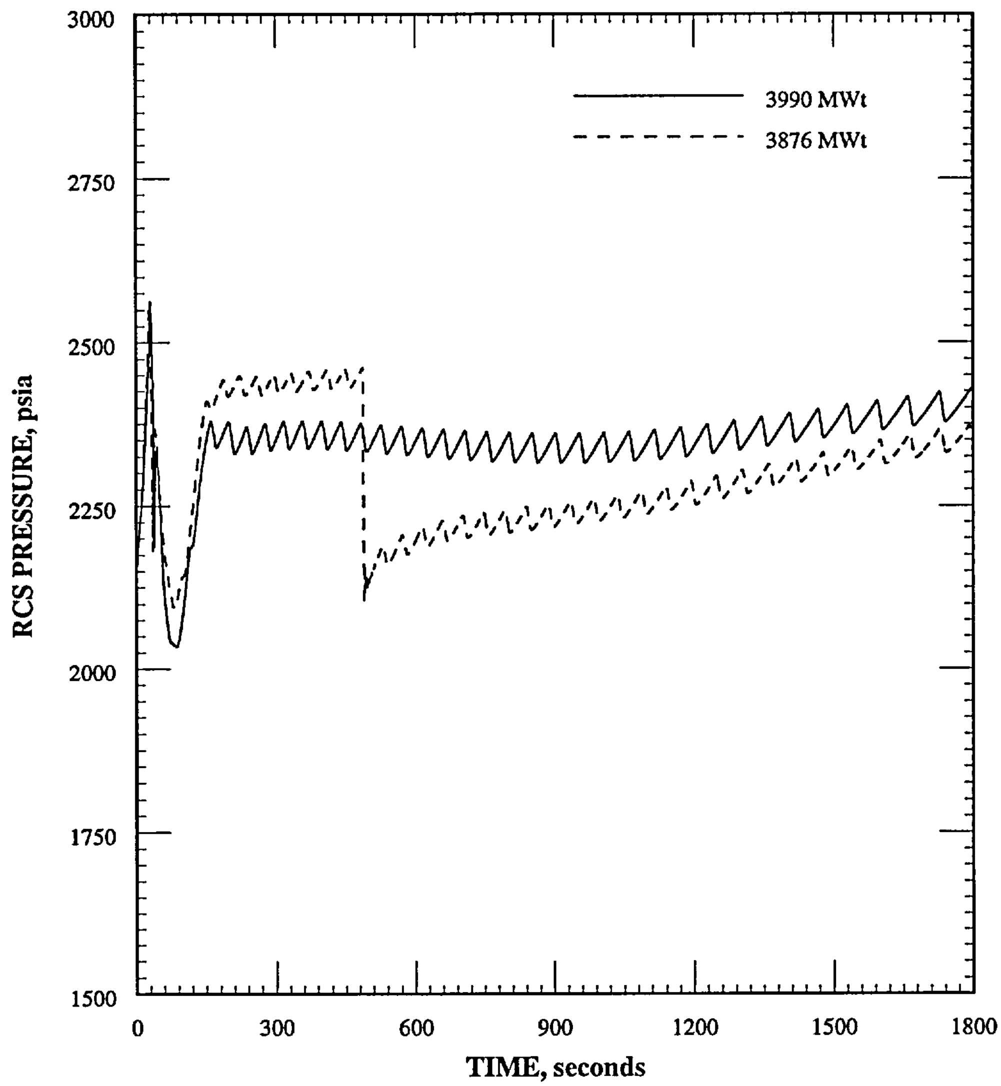

Figure 6.3-113
FWLB with LOP Long-Term Cooling – Pressurizer Pressure vs. Time

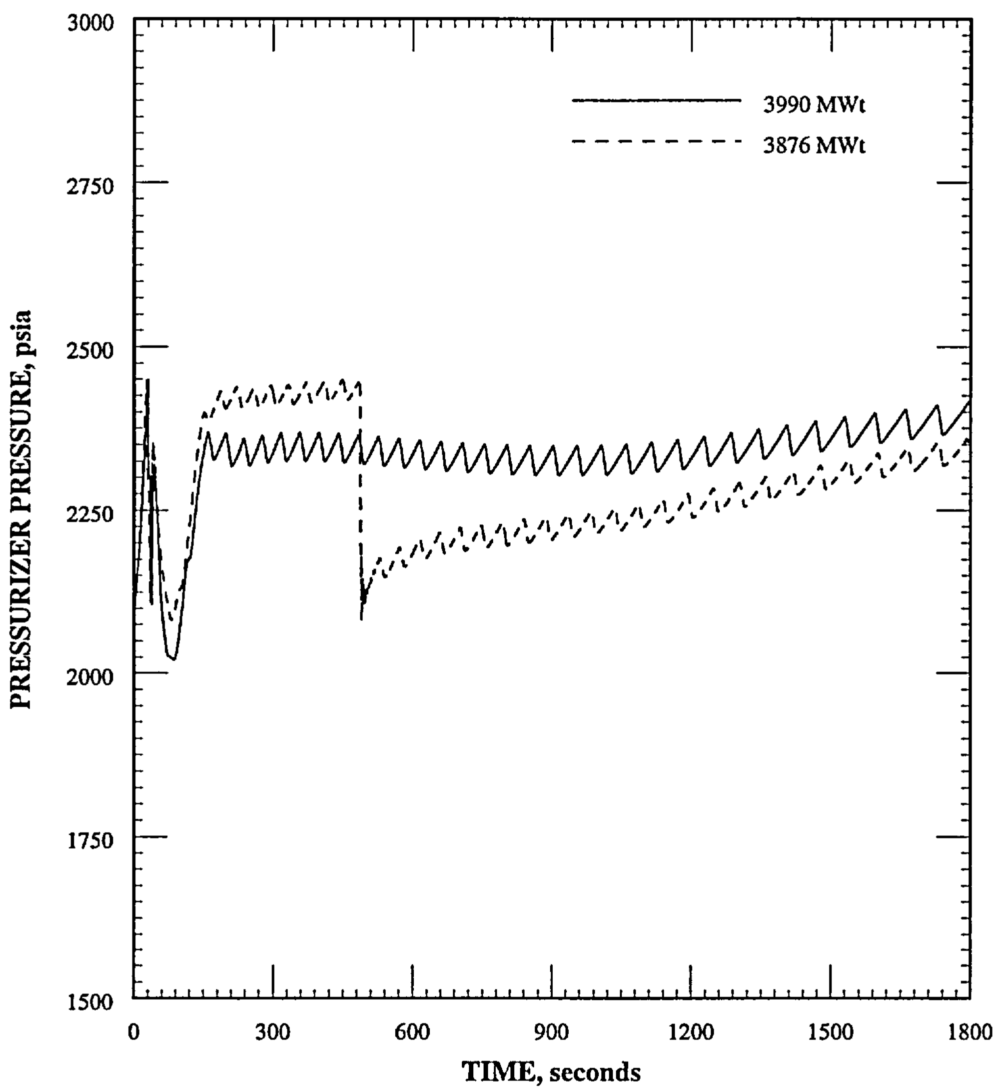

Figure 6.3-114
FWLB with LOP Long-Term Cooling – Pressurizer Water Volume vs. Time

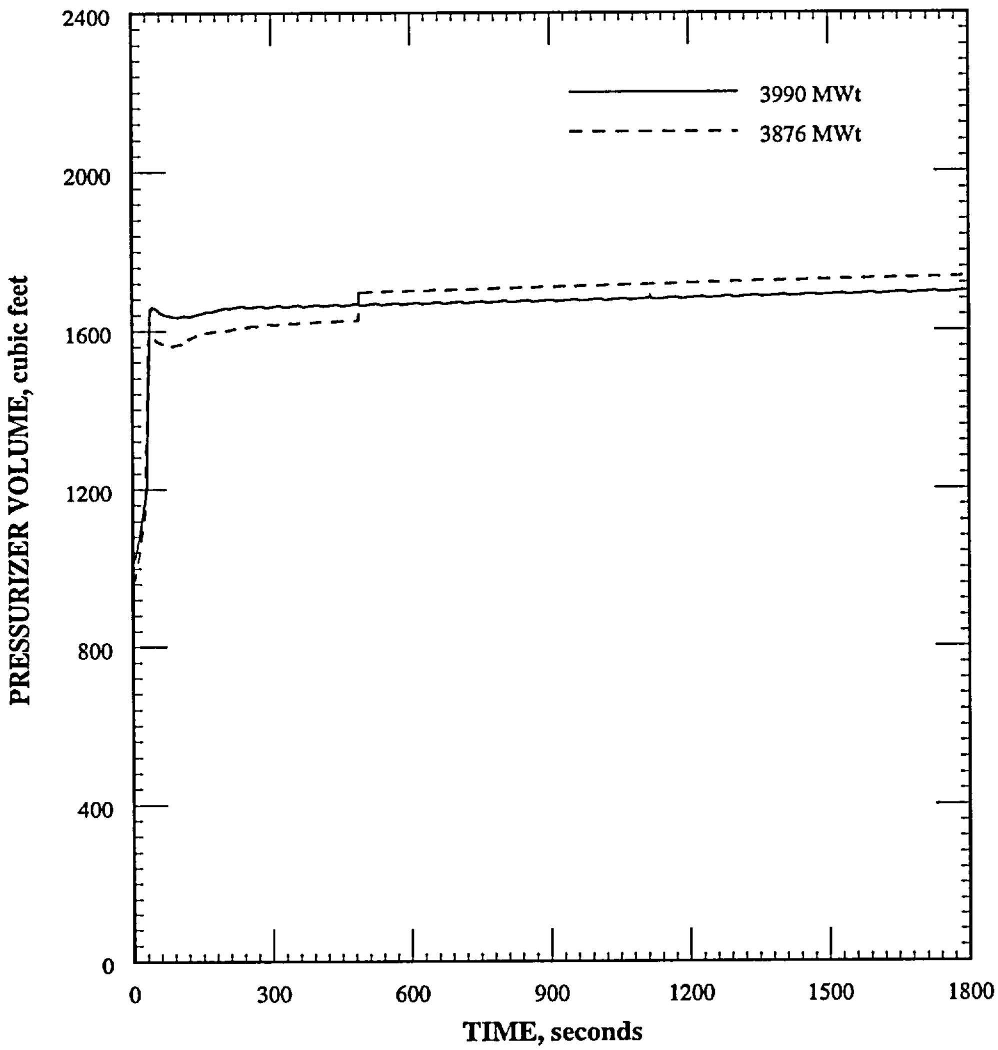

Figure 6.3-115
FWLB with LOP Long-Term Cooling – SG Pressure vs. Time

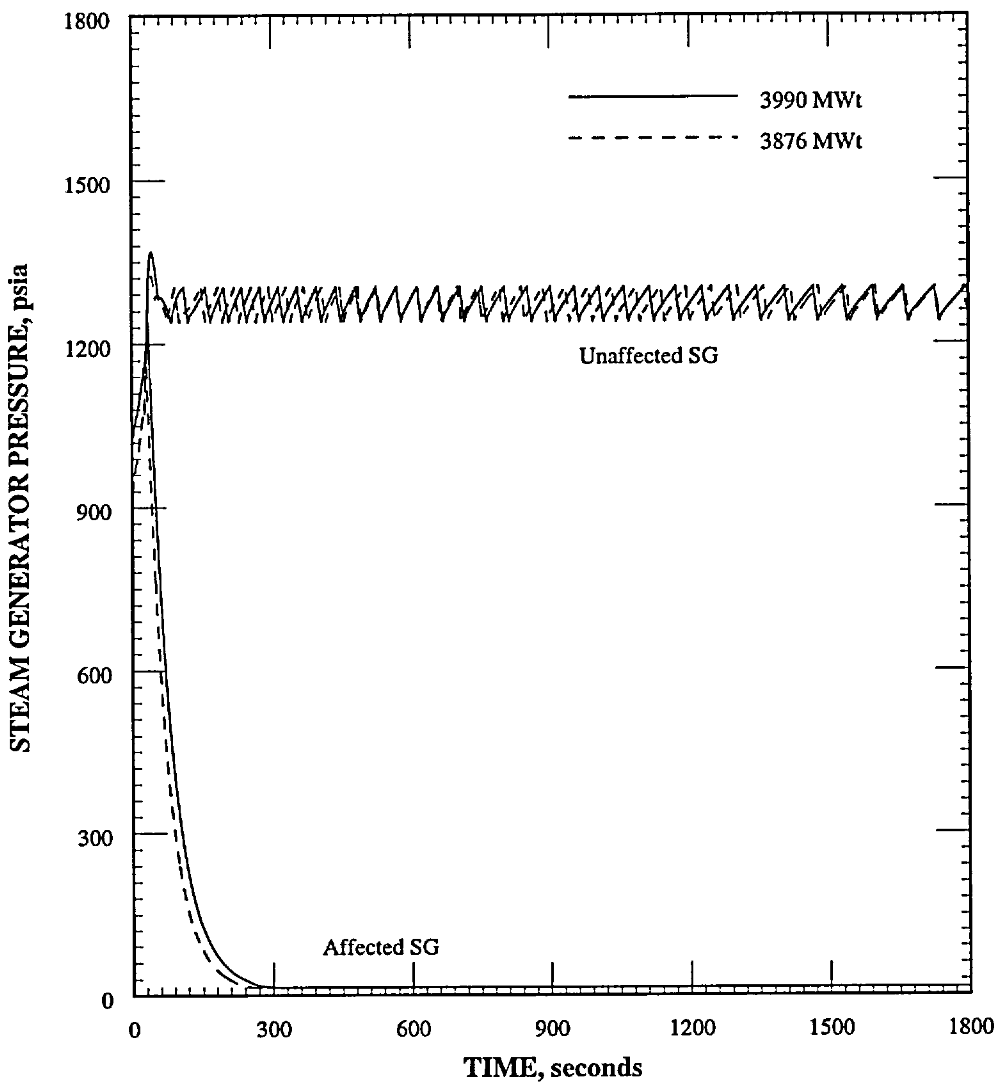

Figure 6.3-116
FWLB with LOP Long-Term Cooling – Unaffected SG Levels vs. Time

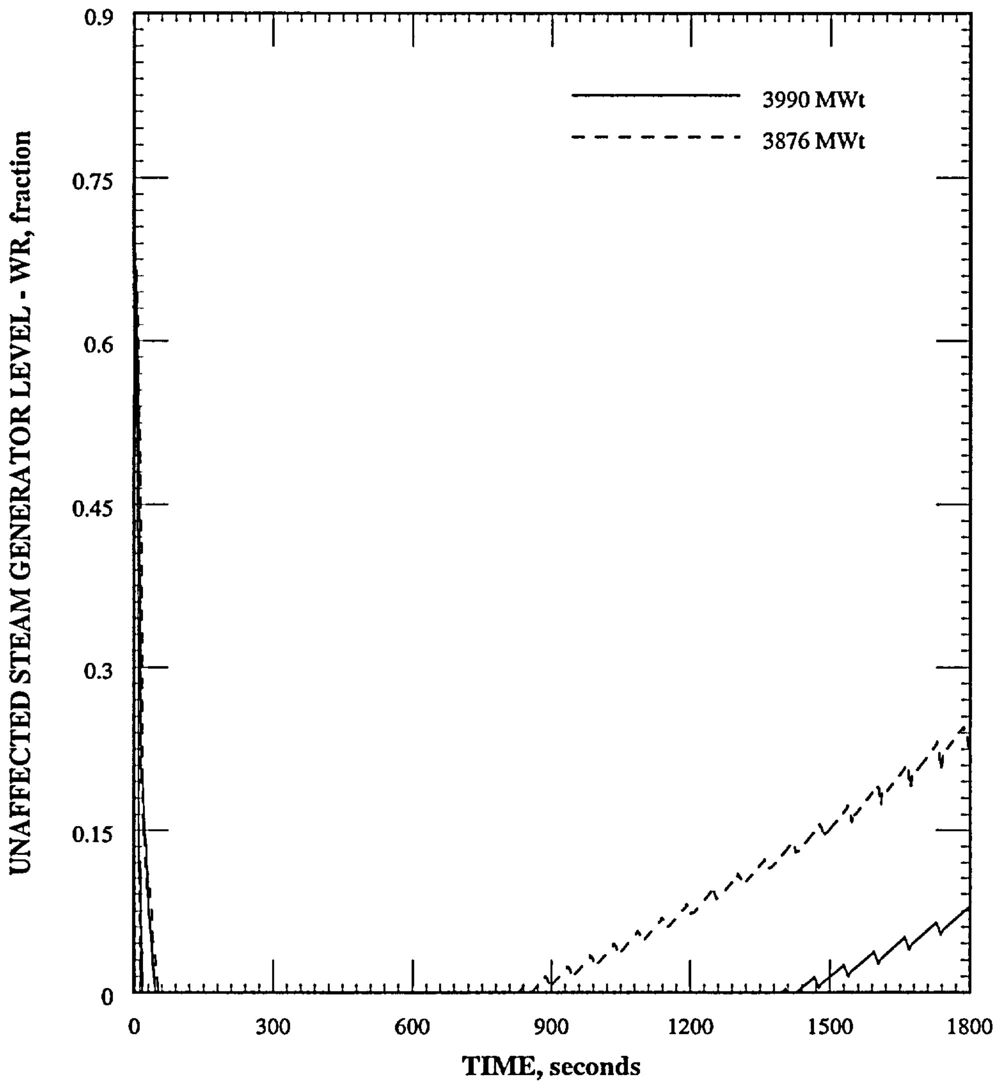

Figure 6.3-117
FWLB with LOP Long-Term Cooling – SG Liquid Inventories vs. Time

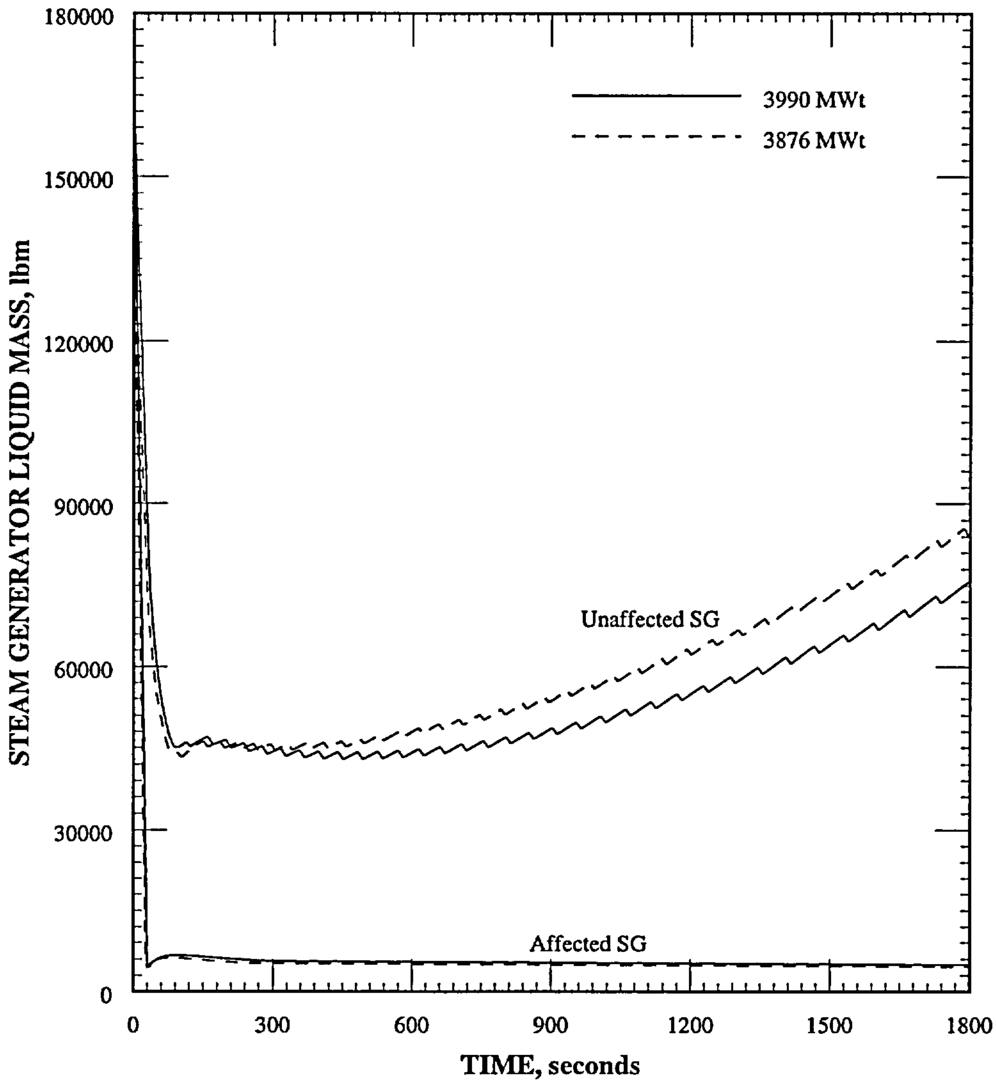

Figure 6.3-118
FWLB with LOP Long-Term Cooling – RCS Loop Flow vs. Time

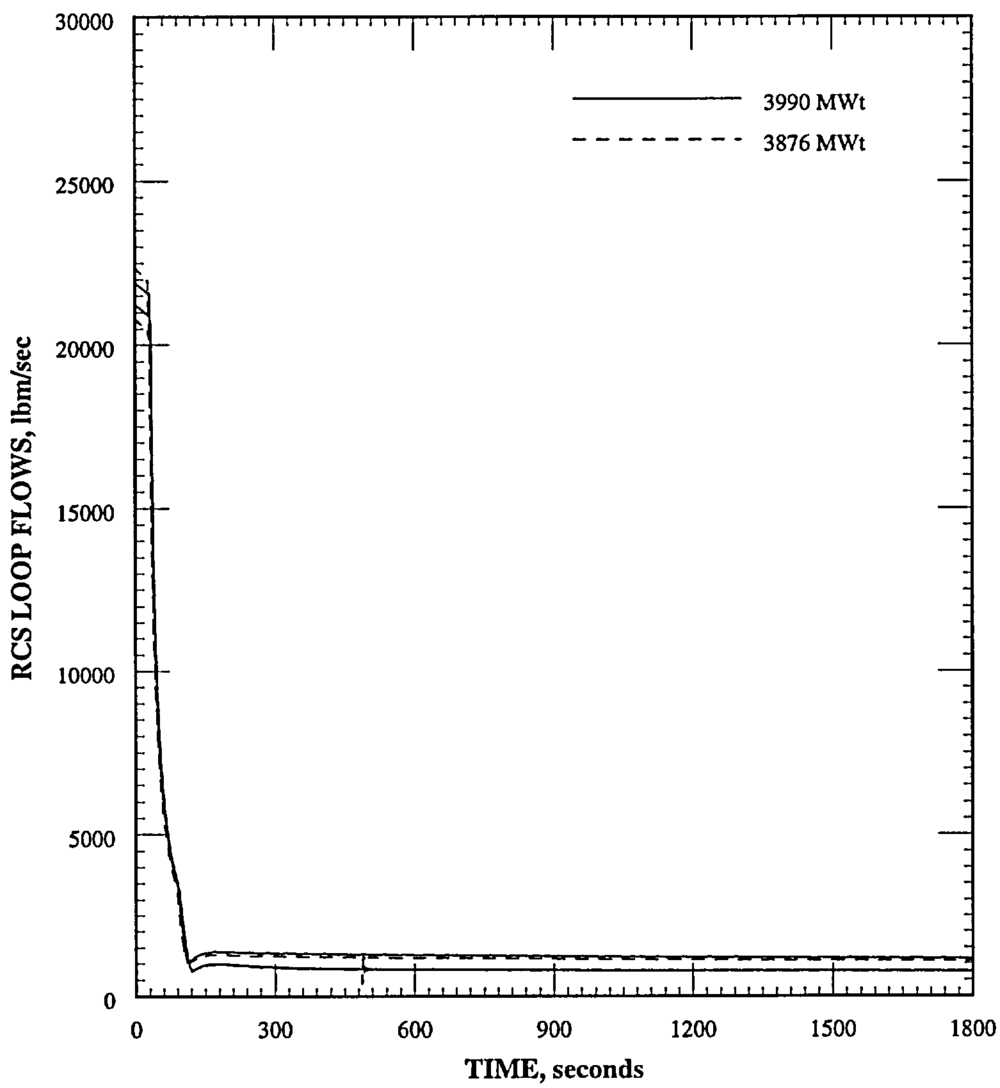

Figure 6.3-119
FWLB with LOP Long Term Cooling – Affected SG AFW Flow vs. Time

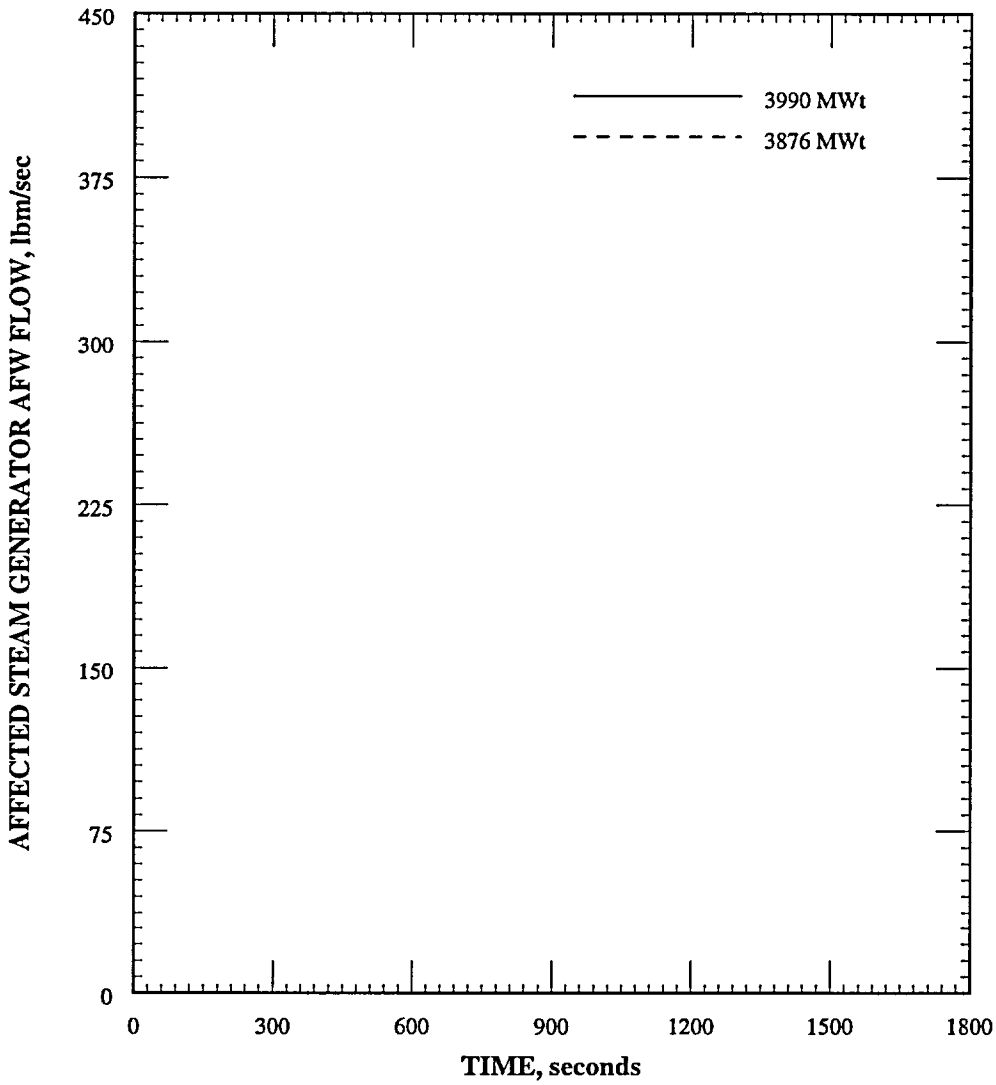

Figure 6.3-120
FWLB with LOP Long-Term Cooling – Unaffected SG AFW Flow vs. Time

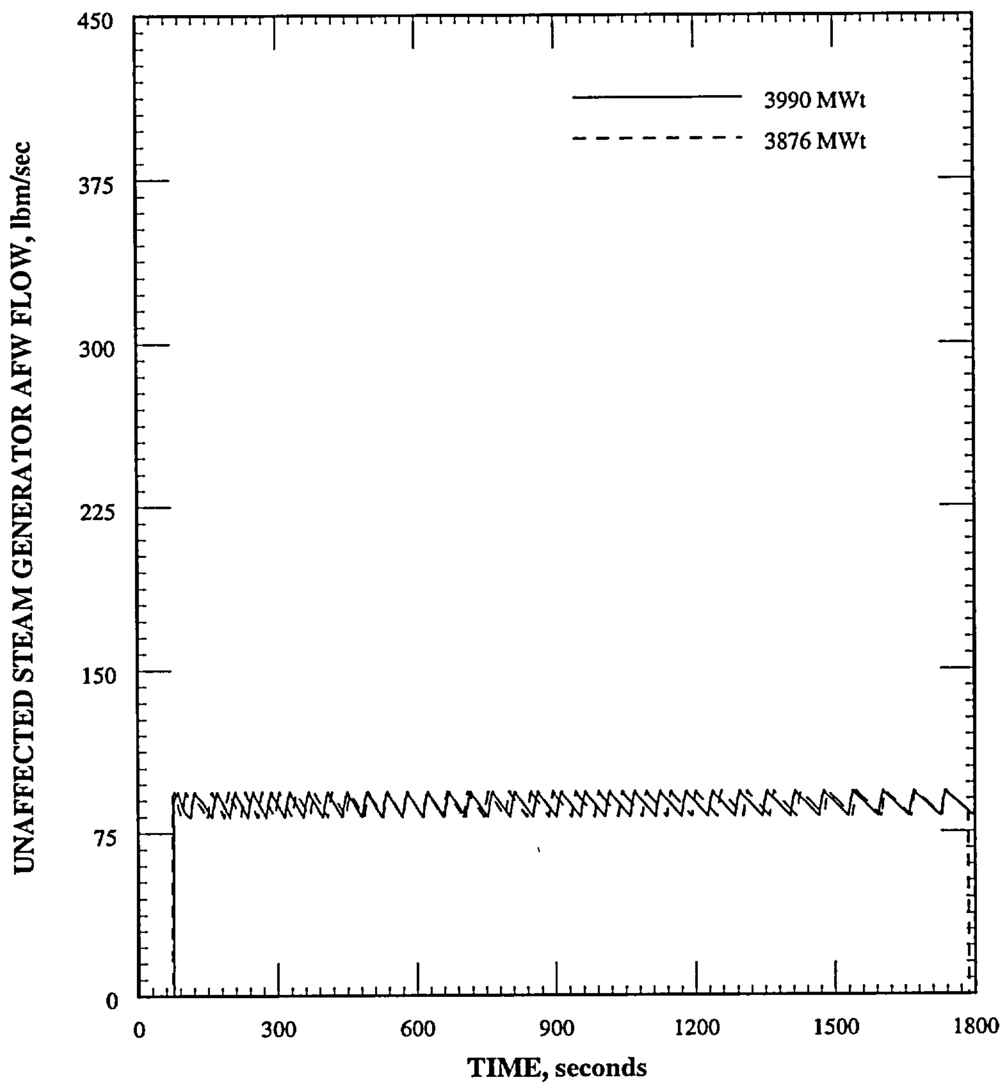

Figure 6.3-121
FWLB with LOP Long Term Cooling – Break Flow vs. Time

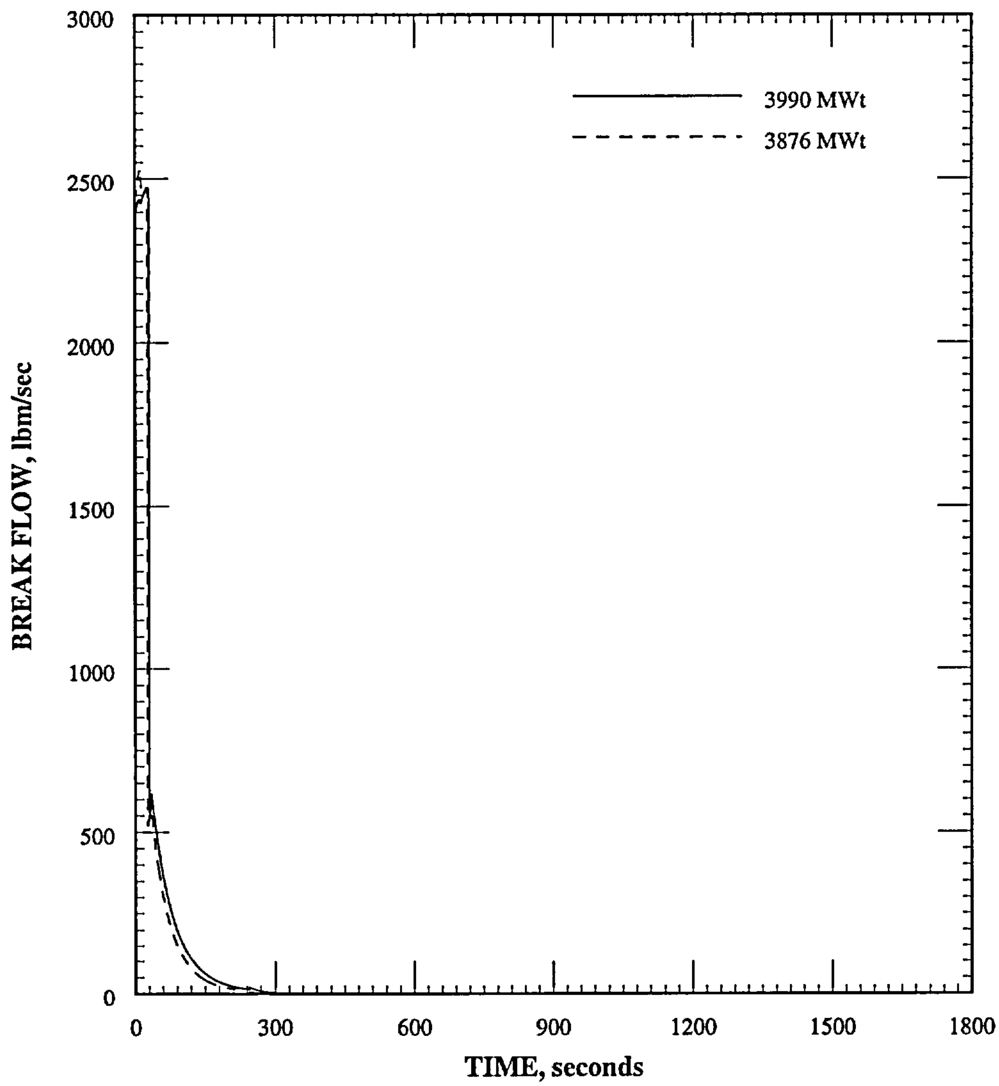

Figure 6.3-122
FWLB with LOP Long-Term Cooling – PSV Flow vs. Time