Palo Verde Nuclear
Generating Station

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10 CFR 50.90

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102-04641-CDM/RAB
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, D.C. 20555

Dear Sirs:

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

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) submits herewith a request to amend Facility Operating License NPF-51 and Appendix A, Technical Specifications for PVNGS Unit 2. The proposed changes support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94% increase. Operating License and Technical Specification changes associated with this Power Uprate (PUR) amendment request are described in Attachment 2, "License Amendment Request Analysis." As noted in Attachment 2, some of the proposed changes are being made to accomplish the PUR and others are needed both to accomplish the PUR as well as the steam generator replacement.

The uprate program included a reanalysis or evaluation of the Large Break Loss of Coolant Accident (LBLOCA), Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) Structures, Systems and Components (SSCs). Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators) and BOP components (e.g., safety injection, auxiliary feedwater, shutdown cooling, electrical distribution, emergency diesel generators, containment cooling and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., reactor regulating, pressurizer pressure and level, turbine control, feedwater control and steam bypass control) have been evaluated for operation at uprated power conditions. The results of the above analyses and evaluations have yielded acceptable results and demonstrate that applicable design basis acceptance criteria will continue to be met during uprated power operations.

The Reactor Protective System and Engineered Safety Features Actuation System set points assessment determined that low steam generator pressure set points for reactor trip and main steam isolation need to be changed and, as such, are part of this amendment request as described in Attachment 2.

The analyses performed to support PUR assume that the Replacement Steam Generators (RSGs) have been installed. These analyses demonstrate that PVNGS Unit 2 continues to meet applicable licensing criteria. Analyses and evaluations that have been performed in support of PUR include the application of methods and assumptions not previously used for PVNGS. These changes are identified in the Executive Summary of Attachment 6.

In addition to the plant changes directly associated with the Operating License and Technical Specification revisions described in Attachment 2, the containment spray flow instrumentation will be changed to provide increased margin for surveillance testing, the spray pond temperature monitoring system will be improved, and the steam admission logic for the high pressure turbine will be changed from partial arc admission to full arc admission. Additionally, setpoints in the core operating limit supervisory system, pressurizer level control system, feedwater control system, and steam bypass control system will be adjusted. These changes are discussed in Attachment 6 and do not require changes to the Technical Specifications.

The proposed PUR would be implemented during the plant start-up after the steam generators are replaced in Unit 2 during refueling No. 11, scheduled for the fall 2003 outage. APS requests approval of these proposed amendments by September 1, 2002 so that approved values may be used in the core reload design scheduled to begin in September 2002. APS requests to implement these changes prior to the entry into Mode 4 after refueling outage No. 11, currently scheduled for December 2003.

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 amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

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/RAB/kg

Attachments:

1. Notarized Affidavit
2. License Amendment Request Analysis
3. Markup of Technical Specification Pages
4. Retyped Technical Specification Pages
5. Associated Changes to the Technical Specification Bases (for information only)
6. Power Uprate Licensing Report

Enclosure:

CD-ROM (PDF Normal format) - Palo Verde Nuclear Generating Station, Unit 2, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations

cc:	E. W. Merschoff	(NRC Region IV)	(w/attachments)
	L. R. Wharton	(NRR Project Manager)	(w/attachments & enclosure)
	J. H. Moorman	(NRC Resident Inspector)	(w/attachments)
	A. V. Godwin	(ARRA)	(w/attachments)

Attachment 1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

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

David Mauldin

David Mauldin

Sworn To Before Me This 21st Day Of December, 2001.

Karen D Greiner

Notary Public

"OFFICIAL SEAL"
Karen D. Greiner
Notary Public-Arizona
Maricopa County
My Commission Expires 8/28/2004

Notary Commission Stamp

ATTACHMENT 2

LICENSE AMENDMENT REQUEST ANALYSIS

1.0 DESCRIPTION OF PROPOSED AMENDMENT

2.0 BACKGROUND

3.0 SAFETY ANALYSIS

4.0 SIGNIFICANT HAZARDS ANALYSIS

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

7.0 PRECEDENT

1.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed amendment would allow operation of PVNGS Unit 2 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94 percent above the current licensed power level of 3876 MWt. Specifically, the following Unit 2 Facility Operating License and Technical Specification changes are requested to support the increased power operation:

A. Revise paragraph 2.C.(1) of the Facility Operating License to increase the authorized 100% reactor core power (rated thermal power) from 3876 MWt to 3990 MWt, an increase of 2.94%. The new power level of 3990 MWt represents an increase of 5% above the originally licensed power level of 3800 MWt. The increase to 3876 MWt was authorized by the NRC in a letter dated May 23, 1996, Amendment No. 100 for Unit 2.

B. Revise Technical Specification Section 1.1, Definition of Rated Thermal Power, for Unit 2, from 3876 MWt to 3990 MWt.

C. Revise Table 3.3.1-1, Reactor Protective System Instrumentation (referenced in LCO 3.3.1), item 6, Steam Generator #1 Pressure - Low and item 7, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable reactor protection system response. Both the power uprate and the Replacement Steam Generators (RSGs) affect this specification.

D. Revise Table 3.3.2-1, Reactor Protective System Instrumentation - Shutdown (referenced in LCO 3.3.2), item 2, Steam Generator #1 Pressure - Low and item 3, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable reactor protection system response. Both the power uprate and the RSGs affect this specification.

E. Revise Table 3.3.5-1, Engineered Safety Features Actuation System Instrumentation (referenced in LCO 3.3.5), item 4.a, Steam Generator #1 Pressure - Low and item 4.b, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable engineered safety features system response. Both the power uprate and the RSGs affect this specification.

F. Revise Figure 3.4.1-1, Reactor Coolant Cold Leg Temperature vs. Core Power Level, to change the upper limit in the area of acceptable operation for Unit 2. The new upper limit line would allow a cold leg temperature of 570 °F at 0% power, decreasing linearly to 564 °F at 100% power. The upper limit line of Figure 3.4.1-1 in the current Technical Specification decreases linearly from 570 °F at 0% power to 568 °F at 30% power. At 30% power the current figure then decreases linearly from 568 °F to 560 °F at 100% power. The increase in cold leg temperature at 100% power will allow a more optimum main steam pressure for turbine operation. Additionally, editorial changes will be made to Figure 3.4.1-1 for Units 1 and 3. Both the power uprate and RSGs affect this specification.

G. Revise Table 3.7.1-1, Variable Overpower Trip (VOPT) Setpoint versus Operable Main Steam Safety Valves for Unit 2, to decrease the Maximum Power and the Maximum Allowable VOPT Setpoint when the Minimum Number of Main Steam Safety Valves (MSSVs) per Steam Generator Required Operable is less than ten. The reduction in allowable power levels and VOPT setpoints for Unit 2 are required to offset the impacts of increased core power and increased cold leg temperature on the consequences of the UFSAR Chapter 15 design basis events. The power uprate affects this specification.

H. Revise Technical Specification 5.5.16, Containment Leakage Rate Testing Program, to increase the peak calculated containment internal pressure for the design basis loss of coolant accident (P_a) for Unit 2 from 52.0 psig to 58.0 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig. The calculated peak containment pressure remains below the containment internal design pressure of 60.0 psig. Both the power uprate and the RSGs affect this specification.

The bases for Technical Specifications 3.6.1, 3.6.2, 3.6.4, 3.6.6 and 3.7.1 would also be revised to reflect these changes and are included in Attachment 5 of this submittal.

2.0 BACKGROUND

This proposed amendment is requested to improve the economic performance of PVNGS Unit 2. Increasing the rated thermal power limit of PVNGS Unit 2 from 3876 MWt to 3990 MWt would result in an increase in electrical output of approximately 55 Megawatts electric (MWe).

The original full power operating license for Unit 2, issued in April 1986, authorized a Rated Thermal Power (RTP) of 3800 MWt. In May 1996, the NRC issued Amendment Nos. 108, 100 and 80 to Units 1, 2 and 3, respectively, to increase the authorized RTP to 3876 MWt. This amendment request to increase RTP to 3990 MWt would be a 2.94% increase above that authorized in Amendment 100, and represents a 5% increase from the original RTP.

Paragraph 2.C.(1) of the Facility Operating License specifies, as a license condition, the maximum reactor core thermal power level at which the unit is authorized to operate. The maximum authorized reactor thermal power level is specified as a license condition in order to limit thermal power to the value used in the safety analyses. The maximum reactor core thermal power specified in the operating license is also known as the Rated Thermal Power (RTP). Regulatory Guide 1.49 recommends a 2% uncertainty be included in the power level used in the safety analysis, as appropriate. Thus, the safety analysis supporting this amendment uses a reactor core thermal power of 4070 MWt, which is 102% of 3990 MWt, the proposed new RTP. The definition of Rated Thermal Power in Technical Specification 1.1 identifies the licensed limit of the total reactor core heat transfer rate to the reactor coolant.

LCO 3.3.1, Reactor Protective System Instrumentation - Operating and Table 3.3.1-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 6 and 7 in Table 3.3.1-1) provides protection against an excessive rate of heat extraction from the steam generators and the resulting rapid, uncontrolled cooldown of the Reactor Coolant System (RCS). This trip is needed to shut down the reactor and assist the Engineered Safety Features (ESF) system in the event of a Main Steam Line Break (MSLB) or Main Feedwater Line Break (MFWLB) accident. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

LCO 3.3.2, Reactor Protective System Instrumentation - Shutdown and Table 3.3.2-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 2 and 3 in Table 3.3.2-1) provides shutdown margin to prevent or minimize the return to power following a large MSLB in Mode 3.

LCO 3.3.5, Engineered Safety Features Actuation System Instrumentation and Table 3.3.5-1, which it references specify the required number of channels operable for each reactor trip function, the applicable modes for each function, and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low signal actuates a MSIS to prevent an excessive rate of heat extraction and subsequent cooldown of the RCS in the event of a MSLB or MFWLB.

Figure 3.4.1-1, Reactor Coolant Cold Leg temperature vs. Core Power Level, referenced in LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, provides parametric limits to ensure that the actual value of the reactor coolant cold leg temperature is maintained within the range of values used in the safety analysis. The safety analysis supporting this requested amendment uses the proposed new allowable cold leg temperature range (550 °F to 570 °F), and this proposed change maintains the basis for the cold leg temperature limits.

Table 3.7.1-1, Variable Overpower Trip Setpoint versus Operable Main Steam Safety Valves, referenced in LCO 3.7.1, Main Steam Safety Valves (MSSVs), specifies maximum power levels and overpower reactor trip setpoints for specified numbers of

OPERABLE MSSVs. Adherence to the values in the table will ensure that the available relieving capacity maintains secondary system pressure within allowable limits.

Technical Specification 5.5.16, Containment Leakage Rate Testing Program, provides the requirements for the Containment Leakage Rate Testing Program. The calculated peak containment internal pressure for the design basis LOCA (P_a) is the basis for the containment leakage rate in the testing program.

3.0. SAFETY ANALYSIS

Refer to Attachment 6 (Power Uprate Licensing Report).

4.0. NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated,
2. create the possibility of a new or different kind of accident from any previously analyzed, or
3. involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

APS has completed a comprehensive reanalysis or evaluation of Large Break Loss of Coolant Accidents (LBLOCA), Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) structures, systems, and components to demonstrate the acceptability of increasing the licensed reactor power from 3876 Megawatts-thermal (MWt) to 3990 MWt for Unit 2.

Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators), BOP components (e.g., main turbine, generator, and condensate and feedwater pumps), and major systems and sub-systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment spray, and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., reactor regulating system, pressurizer pressure and level control, turbine control, feedwater control, and steam bypass control) have been evaluated for operation at uprated power conditions. Reactor trip and ESF actuation setpoints have been assessed, and only the changes to low steam generator pressure, were identified as a result of uprated power operations or SG replacement. The analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations. This detailed analysis is presented in the "Power Uprate Licensing Report," submitted as Attachment 6 to this license amendment request.

Standard 1 - Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

a. Evaluation of the Probability of Previously Evaluated Accidents

Plant Structures, Systems and Components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, a small number of minor modifications will be made prior to implementation of uprated power operations so that surveillance test acceptance criteria continues to be met. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current technical specification surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated within current operating requirements at uprated conditions. Therefore, no new structure, system or component interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

b. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (i.e., both LOCA and non-LOCA accidents) previously analyzed in the UFSAR. The analyses showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remained within regulatory and Standard Review Plan (SRP) limits at uprated power conditions.

Standard 2 - Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The configuration, operation and accident response of the PVNGS Unit 2 structures, systems, and components are unchanged by operation at uprated power conditions or by the associated proposed Technical Specification changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident or different scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing SSCs. The basic design function of all SSCs remains unchanged and no new equipment or systems have been installed that could potentially introduce new failure modes or accident sequences.

Based on these analyses, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not have an adverse effect on any safety-related system or design basis function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>Standard 3</u> - Do the proposed changes involve a significant reduction in a margin of safety?

No. The proposed changes do not involve a significant reduction in a margin of safety.

A comprehensive analysis was performed to evaluate the effects of power uprate on PVNGS Unit 2. This analysis identified and defined the major input parameters to the NSSS, reviewed NSSS design transients, and reviewed the capabilities of the NSSS and BOP fluid systems, NSSS/BOP interfaces, NSSS and BOP control systems, and NSSS and BOP SSCs. NSSS accident analyses were re-performed or reviewed to confirm that acceptable results were maintained and that the radiological consequences remained within regulatory and Standard Review Plan (SRP) limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analyses confirmed that all NSSS and BOP SSCs are capable, some with minor modifications, to safely support operations at uprated power conditions.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicates that the calculated peak containment pressure (P_a) increases from 52.0 psig to 58.0 psig, but remains less than the containment internal design pressure of 60 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig.

Radiological consequences of the following accidents were reviewed: Main Steam Line Break, Locked Reactor Coolant Pump (RCP) Rotor, CEA Ejection, Small Steam Line Break Outside Containment, Steam Generator Tube Rupture, LBLOCA, SBLOCA, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents remained within regulatory and SRP limits at uprated power conditions.

The analyses supporting operation at power uprate conditions have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

Based upon the above analyses and evaluations, APS has concluded that the proposed changes to the Unit 2 operating license and Technical Specifications involve no significant hazards consideration.

5.0. <u>ENVIRONMENTAL CONSIDERATION</u>

APS has determined that the proposed amendment does not involve an unreviewed environmental question, in accordance with Section 3.1 of Appendix B of the Operating License. A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may 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) a significant change in the effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) above, which may have a significant adverse environmental impact. Based on the following, this amendment request does not constitute an unreviewed environmental question:

1) A matter which may 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.

APS reviewed the FES and determined that this amendment request does not significantly increase any adverse environmental impact. The plant is not being modified in any way which would significantly increase or change the type of effluents currently produced. The 2.94% increase in RTP is within the 4100 MWt design stretch power evaluated in the FES - Construction Permit Stage (FES-CP). Thus the environmental effects as a result of the uprate are bounded by those previously evaluated during FES-CP phase.

Radiological releases are controlled in accordance with PVNGS Offsite Dose Calculation Manual and the results are reported annually to the NRC. Design Basis Event radiological releases have been demonstrated, in the safety analysis provided with this amendment request, to not significantly increase offsite exposure and remain within regulatory limits. The radiological exposure to plant workers is controlled under the PVNGS As Low As Reasonably Achievable (ALARA) Program and will not significantly change.

2) A significant change in the effluents or power level.

A 2.94% increase in RTP is not a significant increase in power level. The Final Environmental Statement (NUREG 0841) recognized in the Summary and Conclusions Section that the maximum design thermal output for each unit is 4100 MWt. The proposed increase is less than the FES-CP evaluated maximum design thermal output of the units. Thus the environmental effects previously evaluated for land and water usage are bounded by those previously evaluated. The increase in RTP does not change any of the conclusions of NUREG 0841.

Effluents as discussed above will not be significantly increased and are controlled by PVNGS programs and applicable regulations.

3) A matter not previously reviewed and evaluated in the documents specified in (1) above which may have a significant adverse environmental impact.

The increase in RTP does not change the processes, plant equipment, types of effluents, or significantly affect operation of the units. The changes are within the design basis of the NSSS and BOP SSCs at the increased RTP conditions. Safety analyses of design basis events affected by the increase have been reviewed or reanalyzed and the consequences found to be bounded by current UFSAR consequences or within regulatory requirements. The FES-CP, FES-OL, and NUREG-0841 all evaluated the environmental impact assuming the maximum design thermal output of 4100 MWt for each unit. Thus the proposed increase in rated thermal power is within the scope of the previous reviews performed to assess the environmental impact associated with the operation of each unit.

Based on the above, no unreviewed environmental question exists concerning this amendment request for increased RTP and associated Technical Specification changes.

6.0 REFERENCES

References used to develop this request are listed at the end of each section in the Power Uprate Licensing Report (Attachment 6).

7.0 PRECEDENT

A similar amendment request has been approved for the following facilities:

Facility	Amendment #	Approval Date	Accession #
Farley 1, 2	137, 139	April 29, 1998	Not Available
Byron 1, 2	119, 119	May 4, 2001	ML011420274
Braidwood 1, 2	113, 113	May 4, 2001	ML011420274

ATTACHMENT 3

MARKED-UP OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

Marked-up Operating License and Technical Specification Pages

Unit 2 Operating License

Page 5

Technical Specifications

Page 1.1-6
Page 3.3.1.8
Page 3.3.2-5
Page 3.3.5-4
Page 3.4.1-3
Page 3.7.1-3
Page 5.5-23

(8)(a) Arizona Public Service Company is authorized to transfer all or a portion of its 29.1% ownership share in Palo Verde, Unit 2 to certain equity investors identified in its submissions of August 6, August 8 and December 5, 1986, and at the same time to lease back from such purchasers such interest sold in the Palo Verde, Unit 2 facility. The term of the lease is for approximately 29-½ years subject to a right of renewal. Additional sale and leaseback transactions of all or a portion of APS's remaining ownership share in Palo Verde, Unit 2 are hereby authorized until June 30, 1987. Any such sale and lease back transaction is subject to the representations and conditions set forth in the aforementioned application of May 2, 1986, and the subsequent submittals dated July 30, August 2, August 6, August 7, August 8, August 13, October 16 and December 5, 1986, as well as the letters of the Director of the Office of Nuclear Reactor Regulation dated August 15, and December 11, 1986, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 2. For purposes of this condition the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to the lessor as long as the license for Palo Verde, Unit 2 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 2 as specified in licensee counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of ~~3876~~ 3990 megawatts thermal (100% power) in accordance with the conditions specified herein.

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt FOR UNITS 1 AND 3, AND 3990 MWt FOR UNIT 2
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length CEA position.

(continued)

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High(d)	3(a), 4(a), 5(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP(c)
2. Steam Generator #1 Pressure-Low(b)	3(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA
3. Steam Generator #2 Pressure-Low(b)	3(a)	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	UNIT 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[(a)]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[(a)]		≥ 1821 psia
4. Main Steam Isolation Signal[(c)]		
a. Steam Generator #1 Pressure–Low[(b)]	1,2,3	UNITS 1 & 3: ≥ 890 psia UNIT 2: ≥ 955 PSIA
b. Steam Generator #2 Pressure–Low[(b)]		UNITS 1 & 3: ≥ 890 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level–Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid

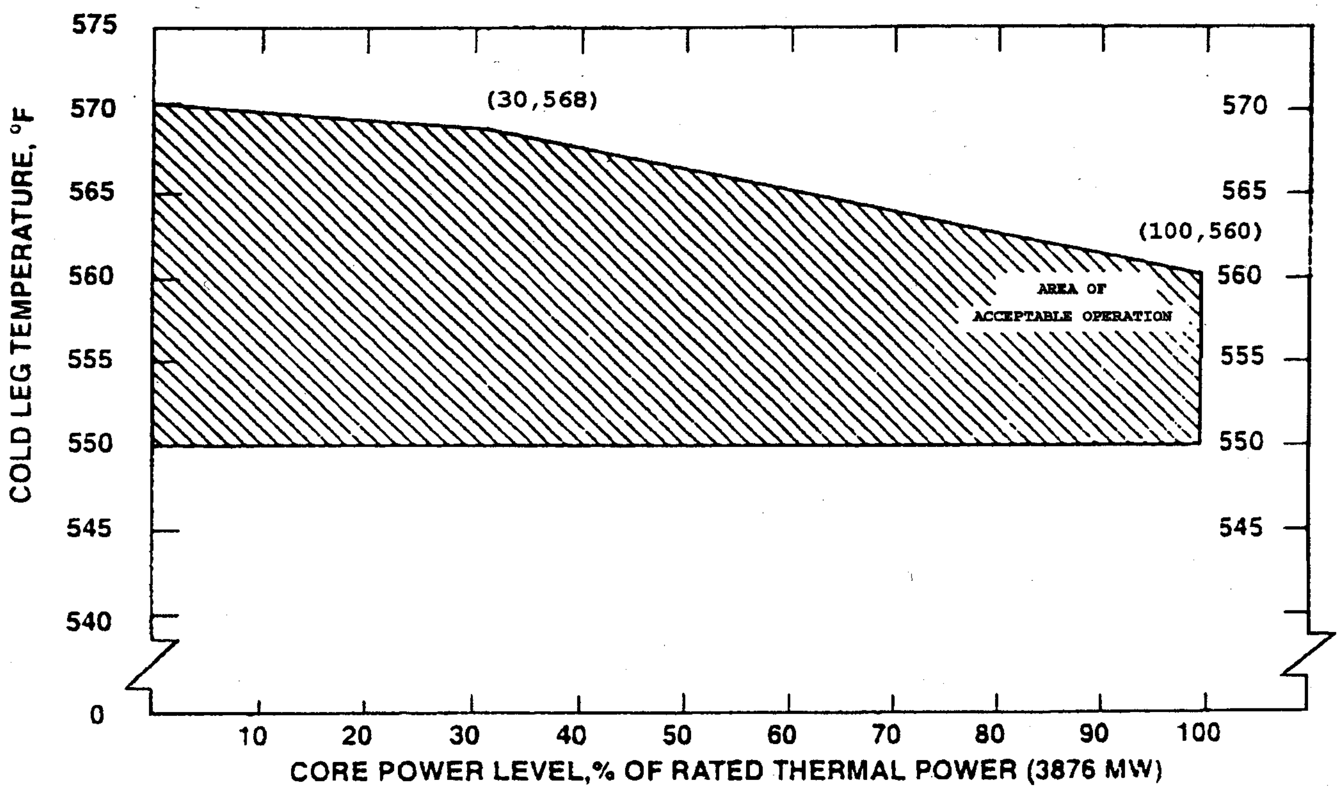
(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

Figure 3.4.1-1
Reactor Coolant Cold Leg Temperature vs. Core Power Level

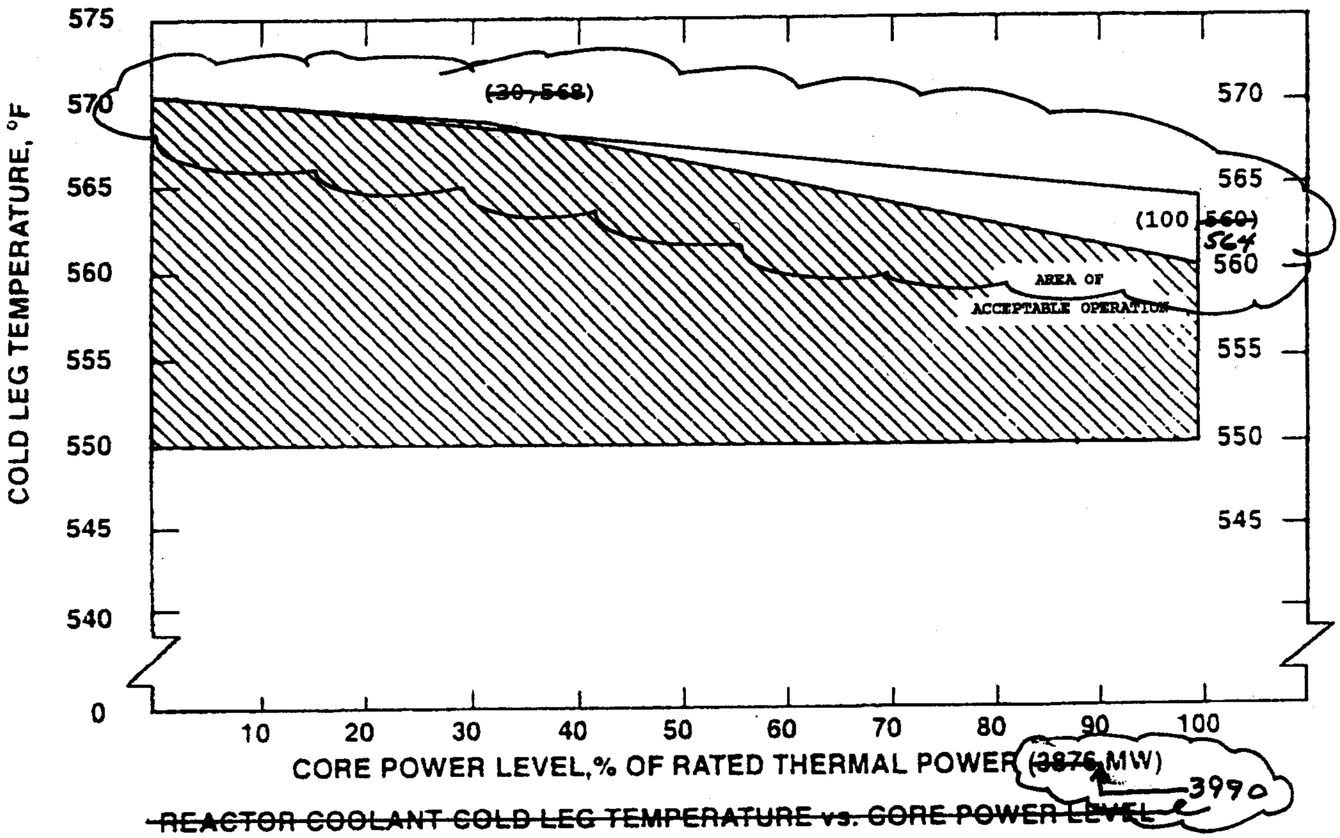
UNITS 1 AND 3



~~REACTOR COOLANT COLD LEG TEMPERATURE vs. CORE POWER LEVEL~~

Figure 3.4.1-1
Reactor Coolant Cold Leg Temperature vs. Core Power Level

UNIT 2



~~REACTOR COOLANT COLD LEG TEMPERATURE vs. CORE POWER LEVEL~~

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	UNITS 1 & 3	UNIT 2	UNITS 1 & 3	UNIT 2
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

FOR UNITS 1 AND 3, AND 55.0 PSIG FOR UNIT 2

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

ATTACHMENT 4

RETYPED OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

Retyped Operating License and Technical Specification Pages

<u>Unit 2 Operating License</u>

Page 5

<u>Technical Specifications</u>

Page 1.1-6
Page 3.3.1-8
Page 3.3.2-5
Page 3.3.5-4
Page 3.4.1-3
Page 3.4.1-4
Page 3.7.1-3
Page 5.5-23

(8)(a) Arizona Public Service Company is authorized to transfer all or a portion of its 29.1% ownership share in Palo Verde, Unit 2 to certain equity investors identified in its submissions of August 6, August 8 and December 5, 1986, and at the same time to lease back from such purchasers such interest sold in the Palo Verde, Unit 2 facility. The term of the lease is for approximately 29-½ years subject to a right of renewal. Additional sale and leaseback transactions of all or a portion of APS's remaining ownership share in Palo Verde, Unit 2 are hereby authorized until June 30, 1987. Any such sale and lease back transaction is subject to the representations and conditions set forth in the aforementioned application of May 2, 1986, and the subsequent submittals dated July 30, August 2, August 6, August 7, August 8, August 13, October 16 and December 5, 1986, as well as the letters of the Director of the Office of Nuclear Reactor Regulation dated August 15, and December 11, 1986, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 2. For purposes of this condition the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to the lessor as long as the license for Palo Verde, Unit 2 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 2 as specified in licensee counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power) in accordance with the conditions specified herein.

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt for Units 1 and 3, and 3990 MWt for Unit 2.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length CEA position.

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[(a)]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	3[a], 4[a], 5[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
3. Steam Generator #2 Pressure-Low[b]	3[a]	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced. provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[(a)]		≥1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[(a)]		≥ 1821 psia
4. Main Steam Isolation Signal[(c)]		
a. Steam Generator #1 Pressure—Low[(b)]	1,2,3	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
b. Steam Generator #2 Pressure—Low[(b)]		Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level—Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level—Low	1,2,3	≥ 25.3%
b. SG Pressure Difference—High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level—Low	1,2,3	≥ 25.3%
b. SG Pressure Difference—High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

Figure 3.4.1-1, (Page 1 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

Units 1 and 3

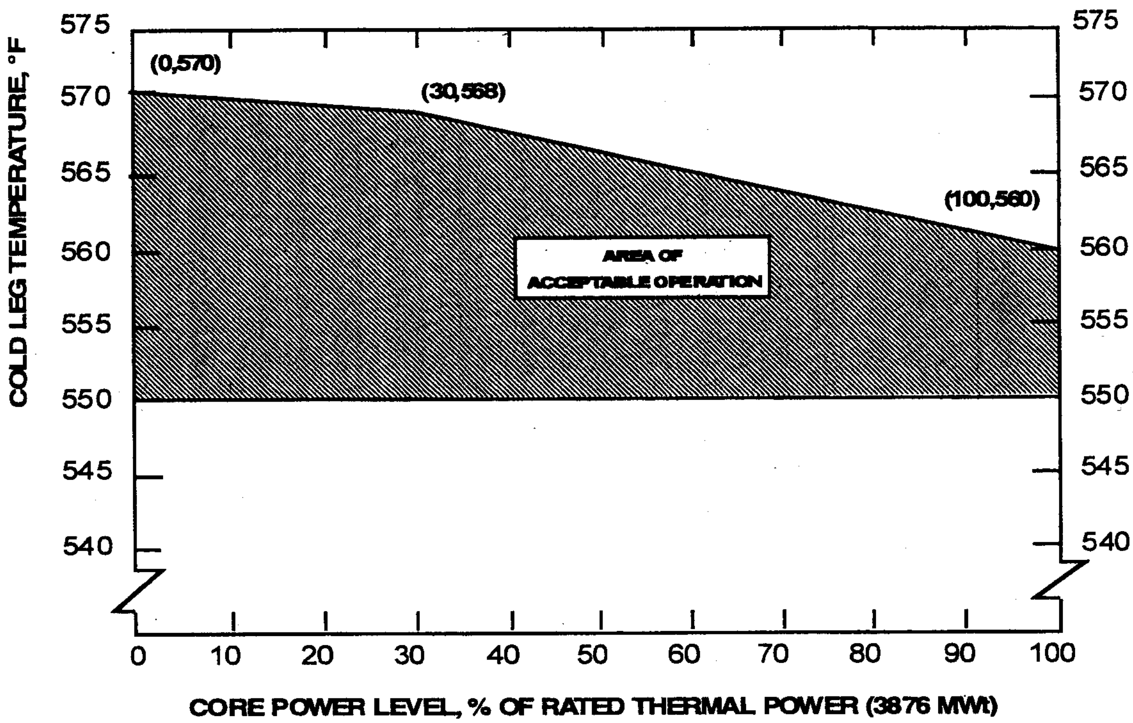


Figure 3.4.1-1, (Page 2 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

Unit 2

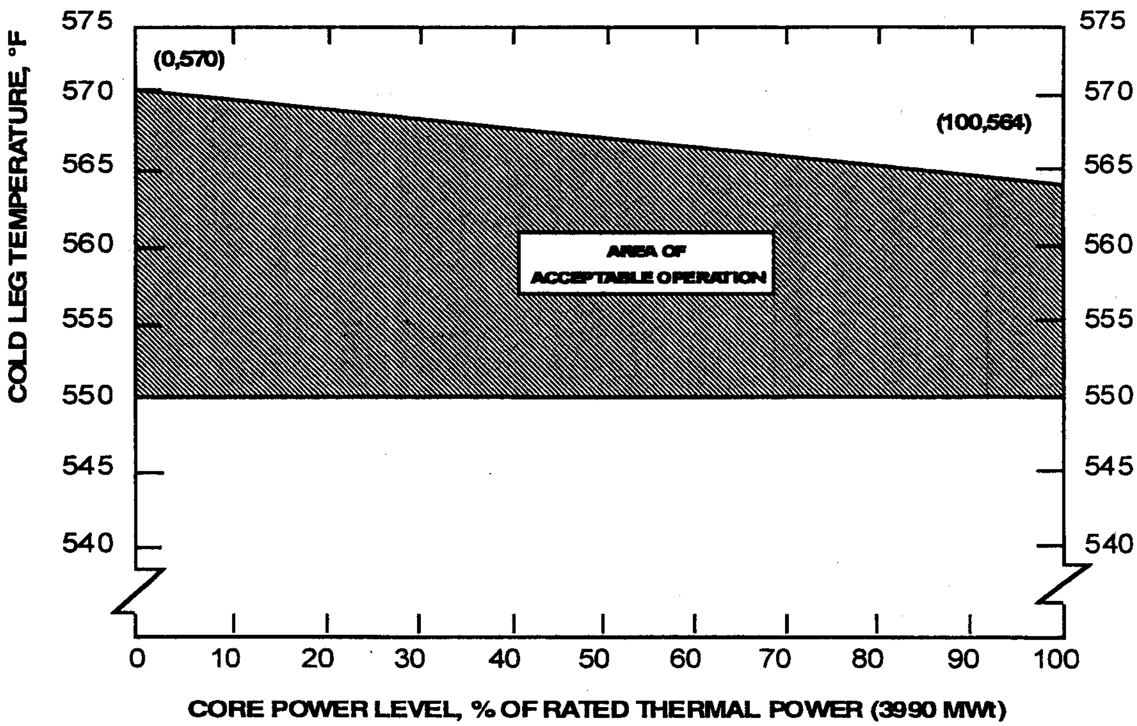


Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	Units 1 and 3	Unit 2	Units 1 and 3	Unit 2
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Units 1 and 3, and 58.0 psig for Unit 2. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

ATTACHMENT 5

ASSOCIATED CHANGES TO PVNGS TECHNICAL SPECIFICATION BASES

(Information Only)

Associated Changes to Technical Specification Bases

<u>Bases</u>

Page B 3.6.1-2
Page B 3.6.2-2
Page B 3.6.4-1
Page B 3.6.6-3
Page B3.7.1-1
Page B 3.7.1-2
Page B 3.7.1-3
Page B 3.7.1-4
Page B 3.7.1-3 (modified for Unit 2 only)
Page B 3.7.1-4 (modified for Unit 2 only)
Page B 3.7.1-5
Page B 3.7.1-6

BASES (continued)

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of 52.0 psig FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2, which results from the limiting design basis LOCA.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a [52.0 psig], FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2 following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs ~~associated with the~~ air lock.

The containment air locks satisfy ~~Criterion 3 of~~ 10 CFR 50.36 (c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. A double ended discharge line break LOCA with maximum ECCS results in the highest calculated internal containment pressure of 52.0 psig, which is below the internal design pressure of 60 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) Systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis bounds the containment pressure allowed during normal operation. The LCO limit of 2.5 psig ensures that, in the event of an accident, the maximum peak containment internal pressure, 52.0 psig, and the maximum accident design pressure for containment, 60 psig, are not exceeded.

(continued)

BASES

BACKGROUND (continued)

The Containment Spray System accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

APPLICABLE SAFETY ANALYSES

The Containment Spray System limits the temperature and pressure that could be experienced following a DBA. The Containment Spray System is required to be capable of reducing containment pressure to 1/2 the peak pressure within 24 hours following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the Loss Of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable.

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 52.0 psig (experienced during a LOCA). The analysis shows that the peak containment vapor temperature is 405.65°F (experienced during a MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 102% RTP, one containment spray train operating, and initial (pre-accident) conditions of 120°F and 16.7 psia (LOCA) and 13.22 psia (MSLB). The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to -2.6 psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specification 3.6.4.

(continued)

NO CHANGES ON THIS PAGE
INCLUDED FOR CONTINUITY

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Reactor Coolant Pressure Boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each of the four main steam lines, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 5.2 (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering if there is insufficient steam pressure to fully open all valves.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power Loss Of Condenser Vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater to the steam generators, RCS pressure reaches ≤ 2742 psia. This peak pressure is < 110% of the design pressure of 2500 psia, but high enough to actuate the pressurizer safety valves.

(continued)

NO CHANGES ON THIS PAGE
INCLUDED FOR CONTINUITY

APPLICABLE SAFETY ANALYSES (continued)

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2843 psia, with the pressurizer safety valves providing relief capacity. These results were found acceptable by the NRC based on the low probability of the event.

The MSSVs satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1 and Required Action A.2 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

(continued)

BASES

APPLICABILITY

In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2 — UNITS 1 AND 3 ONLY

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued) - UNITS 1 AND 3 ONLY

$$\text{Allowable THERMAL POWER} = \frac{(10 - N)}{10} \times 109.2$$

With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.

$$SP = \text{Allowable THERMAL POWER} + 9.8$$

where:

SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = Total number of MSSVs per steam generator.

N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.

109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).

9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

APPLICABILITY

In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES..

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2 - UNIT 2 ONLY

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.

IN ACCORDANCE WITH TABLE 3.7.1-1, THESE REDUCED POWER LEVELS, DERIVED FROM THE TRANSIENT ANALYSIS, COMPENSATE FOR DEGRADED RELIEVING CAPACITY AND ENSURE THAT THE RESULTS OF THE TRANSIENT ANALYSIS ARE ACCEPTABLE.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued) – UNIT 2 ONLY

$$\text{Allowable THERMAL POWER} = \frac{(10 - N)}{10} \times 109.2$$

With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.

SP = Allowable THERMAL POWER + 9.8

where:

SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = Total number of MSSVs per steam generator.

N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.

109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).

9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than six MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 5.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV.
5. ANSI/ASME OM-1-1987.

Attachment 6

ATTACHMENT 6

POWER UPRATE LICENSING REPORT

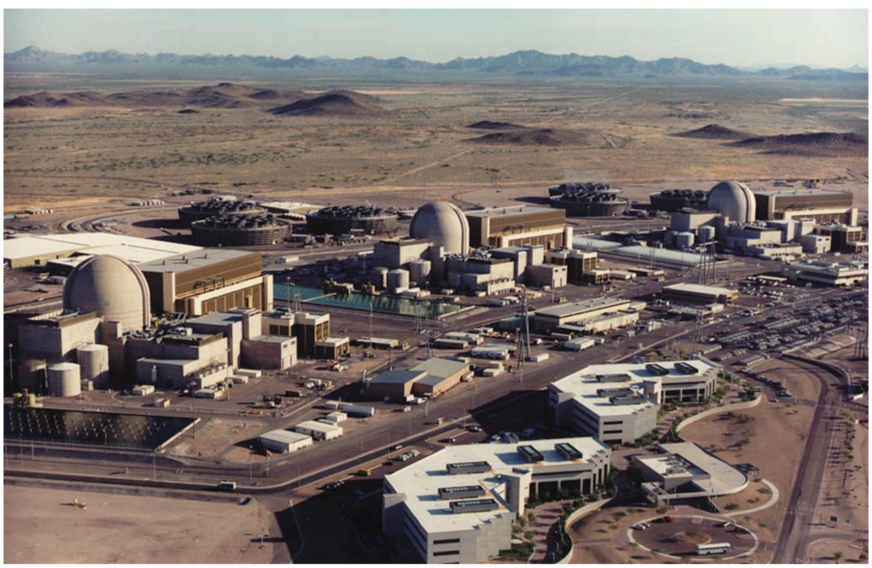
Arizona Public Service Company
Palo Verde Nuclear Generating Station

Power Uprate Licensing Report

for the

Palo Verde Nuclear Generating Station

Unit 2



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POWER UPRATE LICENSING REPORT EXECUTIVE SUMMARY

The purpose of this license amendment request is to increase the electrical output (MW_e) of the Palo Verde Nuclear Generating Station (PVNGS) Unit 2. This will be accomplished by requesting NRC approval to increase the licensed 100% reactor core power level from 3876 MW_t to 3990 MW_t, a 2.94% increase.

In support of the Power Uprate (PUR), Arizona Public Service (APS), the operator of PVNGS, and Westinghouse Combustion Engineering Nuclear Power LLC (W CENP) have performed analyses and evaluations for the Nuclear Steam Supply System (NSSS). These analyses demonstrate that the APS complies with applicable licensing criteria and design requirements at the uprated reactor power of 3990 MW_t (NSSS thermal power of 4013 MW_t, which is the sum of core and Reactor Coolant Pump (RCP) heat). The scope of the analyses and evaluations included the:

- NSSS performance parameters,
- design transients (used in stress analysis),
- Structures, Systems, and Components (SSCs),
- Design Basis Accidents (DBAs),
- nuclear fuel design, and
- secondary side Balance of Plant (BOP).

This report provides a description of the analyses and evaluations performed for the PUR. The focus of this report is on providing the information required by the NRC to approve PUR for the PVNGS Unit 2. Before implementing PUR, APS is replacing Steam Generators (SGs) with larger generators. The design and installation of the Replacement Steam Generators (RSGs) is being conducted under the licensing provisions of 10 CFR Part 50.59. Evaluations and analyses supporting the PUR assume the installation of the RSGs. Therefore, conclusions of the analyses that are reported in this document include the RSG design and operational characteristics. Section 5.5 of this report provides an overview of the changes between the RSGs and the existing installed SGs.

In addition to the PUR, the following change has been applied to methods of evaluation using computer codes:

- change in the input model of computer code SGNIII to demonstrate more plant specific results as described in Section 6.2.4.3.2.

The following revised methods/assumption changes have been applied to Non-Loss-of-Coolant Accident (LOCA) transient analyses:

- 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 6.3.1.5.5.1.
- 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 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.

Section 1 INTRODUCTION

Section 1.1 Purpose and Scope

Various analyses have been performed for Power Uprate (PUR) to demonstrate compliance with applicable regulatory, design, and operational requirements at the increased thermal power conditions.

Arizona Public Service (APS), Westinghouse Combustion Engineering Nuclear Power LLC (W CENP), and Ansaldo (the steam generator (SG) fabricator) performed the various analyses/evaluations for the PUR. The scope includes the Nuclear Steam Supply System (NSSS) and secondary side Balance of Plant (BOP) performance parameters. The analyses included design transients (used in stress analyses); safety analyses; Structures, Systems, and Component (SSCs) evaluations; and the response of nuclear fuel to the PUR condition. Before the PUR, NSSS analyses for Emergency Core Cooling System (ECCS) performance were analyzed to PUR conditions, making reanalysis unnecessary (see Section 6.1).

The PUR analyses and evaluations described in this report were based on the parameters listed in Table 2.1-1.

Section 1.2 Methodology and Acceptance Criteria

The analyses performed by APS employed methodologies that have been previously approved by the NRC unless noted. Changes in analytical methodologies, specifically used in Section 6 are outlined in the Executive Summary.

All PUR analyses and evaluations were performed in accordance with the quality assurance requirements defined in the APS Quality Assurance (QA) Plan (as specified in Updated Final Safety Analysis Report (UFSAR), Reference 1-2) and procedures that comply with 10 CFR Part 50, Appendix B criteria (Reference 1-1). These analyses and evaluations are in conformance with industry codes, standards, and regulatory requirements. Assumptions and acceptance criteria are provided in the appropriate sections of this report.

Section 1.3 Technical Basis for No Significant Hazards Consideration Determination

This report provides the technical basis for the No Significant Hazards Consideration Determination included with the proposed license amendment changes for the PVNGS Unit 2 PUR.

Section 1.4 Regulatory Guide Compliance

PVNGS UFSAR Section 1.8 discusses the conformance of plant design with the guidelines presented in the NRC Regulatory Guides. This PUR does not deviate from the regulatory compliance as listed in Section 1.8.

Section 1.5 Conclusions

The analyses and evaluations conclude that PVNGS Unit 2 can operate within licensed parameters at the PUR conditions.

Section 1.6 References

Reference 1-1 Code of Federal Regulations, Title 10, Part 50, “Domestic Licensing of Production and Utilization Facilities.”
Code of Federal Regulations, Title 10, Part 50, Section 50.59 “Changes, Tests, and Experiments.”
Code of Federal Regulations, Title 10, Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

Reference 1-2 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Section 2 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

Arizona Public Service (APS) performed analyses to develop bounding Nuclear Steam Supply System (NSSS) parameters. These parameters include the Reactor Coolant System (RCS) and secondary system operating conditions (temperature, pressure, flow) that are used in the analyses of the NSSS design transients, Structures, Systems and Components (SSC) performance, as well as selected Design Basis Accidents (DBAs) analyses, and nuclear fuel design.

Section 2.1 Performance Parameters

Section 2.1.1 Introduction and Background

Conservative assumptions provide bounding parameters for NSSS analyses. The assumptions rely on engineering judgment, guidance from topical reports, NUREGs, industry experience, etc.

Section 2.1.2 Input Parameters and Assumptions

The input parameters used in the PUR calculations are summarized in Table 2.1-1. These parameters may be adjusted to yield conservative results in each specific analysis. Input parameter adjustments are identified in the discussion for each specific analysis. These conservative adjustments are discussed in the specific analysis section. The major inputs used in generating the parameters are listed below:

- The power level for PUR was set at 4013 MW_t NSSS (3990 MW_t core plus 23 MW_t Reactor Coolant Pump (RCP) heat). This is 2.94% higher than the existing NSSS power rating of 3899 MW_t (3876 MW_t core).
- Allowance for Steam Generator (SG) tube plugging up to 10%.
- RCS best estimate flow (total vessel) is 462,892 gpm (0% plugging), 456,064 gpm (10% plugging).
- A range of RCS SG outlet/inlet best estimate operating temperatures from 556.2 to 614.4 °F (clean and unplugged); 560.4 to 618.9 °F (fouled and plugged) was selected for the analyses.

Section 2.1.3 Acceptance Criteria for Determination of Parameters

The primary acceptance criteria for the determination of the PUR parameters are that the parameters must provide APS with adequate flexibility and margin for plant operation.

Section 2.1.4 Discussion of Parameters

Table 2.1-1 provides the operating parameters that were generated and used as the basis for the PUR. The new parameters were generated considering the Replacement Steam Generators (RSGs). The existing parameters are also shown for comparison purposes. Note that these PUR values are within the existing operating process parameters. The RCS and secondary piping and SSC design pressures and temperatures bound this license amendment request.

Table 2.1-1
Reactor Core and Coolant System Normal Parameters

(Page 1 of 2)

Item	Original UFSAR Table 1.3-1 (1)	Existing Condition (2) (T_{hot} Reduction-Stretch Power)	PUR Condition (3) (0% plugged)	PUR Condition (3) (10% plugged)
Hydraulic and Thermal Design Parameters				
% of original license power	100	102	105	105
Rated core heat output, MW_t	3800	3876	3990	3990
Rated core heat output, Btu/hr	12,970E+06	13,225E+06	13,614E+06	13,614E+06
Principal design parameters of the RCS				
Design pres., psia	2500	2500	2500	2500
Design temp., °F	650	650	650	650
Total flowrate (nominal), lb_m/hr	164E+06 (6)	162.47E+06	172.37E+06	169.82E+06
Operating pres., psia	2250	2250	2250	2250
Reactor inlet temp., (T_{cold}), °F (4)	564.5	554	556.7	560.9
Reactor outlet temp., (T_{hot}), °F	621.2	611	614.4	618.9
Vessel average (T_{ave})	593	582.5	585.6	589.9
Total RCS vol., ft^3 (w/o pressurizer)	12,353	12,353	13,556	13,461
Principal design parameters of the SGs				
Tube side design pres., psia	2500	2500	2500	2500
Tube side design temp., °F	650	650	650	650
Shell side design pres., psia	1270	1270	1270	1270
Shell side design temp., °F	575	575	575	575
Tube side flow, 10^6 lb_m/hr (each)	82 (6)	81.2	86.2	84.9
Steam temp. at full power, °F	552.9	542	549.2	549.2
Max. moisture at outlet, full load, %	0.25	0.25	0.1	0.1
Steam pressure at full power, psia (5)	1070	980	1039	1039
Steam flow at full power, lb_m/hr	N/A (7)	16.9E+06	17.9E+06	17.9E+06

Table 2.1-1
Reactor Core and Coolant System Normal Parameters

(Page 2 of 2)

Notes:
(1) Reference 2-1 original design condition. Note that the original NSSS rating was 3823 MW_t, (3800 MW_t core). The original power level was increased in Technical Specification amendment 100 (Reference 2-3). The applicable Technical Specification (Reference 2-2) change pages are included in this package.
(2) Existing actual operating condition.
(3) Proposed operating condition. Note that the PUR proposed operating condition is bounded by the original UFSAR design condition.
(4) Revised Technical Specification Figure 3.4.1-1, range of acceptable T_{cold} operation is included in this package.
(5) For PUR the steam pressure at zero power (1170 psia) remains unchanged from existing/original conditions.
(6) Original design tube side flow range was 95% to 116% of 82E+06 lb_m/hr or 78 to 95E+06 lb_m/hr. Total RCS flow is simply double tube side flow or 156 to 190E+06 lb_m/hr. The Unit 2 Beginning of Core 1 (BOC1) measured flow was 173.9E+06 lb_m/hr.
(7) The UFSAR Table 1.3-1 does not contain the referenced flows. The turbine and secondary systems have been evaluated for the increased flow and the design is acceptable for operation at PUR conditions.

Section 2.2 References

Reference 2-1 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Reference 2-2 Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Technical Specifications, through Amendment No. 137, December 6, 2001 (Amendment No. 136 not yet implemented).

Reference 2-3 Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 108 to Facility Operating License No. NPF-41, Amendment No. 100 to Facility Operating License No. NPF-51, and Amendment No. 80 to Facility Operating License No. NPF-74, Arizona Public Service Company, et al, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Docket Nos. STN 50-528, STN 50-529, and STN 50-530, dated May 23, 1996.

Section 3 DESIGN TRANSIENTS

This section discusses the Nuclear Steam Supply System (NSSS) and non-NSSS ASME Boiler and Pressure Vessel Code (Reference 3-1) equipment design transients used in stress analyses to support Power Uprate (PUR) conditions. Parameters used for NSSS and non-NSSS design transients are described in UFSAR Section 3.9 (Reference 3-2). Existing design transients were evaluated to ensure their continued applicability.

Section 3.1 Nuclear Steam Supply System Design Transients

Section 3.1.1 Introduction and Background

Design transients used in ASME stress analyses were specified as part of the original design and analyses of the NSSS Structures, Systems, and Components (SSCs). All transients were classified for the SSC operating condition categories identified as normal, upset, emergency, faulted, and testing as defined in the ASME Code, Section III. The original design transients were developed conservatively and have been found to bound actual transients. The original design transients also bound normal operations. All transients are accounted for, and the number and severity of the design transients considered in analysis exceeds those anticipated to occur during the life of the plant.

Section 3.1.2 Input Parameters and Assumptions

The NSSS transients used in stress analyses of the ASME Code class 1 SSCs are outlined in UFSAR Table 3.9.1-1. Comparisons between the parameters of the original NSSS design transients and those of the PUR show equivalent or bounding conditions for most areas. Key parameters such as T_{cold}, T_{hot}, Reactor Coolant System (RCS) flow, and Steam Generator (SG) pressure remain within the band of original/existing design values. The key parameters that are different are reactor power, core flux, RCS volume, and heat transfer capacity of the SGs. The impacts of these differences are discussed below.

Section 3.1.3 Description of Analyses and Evaluation

To ensure that the NSSS SSC design specifications remain adequate following the PUR, the thermal-hydraulic transients in the original design specifications were reviewed to evaluate the impact of the key parameter changes. The focus of the evaluation was to determine if the post-PUR plant response for Operational Basis Earthquakes (OBEs) and Design Basis Events (DBEs) would have more severe consequences than the existing plant response.

The UFSAR Sections 6 and 15 safety analyses performed for the PUR (Section 6) along with the controls systems evaluations performed for normal NSSS design transients (Section 4.3) have shown that the PUR core power does not change the overall dynamic response characteristics of the NSSS so as to invalidate the original design basis transients. The impact of the increase in core flux and neutron fluence on structural SSCs was evaluated and found acceptable. The results are discussed in Section 5.1.2. The larger RCS volume and greater SG heat transfer area will affect the actual plant response for design transients. The loading has changed on NSSS SSCs from original design. The changes in loads are evaluated in Section 5 and meet all applicable code requirements.

Section 3.1.4 Results and Conclusions

The original SSC design specifications are conservative regarding the rate and extent of pressure/temperature changes during DBEs. The specified frequency of occurrence for the DBEs is also conservative. Normal operating plant transients (e.g., plant heatup and cooldown, main and auxiliary spray operation) are limited by administrative controls and/or process limits (e.g., maximum flowrates). The SSC designs bound the new pressure/temperature changes to DBEs associated with PUR.

In conclusion, the original design transients are more limiting than the corresponding limiting calculated transients associated with PUR. Hence, the original SSC design specifications remain bounding and bound the new operating conditions associated with PUR.

Section 3.2 Non-Nuclear Steam Supply System Design Transients

Section 3.2.1 Introduction and Background

As part of the original design and analyses of the non-NSSS SSCs (e.g., pumps, valves, and heat exchangers) design transients (e.g., temperature and pressure transients) are specified in UFSAR Table 3.9-1. These design transients were specified for use in the analyses of the cyclic behavior of the ASME piping systems outside the NSSS scope. To provide the necessary high degree of integrity for the non-NSSS SSCs, the transient parameters selected for SSCs fatigue analyses were based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from changes to various plant operating conditions. The transients selected for use in SSCs fatigue analyses are representative of operating conditions that would be considered to occur during plant operations and are considered sufficiently severe or frequent to be of possible significance to SSCs cyclic behavior. The transients were selected to be conservative representations of transients which, when used as a basis for SSCs fatigue analyses, would provide confidence that the SSC was appropriate for its application over the operating license period of the plant.

Non-NSSS ASME Class 2 and 3 SSCs, supports, and ASME Class 1 piping are designed to an appropriate combination of plant conditions and design loadings. The plant conditions are design, normal, upset, emergency, faulted, and testing conditions. The design loadings are pressure, temperature, dead weight, seismic, and dynamic loads.

Section 3.2.2 Input Parameters and Assumptions

The relationship between selected NSSS parameters and non-NSSS equipment design transients was established. Then the corresponding PUR NSSS parameters were reviewed and the effect on related non-NSSS equipment design transients was determined.

Section 3.2.3 Description of Analyses and Evaluation

A review of the existing non-NSSS equipment design transients determined that the PUR conditions remain within the envelope of the original design transients. These original design transients were selected to be conservative for a wide range of operating parameters and remain bounding.

Section 3.2.4 Results and Conclusions

Based on the above discussion, the original non-NSSS equipment design transients remain bounding following PUR.

Section 3.3 References

Reference 3-1 ASME Code, specifically the 1974 edition of Section III, up to and including the Summer 1974 addenda (original piping).
ASME Code, specifically the 1974 edition of Section III, up to and including the Winter 1975 addenda (tributary piping).
ASME Code, specifically the 1989 edition of Section III (Replacement Steam Generator (RSG) piping).

Reference 3-2 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Section 4 NUCLEAR STEAM SUPPLY SYSTEM

This section discusses the impact of the Power Uprate (PUR) on the functional design requirements and structural integrity of the:

- Nuclear Steam Supply System (NSSS),
- Nuclear Steam Supply System/Balance of Plant Fluid Systems Interfaces, and
- Instrumentation and Controls (I&C) systems.

The NSSS and its Structures, Systems, and Components (SSCs) were verified to be capable of performing their intended design functions at PUR conditions. The control systems associated with the NSSS were evaluated and found acceptable for operation at PUR conditions.

The structural integrity of the NSSS has been evaluated and is discussed in Section 5 of this report.

Section 4.1 Nuclear Steam Supply System Fluid Systems

The following NSSS fluid systems were reviewed:

- the Reactor Coolant System (RCS),
- the Chemical and Volume Control System (CVCS),
- the Safety Injection System (SIS),
- the Shutdown Cooling System (SCS), and
- the Containment Spray System (CSS).

The PUR evaluations considered:

1. the design functions of these systems,
2. the impact of PUR on the design functions, and
3. the ability of each system to carry out its design functions under PUR conditions.

Section 4.1.1 Reactor Coolant System

The RCS is described in UFSAR Chapter 5 (Reference 4-1). The reactor is a Pressurized Water Reactor (PWR) with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary (steam generating) system. The Steam Generators (SGs) provide the interface between the RCS (primary) and the main steam (secondary) system. The SGs are vertical U-tube heat exchangers.

The following SSCs were considered for evaluation:

- Reactor Vessel (RV) including internals, upper head and vent connection,
- pressurizer, including heaters, surge line, ASME Code safety valves, spray piping, spray valves and vent connections,
- RCS piping (hot legs, Reactor Coolant Pump (RCP) suction piping and RCP discharge piping),
- SIS and SCS nozzles and CVCS charging and letdown nozzles,
- primary side of the SGs (nozzles, plenum, and U-tubes), and
- RCPs.

Functional design requirements of the RCS are identified in UFSAR Chapters 4 and 5. Effects of PUR on the functional design requirements were evaluated. The design functional requirements include but are not limited to the following:

- Integrity of the RCS pressure boundary (UFSAR Section 5.2),
- overpressure protection of the RCS pressure boundary during normal operation (UFSAR Appendix 5B),
- overpressure protection of the RCS during low temperature conditions (UFSAR Section 5.2.),
- the primary coolant flow with all four RCPs in operation shall be neither less than the design minimum nor greater than the design maximum (UFSAR Section 4.4),
- plant systems can support a natural circulation cooldown to cold shutdown conditions within a reasonable time without jeopardizing critical safety functions (UFSAR Appendix 5C), and
- the pressurizer is sized to provide sufficient steam volume to allow acceptance of the in-surge resulting from load reduction from any load to any load without the vessel water level reaching the primary safety valves (UFSAR Section 5.4.10).

RCS operating conditions are discussed in Table 2.1-1. A comparison of the PUR operating conditions versus existing and/or design operating condition indicated that:

- normal RCS operating pressure/temperature conditions after PUR are bounded by the original design,
- RCS flow rate is increased after PUR but remains bounded by the minimum and maximum design flow rate, and
- core decay heat load and neutron flux is increased after PUR in comparison with the existing operation.

Effects of PUR on the design functional requirements are as follows:

- Effect of increased neutron flux on the structural integrity of the RCS structures, systems and components are documented in Section 5 of this licensing report. Results of the evaluation demonstrate that existing pressure/temperature limits are bounding for operation at PUR.
- Existing ASME Code safety valves are capable of adequately maintaining the reactor coolant pressure boundary within the ASME Code limits after PUR.

- The most limiting transients for a Low Temperature Overpressure Protection (LTOP) that are described in UFSAR Section 5.2.2 were re-analyzed. Result of the re-analysis indicated that existing LTOP relief valves provide sufficient pressure relief capacity to mitigate the most limiting LTOP events identified. Section 8.13 summarizes the results of the evaluation of LTOP.
- PUR results in higher core decay heat load. This added heat load coupled with the increased heat transfer area of the larger SGs increases the natural circulation driving force. Therefore, inherent natural circulation capacity of the RCS is enhanced by the PUR.
- The NSSS integrated control system analysis was re-performed for PUR. This analysis demonstrated adequacy of the pressurizer and its associated control programs to allow acceptance of in-surge due to load reductions without vessel water level reaching the primary safety valve nozzles at PUR conditions. Section 4.3 summarizes the results of the control systems analysis.

In conclusion, analyses confirmed that the RCS mechanical components are capable of performing their design functions under PUR conditions without modification.

<u>Section 4.1.2</u> <u>Chemical and Volume Control System</u>

As described in UFSAR Section 9.3.4, the CVCS controls the purity, volume, and boric acid content of the reactor coolant.

Affects of PUR on the functional design requirements were evaluated. The design functional requirements include but not limited to the following:

- The CVCS is designed to supply makeup water or accept letdown to support:
  a. 10% step power increases between 15% and 90% of full power,
  b. 10% step power decreases between 100% and 25% of full power, and
  c. ramp changes of ±5% of full power per minute between 15 and 100% power.
- The CVCS Volume Control Tank (VCT) is sized with sufficient capacity to accommodate the inventory change resulting from a 100% to 0% power decrease (reactor trip) with no makeup system operations, assuming that the VCT level is initially in the normal operating level band.
- The CVCS is operated to maintain the reactor coolant chemistry within the limits specified in the EPRI PWR Primary Water Chemistry Guidelines (Reference 4-3), as revised.
- The CVCS provides for boron concentration adjustment in the RCS by feed and bleed. The maximum possible rate of boron dilution is limited, such that the operator has sufficient time to identify and terminate a boron dilution incident before reaching criticality during any refueling operations.
- The CVCS provides an emergency boration capability for recovery of lost Shutdown Margin (SDM). As described in the basis for the Technical Specification, the CVCS can nominally add 1% $\Delta k/k$ of negative reactivity in approximately 4 hours.

- The CVCS design supports the plant capability for conducting a natural circulation cooldown in accordance with the requirements of Branch Technical Position (BTP) RSB 5-1 for a Class 2 plant (see Section 9.10).

A comparison of the CVCS operating conditions after PUR versus existing and/or operating condition indicated that

- CVCS operating pressure and temperature conditions after PUR are bounded by the original design and
- RCS inventory after PUR is increased due to larger SGs.

Effects of PUR on the design functional requirements of the CVCS are as follows:

- makeup water or letdown to support 10% step power increases or decreases is adequately provided by the existing CVCS following operation at PUR,
- existing VCT capacity remains bounding for operation at PUR,
- existing CVCS will take longer to adjust the RCS water chemistry, but the capability to maintain PH and other concentrations within the EPRI limits is maintained for operation at PUR,
- existing CVCS capability to limit boron dilution remains bounding for PUR,
- existing CVCS capability to add a nominal 1% $\Delta k/k$ of negative reactivity in approximately 4 hours is maintained for operation at PUR, and
- existing CVCS capability to support natural circulation cooldown is maintained for operation at PUR. For further details, see Section 9.10.

In conclusion, analyses confirmed that the existing CVCS is capable of performing its design functions under PUR conditions without modification.

<u>Section 4.1.3</u> <u>Emergency Core Cooling System</u>

As described in UFSAR Section 6.3, the Emergency Core Cooling System (ECCS) or SIS is designed to provide core cooling in the unlikely event of a Loss-of-Coolant Accident (LOCA). The ECCS prevents significant alteration of core geometry, precludes fuel melting, limits the cladding metal-water reaction, removes the energy generated in the core, and maintains the core subcritical during the extended period of time following a LOCA. The SIS consists of two active subsystems and one passive subsystem as follows:

- the High Pressure Safety Injection (HPSI) system (active),
- the Low Pressure Safety Injection (LPSI) system (active), and
- the Safety Injection Tanks (SIT) (passive).

The SSCs of the high pressure and low pressure subsystems are arranged in two separate and redundant trains, each of which is capable of performing 100% of the required system design functions. There are four SITs, one tank connected to each RCS cold leg. In the event of a Large Break Loss-of-Coolant Accident (LBLOCA), the

SITs function to reflood the core following blowdown and to provide cooling until the active subsystems begin to inject cooling water.

The adequacy of the SIS is verified by the various safety analyses performed in support of the PUR. The review concluded that the design requirements are acceptable for operation at PUR conditions. An evaluation of the ECCS LOCA analyses is presented in Section 6.1. The evaluation demonstrated that the SIS will perform its required safety functions for PUR without the need for any increase in system performance capability.

In conclusion, the existing SIS system is capable of performing its intended design function after PUR. No system modifications are required for PUR.

Section 4.1.4 Residual Heat Removal System

The functional design requirements of the Residual Heat Removal (RHR) system or the SCS are described in UFSAR Section 5.4.7. The SCS is used in conjunction with the main steam and Auxiliary Feedwater (AFW) systems to reduce the temperature of the Reactor Coolant System (RCS) in post-shutdown periods from normal operating temperature to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the SGs to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to below 350 °F and 400 psia, the SCS is put into operation to reduce the reactor coolant temperature to the refueling temperature and to maintain this temperature during refueling.

The SCS in conjunction with the Essential Cooling Water System (ECWS), the Spray Pond System, AFW system, and the Atmospheric Dump valves (ADVs) are used to cooldown the RCS in various postulated accident conditions.

The capability of the SCS to lower the RCS temperature to 212 °F in approximately 5.5 hours after reactor shutdown with two trains operating is maintained. The capability of the SCS to lower the RCS temperature to 135 °F in ≤ 27.5 hours is maintained. The capability of the SCS to lower temperature to 212 °F within 6.5 hours with one train operating is maintained. The capability of the SCS to achieve cold shutdown with one train within 36 hours as required by the NRC guidance in BTP RSB 5-1 (UFSAR Appendix 5C, Reference 4-4) is maintained. Finally, the SCS remains capable of maintaining refueling temperatures and uniform boron concentration in the RCS under PUR condition.

In conclusion, the existing SCS is capable of performing its intended design functions under PUR conditions without modification.

Section 4.1.5 Containment Heat Removal System

The functional design requirements of the Containment Heat Removal System (CHRS) or the CSS are described in UFSAR Section 6.2.2. The CSS is required to rapidly reduce the containment temperature and pressure following a LOCA or Main Steam Line Break (MSLB) accident, by removing thermal energy from the containment atmosphere. This cooling system also serves to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for leakage of fission products from the containment to the environment.

The CSS consists of two 100% capacity, redundant trains. Each train consists of a CSS pump, one SCS heat exchanger, and a dedicated set of CSS headers inside containment.

Increased predicted peak containment pressure/temperature conditions resulting from PUR can affect containment spray distribution in the post-accident containment. A uniform spray distribution is required for adequate removal of thermal energy and fission products. Spray distribution parameters have been evaluated at the PUR containment pressure/temperature conditions and it was found that:

1. spray coverage in the main spray region is acceptable, and
2. the spray flow to volume ratio for the auxiliary spray region remains unchanged (UFSAR Section 6.5.2.2).

The evaluation of the CSS demonstrated that containment spray distribution/coverage area under PUR conditions is adequate to support proper thermal heat and fission product removal.

The capability of the CSS to reduce containment pressure from peak value to ½ peak value in less than 24 hours and to maintain acceptable containment sump pH level post-RAS is maintained after operation at PUR.

With the implementation of the PUR, CSS system surveillance testing margin will be reduced. The reduction in test margin is due to increased containment pressure in a postulated LOCA condition. Therefore, a modification will be implemented to increase system margin for surveillance testing. Details of this modification are discussed in Section 9.1 of this report.

In conclusion, the CSS is capable of performing its intended design function after PUR. A modification will be performed to increase the surveillance test margin for the system.

Section 4.2 Nuclear Steam Supply System/Balance of Plant Fluid Systems Interfaces

The following Balance of Plant (BOP) fluid systems were reviewed to assess compliance with NSSS/BOP interface guidelines:

- main steam,
- ADVs,
- Condensate (CD) and Feedwater (FW) systems,
- AFW system, and
- SG blowdown/Secondary Chemistry (SC) system.

The safety-related SSCs of the BOP systems include:

- the Main Steam Isolation Valves (MSIVs),
- the MSIV bypass valves,
- the Main Steam Safety Valves (MSSVs)
- the ADVs,
- the Feedwater Isolation Valves (FWIVs), and
- the Seismic Category I portion of the AFW system.

The PUR condition increases the licensed total thermal power to 3990 MW_t. The power increase will result in an increase in steam/FW mass flowrates. These increased values are within the original design specifications for the BOP systems. The increased mass flowrates have been evaluated and found acceptable for the BOP main steam and FW systems.

Section 4.2.1 Main Steam System

The Main Steam system is described in UFSAR 10.3. Main steam system SSCs were evaluated for the PUR condition. The following sections demonstrate the result of these evaluations. Existing design of the main steam system bounds operation at PUR condition. All SSCs with the exception of the MSIV bypass valves will perform their intended design function at the PUR condition.

Section 4.2.1.1 Main Steam Isolation Valves

A description of the MSIVs is found in UFSAR Section 10.3.2.2.2. The MSIVs are located outside the containment in the Main Steam Support Structure (MSSS) downstream of the MSSVs and ADVs. The valve design functions are to prevent the uncontrolled blowdown of more than one SG and to minimize the RCS cooldown and containment pressure to within acceptable limits following a MSLB. The original design requirements are that the MSIVs must be capable of closure within 4.6 seconds of receipt of a Main Steam Isolation Signal (MSIS) against Steam Line Break (SLB). Under PUR conditions, the 4.6-second closure time remains bounding.

Section 4.2.1.2 Main Steam Isolation Valve Bypass Valves

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs before opening the MSIVs. The MSIV bypass valves perform their design function at no-load conditions. As with the MSIVs, the MSIV bypass valves are required to close upon receipt of a MSIS. The MSIS allowable value is raised from 890 to 955 psia for PUR. The MSIV bypass valves were evaluated at the higher MSIS value of 955 psia and will require modification for the increased pressures (See Section 9.1).

Section 4.2.1.3 Main Steam Safety Valves

The MSSVs are described in UFSAR Section 10.3.2.2.3 and Section 5.4.13.2. The MSSVs are located outside containment in the MSSS. There are 5 MSSVs per main steam line for a total of 20 valves (10 per each steam generator). MSSV set pressures remain at existing values for operation at PUR condition. Total design relief capacity of the 20 MSSVs is 19.53 E+06 lb_m/hr. This relief capacity is greater than the total main steam flow at PUR condition. Existing MSSVs will perform their intended function of limiting secondary system pressure at or below 110% of the design pressure. However, because of increased licensed power, the maximum allowable percent power levels with one or more MSSVs inoperable, as described in Technical Specification 3.7.1, are impacted. A Technical Specification Change is requested in this licensing submittal.

Section 4.2.1.4 Atmospheric Dump Valves

The ADVs are described in UFSAR Section 10.3.2.2.4. The ADVs are located outside containment in the MSSS upstream of the MSIVs. There are four ADVs (one per main steam line). The ADVs design pressure is 1333 psia and design temperature is 575 °F. The design function of the ADVs is to provide decay heat removal capability and plant cooldown by discharging steam to the atmosphere when the condenser is not available. The ADVs in conjunction with AFW permit the plant to be cooled down from the lowest MSSV setpoint pressure to the point where the SCS can be placed in service.

Each ADV is sized to hold the plant at hot standby while dissipating NSSS and core decay heat or to allow a flow of sufficient steam to maintain a controlled reactor cooldown rate. The existing ADV nitrogen capacity bounds operation at PUR conditions and permits operation for 4 hours at hot standby and the time required to reach SCS entry conditions (in accordance with BTP RSB 5-1).

In the event of a Steam Generator Tube Rupture (SGTR), the ADVs are used to cooldown the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest MSSV setpoint. RCS cooldown and depressurization are required to preclude SG overfill and to terminate radioactive releases to the atmosphere.

The ADVs are acceptable for operation at PUR conditions operation based on the range of NSSS operating conditions.

Section 4.2.2 Condensate and Feedwater System

The CD and FW systems are described in UFSAR Section 10.4.7. The CD and FW systems are required to provide adequate flow to the SGs at PUR conditions. The CD and FW automatically maintain SG water levels during steady state and transient operations.

The major CD and FW SSCs considered in this section are the FWIVs and the FW pumps. The Digital Feedwater Control System (DFWCS) controls the Feedwater Control Valves (FWCVs). The FWCVs are located on the upper (downcomer) and lower (economizer) SG inlet piping. The DFWCS is discussed in Section 4.3.7.4. Other SSCs, such as the FW heaters are discussed in Section 8.3.

Section 4.2.2.1 Containment Feedwater Line Isolation Valves

The FWIVs are described in UFSAR Section 10.4.7.2.1.9. The containment FWIVs are designed to isolate the FW system from the SG in the event of a SLB, Feedwater Line Break (FLB), or LOCA. This isolation capability precludes radioactive release from the containment due to a CD or FW pipe break.

Operation of the FWIVs may cause potentially large dynamic pressure changes and must be considered in the design of the valves and associated piping. The worst case loads occur following a SLB from no load conditions with both FW pumps in service providing maximum flow following the break. The FWIVs are designed to close against a pressure differential of 1875 psi. Since the FW pumps' maximum discharge pressure (deadhead pressure) is 1636 psia, the existing FWIV design bounds any potential fluid dynamics change associated with operation at PUR condition. In addition, the FWIV closure time is not affected by PUR since the resulting PUR differential pressure is bounded by the original system design.

Section 4.2.2.2 Condensate and Feedwater System Pumps

The CD and FW pumps are described in UFSAR Section 10.4.7.2.1. The CD and FW pump head and the FWCV characteristics must ensure adequate flow to the SGs during steady state and transient operation. The CD and FW pump normal flowrates will increase after PUR. CD flowrates will increase by approximately 1% while FW flowrates will increase by approximately 6%. An evaluation of the pump curves shows that this amount of increase is within the capability of the pumps. Other pump characteristics, including Net Positive Suction Head (NPSH) and pump discharge head, were evaluated at various anticipated systems transients. Evaluations indicate the pumps will perform their intended function under anticipated transients at PUR conditions.

Further evaluations of the CD and FW, including the FW and CD pumps, are contained in Section 8.3.

Section 4.2.2.3 Condensate and Feedwater System Conclusions

The evaluations of the CD and FW system at PUR conditions show that the CD and FW systems will perform their intended design function at PUR conditions.

Section 4.2.3 Auxiliary Feedwater System and Condensate Storage Tank

The AFW system with the Condensate Storage Tank (CST) is described in UFSAR Section 10.4.9. These SSCs provide FW to the SGs during unit startup, hot standby, and cooldown operations. It also provides cooling water in response to system transients and accidents (see Section 6 and Section 9.10 for more detail). The critical AFW system parameters (pump head, system flow, standby CST volume, etc.) required for existing power levels were evaluated at the proposed PUR conditions. In all cases, the existing system design parameters were determined to be bounded for operation at PUR conditions.

The AFW pumps are normally aligned to take suction from the CST. Sufficient water must be available to the AFW pumps to mitigate a transient or accident. The CST contains a minimum usable inventory that is sufficient to bring the unit from full load to hot standby conditions assuming a LOP. The CST can hold the plant in hot standby for 4 hours, and then cooldown the RCS to SCS conditions.

For further discussion of the AFW system, see Section 8.2.

Section 4.2.3.1 Auxiliary Feedwater System and Condensate Storage Tank Conclusions

The AFW system and CST was evaluated for operation at PUR conditions and the conclusions are summarized below:

- The minimum flow requirements of the AFW system, dictated by accident analyses of the limiting transients, were re-analyzed. Results demonstrate that the AFW system performance remains acceptable at the PUR operating conditions.
- The analysis performed to demonstrate compliance with BTP RSB 5-1 under PUR conditions. The RSB 5-1 analysis is described in Section 9.10. The analysis confirmed that the existing CST inventory bounds operation at PUR conditions.

Section 4.2.4 Secondary Chemistry and Steam Generator Blowdown Systems

The SC and SG blowdown systems are described in UFSAR Section 10.4.6 and 10.4.8. The SG blowdown system is used in conjunction with the SC chemical addition system to control the chemical composition of the SG water within specified limits. The blowdown system also controls the buildup of solids in the SG water.

The blowdown flow rates required during operation are based on maintaining chemistry control specifications to control the buildup of solids. These requirements are not impacted by PUR. The capability of the SG blowdown system at PUR conditions is evaluated as part of Section 8.11, and shown acceptable. The SC/water chemistry program is evaluated in Section 8.11.2. The sizing of the SC system has been evaluated and found acceptable for operation at PUR conditions.

Section 4.3 Instrumentation and Controls

Section 4.3.1 Introduction

UFSAR Chapter 7 discusses I&C systems. The impact on the NSSS control systems for the PUR was evaluated. This section summarizes the control system transient analyses performed for uprate and the associated evaluation results.

To ensure that the revised NSSS control systems will provide an acceptable plant response at PUR conditions, the standard NSSS control systems design basis transients were analyzed using the existing control system evaluation code. The anticipated transients include:

1. power ramps up and down at varying rates and from different power levels,
2. 10% step power changes,
3. loss of a FW pump,
4. reactor trip, and
5. turbine trips from several power levels and different plant configurations.

The existing control system evaluation code is a best-estimate power plant simulation tool that analyzes the thermal-hydraulic response of the NSSS. The results demonstrated an acceptable plant response during the analyzed transients. The criteria used to evaluate the performance of the control systems were as follows:

1. For transients of greater than 15% power, the control systems response maintains plant process parameters within their operating band and minimizes process overshoot or undershoot as much as practical.
2. With all control systems in the automatic mode, none of the following design basis NSSS maneuvering transients will cause a reactor trip:
    a. steady state operations below 15% power,
    b. 1% per minute turbine load ramps up or down below 15% power,
    c. steady state operations between 15% and 100% power using economizer FWCVs,
    d. 5% per minute turbine load ramps and reductions between 15% and 100% power,
    e. 10% turbine load steps between 15% and 100% power, and
    f. loss of one of two operating FW pumps.
3. There will be no sustained process oscillations during any steady state or design basis maneuvering operations.

Various control system will be tuned during PUR implementation. Components are tuned at the system level to provide the appropriate responses as specified. The setpoints were determined by a computer modeling code, which is a best estimate simulation code.

The results demonstrated acceptable control system performance. Adequate margin to Engineered Safety Features (ESF) setpoints (e.g., pressurizer low pressure, Low Steam Generator Pressure (LSGP), and SG high level) were maintained. Margins between the design setpoint, limiting setpoint, and analytical limit were evaluated. The simulations demonstrated that the PUR increases in pressure and temperature were adequately mitigated by the control system.

The sizing of the major NSSS control system SSCs (e.g., Turbine Bypass Valves (TBVs), PSVs and heaters, etc.) was evaluated, and the results demonstrated that the installed capacities of these SSCs are adequate for operation at PUR conditions.

Section 4.3.2 Reactor Protection System

As described in UFSAR Section 7.2, the reactor protection system (RPS) consists of sensors, calculators, logic, and other equipment necessary to monitor selected NSSS conditions and to effect reliable and rapid reactor shutdown (reactor trip), if any or a combination of the monitored conditions approach specified limiting safety system settings. The RPS analytical trip setpoints credited in the transient analyses for PUR are listed in Table 6.3-3.

With the exception of low SG pressure, the RPS inputs and design have been evaluated and found acceptable for operation at PUR conditions. The larger SGs and greater power output will result in a higher SG operating pressure. The LSGP setpoint, currently 890 psia, will be changed to 955 psia with the implementation of the PUR per Section 9.1.

Section 4.3.3 Engineered Safety Feature Systems

As described in UFSAR Section 7.3, the Engineered Safety Features Actuation System (ESFAS) consists of the following:

- BOP-ESFAS
- NSSS-ESFAS

Section 4.3.3.1 Balance of Plant Engineered Safety Feature Actuation Signal Setpoints and Regulatory Guide 1.97 Instrumentation

The BOP-ESFAS contains devices and circuitry that generate the following signals when monitored variables reach levels that indicate conditions requiring protective action:

- Fuel Building Essential Ventilation Actuation Signal (FBEVAS)
- Containment Purge Isolation Actuation Signal (CPIAS)
- Control Room Essential Filtration Actuation Signal (CREFAS)
- Control Room Ventilation Isolation Actuation Signal (CRVIAS)

These actuation signals automatically actuate the following ESF systems:

- Fuel building essential ventilation system
- Containment purge isolation system
- Control room essential ventilation system

The BOP-ESFAS setpoints and Regulatory Guide 1.97 (Reference 4-2) instrumentation were reviewed. The conclusions of the review are as follows:

1. There are no changes to the required setpoints and response times.
2. Effects of environmental changes on BOP-ESFAS and Regulatory Guide 1.97 instrumentation results in no changes in instrument setpoints due to increased:

- radiation levels,
- pressures, or
- temperatures.

3. Changes to NSSS and BOP pressure and temperature during transients are discussed in Section 6 of this submittal.
4. No changes required to actuation setpoints for radiation monitors.
5. No changes required for safe shutdown and post-accident (Regulatory Guide 1.97) instrumentation located inside containment. Changes in post-accident instrument uncertainties for impact to the operator decision points were reviewed and revised as required in accordance with PVNGS procedures.

The evaluation concluded that BOP-ESFAS instrumentation is acceptable for operation at PUR conditions.

Section 4.3.3.2 Reactor Trip System/Nuclear Steam Supply System Engineered Safety Feature Actuation System Setpoints

The NSSS-ESFAS contains devices and circuitry that generate the following signals when monitored variables reach levels that indicate conditions requiring protective action:

- Safety Injection Actuation Signal (SIAS),
- Containment Isolation Actuation Signal (CIAS),
- Recirculation Actuation Signal (RAS),
- Containment Spray Actuation Signal (CSAS),
- Main Steam Isolation Signal (MSIS),
- Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1), and
- Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2).

The PPS LSGP setpoint initiates trip functions in the RPS and the ESFAS.

The RPS LSGP trip provides an emergency shutdown of the reactor to protect the core and the RCS pressure boundary. The trip prevents excess heat removal due to a FW system malfunction that could cause excessive thermal cycling of the SG. The trip is also needed to augment the DNBR and Local Power Density (LPD) trips in providing protection against violation of the SAFDLs during an excess load transient initiated from low power.

The NSSS-ESFAS LSGP trip provides those functions required to prevent the release of significant amounts of radioactive material to the environment in the event of a primary or secondary pressure boundary rupture. Specifically, the NSSS-ESFAS trip results in a MSIS that in turn rapidly closes the MSIVs, Main Feedwater Isolation Valves (MFIVs), and the isolation valves for the SG blowdown lines to isolate the system.

The change to the LSGP setpoint will allow the system to function under the new operating conditions. The new LSGP setpoint will maintain an adequate margin of safety. There is no impact to NSSS-ESFAS performance due to PUR. NSSS-ESFAS performance is acceptable for operation at PUR conditions. No other setpoint changes are required for NSSS-ESFAS subsystems.

Section 4.3.4 Systems Required For Safe Shutdown

As described in UFSAR Section 7.4, the following systems are required for safe shutdown of the reactor:

- AFW (discussed in Section 4.2.3),
- ADVs (discussed in Section 4.2.1.4),
- SCS (discussed in Section 4.1.4),
- CVCS, boron addition portion (discussed in Section 4.1.2), and
- CST (discussed in Section 4.2.3).

The following auxiliary support systems are also required to function.

- Essential Spray Pond System (ESPS) (discussed in Section 8.16),
- ECWS (discussed in Section 8.9.4),
- onsite power system, including the Emergency Diesel Generators (EDGs) (discussed in Section 8.14), and
- Heating, Ventilating, and Air Conditioning (HVAC) systems (discussed in Section 8.10.1).

Each of these systems has been evaluated and found acceptable for operation at PUR conditions.

Section 4.3.5 Safety-Related Display Instrumentation

The safety-related display instrumentation located in the control room is described in UFSAR Section 7.5. This instrumentation monitors conditions in the reactor, the RCS, containment, and safety-related process systems, throughout all operating conditions of the plant.

The control room instrumentation and impact of PUR on operators is discussed in Section 9.11. The safety-related display instrumentation has been evaluated and found acceptable for operation at PUR conditions.

Section 4.3.6 All Other Instrumentation Systems Required For Safety

The SCS suction line valve interlocks and the SIT isolation valve interlocks are described in UFSAR 7.6. The SCS is discussed in Section 4.1.4 the SIS is discussed in Section 4.1.3.

The SCS suction line valve interlocks and the SIT isolation valve interlocks have been evaluated and found acceptable for operation at PUR conditions.

Section 4.3.7 Control Systems Not Required for Safety

Section 4.3.7.1 Reactor Regulating System

As described in UFSAR Section 7.7.1.1.1, the Reactor Regulating System (RRS) is used to automatically adjust reactor power and reactor coolant temperature to follow turbine load transients within established limits. The RRS compares a calculated desired RCS reference temperature (T_{ref}) with actual RCS average temperature (T_{ave}). The resulting temperature difference determines if the Control Element Assemblies (CEAs) can be held in position, or the rate of speed at which the CEAs are to be inserted or withdrawn. The output of the RRS is sent to the Control Element Drive Mechanism Control System (CEDMCS).

T_{ave} will increase under PUR conditions. This effect was evaluated in the control system evaluation to verify that the system control design functions were bounded by PUR. The control system evaluation found that the RRS is acceptable for operation at PUR conditions.

Section 4.3.7.2 Pressurizer Pressure Control System

The Pressurizer Pressure Control System (PPCS) is described in UFSAR Section 7.7.1.1.2. The PPCS maintains RCS pressure within specified limits by regulating pressurizer heaters and spray valves. The PPCS provides a mechanism for automatic and manual control of pressurizer pressure at the desired setpoint (2250 psia) during steady state plant operation. If system pressure deviates from the pressure setpoint, heaters are energized to raise pressure to the setpoint or spray is initiated to reduce pressure to the setpoint. The system provides wide range pressure control during plant heatup and cooldown, and equalizes pressurizer and RCS loop boron concentration by continuous circulation through the spray header.

There are no PPCS changes required because of PUR. The control system evaluation demonstrated that the system would maintain its operational parameters with the existing settings.

Section 4.3.7.3 Pressurizer Level Control System

The Pressurizer Level Control System (PLCS) is described in UFSAR Section 7.7.1.1.3. PLCS minimizes changes in RCS coolant inventory by controlling the charging pumps and letdown control valves in the CVCS.

During normal operations the pressurizer level is programmed as a function of RCS T_{ave} in order to minimize charging and letdown flow requirements. T_{ave} goes through a level setpoint program and the setpoint program signal is compared to the actual level signal. The level setpoint program will be adjusted to allow for the increase in T_{ave} because of PUR. The PLCS performance is acceptable at PUR conditions.

Section 4.3.7.4 Digital Feedwater Control System

The DFWCS is described in UFSAR Section 7.7.1.1.4. The DFWCS is based on a two-mode control strategy, a single element and a three-element control mode. At low power levels, the DFWCS is designed to automatically control the SG downcomer water level in a single-element mode. The DFWCS performs dynamic compensation on the level signal to generate an output signal indicative of the required FW flow. The output signal is used to generate the downcomer valve position demand signal. When in this control mode, the economizer valve will be closed and the pump speed setpoint will be at its minimum value. SG level will be controlled during 1% per minute turbine load ramps in this mode (assuming that all other control systems are operating in automatic).

As described in the UFSAR, the DFWCS is designed to automatically control the SG downcomer water level at higher power levels in a three-element mode. The three-element mode continuously solves the SG mass balance equation to keep the FW input equal to the steam flow output. The level measurement acts as a trim on this mass balance and assures that the level is reset to its proper setpoint value following any system disturbances. Thus, the three elements are level, FW flow and main steam flow. The gain and reset control settings are adaptively adjusted by reactor power and FW temperature to control the response for the "shrink/swell" phenomenon. SG level will be controlled during the following conditions (assuming that all other control systems are operating in automatic):

1. steady state operations,
2. 5% per minute turbine load ramps between 15 and 100% NSSS power,
3. 10% turbine load steps,
4. loss of one of two operating FW pumps, and
5. large instantaneous load rejections.

The existing RCS T_{ave} of the RCS and FW flowrate will increase for PUR conditions. These effects were evaluated with the control system evaluation code so that the system control design functions were maintained. The control system evaluation found the DFWCS acceptable.

Section 4.3.7.4.1 Steam Generator Water Level Control System

The SG level indicating instrumentation includes four wide range level transmitters and six narrow range level transmitters per SG. The wide range transmitters provide input to the Plant Protection System (PPS), plant computer, Qualified Safety Parameter Display System (QSPDS), control room indication, and the remote shutdown panel. Four of the narrow range transmitters provide input to the PPS, plant computer, Emergency Response Facilities Data Acquisition, and Display System (ERFDADS) and control room indication. The remaining two narrow range transmitters provide input to the DFWCS, plant computer, ERFDADS, control room indication, and the plant annunciator.

Evaluation of the SG water level control system demonstrated that operating level of the SGs is adequately maintained after PUR. There are no changes needed for the SG level controls and the system is acceptable for operation at PUR conditions.

Section 4.3.7.5 Steam Bypass Control System

The Steam Bypass Control System (SBCS) is described in UFSAR Section 7.1.1.5. The turbine bypass system consists primarily of the TBVs and the SBCS. The SBCS controls the positioning of the TBVs, through which steam is bypassed around the turbine into the unit condenser, with exception of two valves that dump steam to atmosphere. These two valves are the last to open and first to close during steam bypass operation.

The SBCS under PUR conditions provides a means for controlling NSSS thermal conditions during heatup, cooldown and after unit trips by the accommodation of load rejections, and other conditions that result in NSSS/secondary power mismatches. By using the SBCS in conjunction with the Reactor Power Cutback System (RPCS) and the RRS, the TBVs and condenser capacities can accommodate turbine load rejections without lifting the main steam or pressurizer safety valves or tripping the reactor. The SBCS is acceptable for operation at PUR conditions.

Section 4.3.7.6 Reactor Power Cutback System

UFSAR Section 7.7.1.1.6 describes operation of the RPCS. The RPCS is a control system that responds to large load rejections or the failure of either main FW pump by dropping pre-selected Control Element Assemblies (CEAs) and initiating other necessary control actions to obtain a rapid reduction in reactor power. This rapid power cutback capability permits the reactor to remain critical following a load rejection.

Evaluation of the RPCS demonstrated that operating parameters are adequately maintained after PUR. There are no changes needed for the RPCS and the system is acceptable for operation at PUR conditions.

Section 4.3.7.7 Boron Control System

As described in UFSAR Section 7.7.1.1.7, RCS boron control is accomplished by dilution and boration via the CVCS. Refer to Section 4.1.2 for an evaluation of the CVCS.

The evaluation concluded that the dilution and boration of the CVCS is acceptable for operation at PUR conditions.

Section 4.3.7.8 Loose Parts Monitoring System

As described in UFSAR Section 7.7.1.1.8, the Loose Parts Monitoring System (LPMS) is designed to detect and record signals resulting from impacts occurring within the RCS.

There are no LPMS changes required because of PUR. Evaluation concluded that the LPMS is acceptable for operation at PUR conditions.

Section 4.3.7.9 In-Core Instrumentation System

As described in UFSAR Section 7.7.1.1.9, the In-Core Instrumentation (ICI) system is used to monitor the core power distribution. Structural evaluation of the ICIs is included in Section 5.1.1.5 and Section 5.3.3.

Evaluation of the ICIs demonstrated that operating parameters are adequately maintained after PUR. There are no changes needed to the ICIs and the system is acceptable for operation at PUR conditions.

Section 4.3.7.10 Excore Neutron Flux Monitoring System (Non-Safety Channels)

As described in UFSAR Section 7.7.1.1.10, the ex-core neutron flux monitoring system includes neutron detectors located around the reactor core and signal conditioning equipment located in the control room area.

Evaluation of the ex-core neutron flux monitoring system demonstrated that operating parameters are adequately maintained after PUR. There are no changes needed to the ex-core neutron flux monitoring system and the system is acceptable for operation at PUR conditions.

Section 4.3.7.11 Boron Dilution Alarm System

As described in UFSAR Section 7.7.1.1.11, the Boron Dilution Alarm System (BDAS) utilizes the startup channel nuclear instrumentation signals to detect a possible inadvertent boron dilution event while in Modes 3-6.

Evaluation of the BDAS demonstrated that operating parameters are adequately maintained after PUR. There are no changes needed to the BDAS and the system is acceptable for operation at PUR conditions.

Section 4.4 References

Reference 4-1 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Reference 4-2 Regulatory Guide 1.97, Revision 2, December 1975, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident.”

Reference 4-3 EPRI PWR Secondary Water Chemistry Guidelines - Revision 5 (TR-102134-R5).
EPRI PWR Primary Water Chemistry Guidelines Volumes 1 and 2 - Revision 4 (TR-105714-V1RA and TR-105714-V2RA).

Reference 4-4 U. S. Nuclear Regulatory Commission Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

Section 5 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

Evaluations were performed for the Nuclear Steam Supply System (NSSS) to determine the stress and fatigue usage effects of operation at the Power Uprate (PUR) conditions through the remaining plant operating license. The design specifications, engineering calculations, design reports, and other technical documentation that was evaluated with respect to PUR included the effects of the Replacement Steam Generators (RSGs).

Note that these PUR pressure/temperature values are within the existing operating process parameters. The RCS and secondary piping and Structure, System, and Component (SSC) design pressures and temperatures bound this license amendment request.

Section 5.1 Structural Evaluations of the Reactor Coolant System

The general procedure for the structural integrity evaluations for the Reactor Coolant System (RCS) SSCs is as follows:

1. New loads were compared to the original design basis loads at the critical SSC locations. Whenever the new loads were clearly bounded by the existing design basis loads, the Analyses of Record (AOR) were considered to be bounding for RSG and PUR conditions.
2. If the existing stress margins were not adequate to bound the stresses associated with the increased loads, new stresses were determined according to appropriate ASME Code requirements and compared to applicable allowable limits (Reference 5-4 and Reference 5-6).

The following terms are used throughout the remainder of this Section and when used in the context of ASME Code stress analysis, are defined as:

- design conditions: primary stress due to design loads,
- normal conditions (Service Level A): primary - plus secondary stresses, peak stresses, and fatigue usage factors,
- test conditions: primary stresses due to specified test condition loads as applicable,
- upset conditions (Service Level B): primary stresses due to loads that include the effects of normal conditions and Operational Basis Earthquake (OBE), and
- faulted conditions (Service Level D): primary stresses due to loads that include the effects of normal conditions, Safe Shutdown Earthquake (SSE), and pipe break.

Section 5.1.1 Reactor Vessel Structural Evaluation

Reactor Vessel (RV) stress and fatigue usage effects were determined by re-analyzing the RCS at the PUR conditions with the larger Steam Generators (SGs). The new analyses considered the effects of increased dead weight loads, thermal effects, seismic loads using Regulatory Guide 1.61 (Reference 5-11) damping values, and design basis pipe breaks. Regulatory Guide 1.61 allows new higher damping resulting in decreased OBE loads when compared to those used in the AOR. Design basis transients in the AOR that include the effects of the most limiting pipe break locations for the range of stress intensities and fatigue factors, were reviewed. The re-analysis concluded the AOR for stress intensities and fatigue factors are bounding for PUR conditions.

Results of these structural analyses were included in the development of new external loadings on various RCS structures including the RV and its supports. The calculated loads were also used to generate upset and faulted condition RV stress intensities. These stress intensities were compared to the existing design basis results of the AOR and found to comply with the ASME Code.

The evaluation of the RV and RV support columns demonstrates that the RV and its supports are acceptable for plant operation at uprated power conditions and new SGs.

Section 5.1.1.1 Closure Head Flange Region

The RV design specification was updated to provide, a set of faulted loads on the closure head flange, vessel flange ledge, and keyways corresponding to PUR conditions. Faulted condition loads are defined as the combination of SSE and pipe break loads in the design specification. In all cases except one, the SSE loads applicable before PUR bound these faulted loads. The exception was the vertical faulted load on the vessel closure head and vessel flange region that was larger than the vertical SSE load that had been evaluated for the faulted event in the AOR. The analysis with the new faulted loads demonstrated that the total vertical load on the closure head due to operating pressure, dead weight and thermal loads, and revised faulted loads remained less than the closure head stud preload. Therefore, the faulted loads are not a limiting load condition for the vessel closure studs.

The evaluation of the revised faulted loads on the closure head and vessel flange region was performed using the methodology of the AOR. The maximum primary local membrane stress due to faulted loads was determined to be less than the allowable value. Therefore, the closure head and vessel flange region design was demonstrated to be valid for operation at PUR conditions.

Section 5.1.1.2 Reactor Vessel Inlet and Outlet Nozzles

The design specification for PUR provides new increased dead weight plus thermal loads and new decreased OBE loads. The faulted loads do not change. The increase in dead weight plus thermal loads was offset by the decrease in the OBE loads. Therefore, the design condition and faulted condition evaluations documented in the AOR remain applicable to the inlet and outlet nozzles following PUR.

The original fatigue evaluations for these nozzles were, however, affected by revised normal operating (dead weight and thermal) loads. The results of the revised fatigue evaluation for PUR conditions indicate the calculated usage factor is below the ASME

Code allowable value. Therefore, the inlet and outlet nozzle designs remain acceptable for operation at PUR conditions.

Section 5.1.1.3 Reactor Vessel Nozzle Supports

As was the case for the nozzles, increases in dead weight and thermal loads was offset by decreases in OBE loads and AOR faulted loads were bounding for PUR conditions. It was demonstrated that the previous design basis fatigue evaluation for these subcomponents remains bounding for operation at PUR conditions.

Section 5.1.1.4 Control Element Drive Mechanism Nozzles

The Control Element Drive Mechanisms (CEDMs) and the CEDM nozzles were evaluated for the effects of revised OBE and faulted loading conditions. However, only the CEDM nozzles are considered a part of the RV. The CEDMs are addressed in Section 5.3.1 of this report.

The new analysis demonstrates that the OBE and faulted loads on the CEDM nozzles following PUR are less than the maximum allowed loads that were evaluated in the AOR. Therefore, the AOR remains bounding for the PUR.

Section 5.1.1.5 In-Core Instrumentation Nozzles

The In-Core Instrumentation (ICI) and ICI nozzles were evaluated for the effects of revised faulted loading conditions on the ICI nozzles, flanges, and assemblies. However, only the ICI nozzles are considered to be part of the RV. The ICI tubes are addressed in Section 5.3.3 of this report.

The new analysis demonstrates that the new faulted loads on the ICI nozzles are bounded by the previous design basis. Therefore, the stress analysis performed in the AOR remains bounding following PUR.

Section 5.1.1.6 Reactor Vessel Support Columns

An enveloping set of stresses for the RV support columns, both before and after PUR, were compared to allowable ASME Code limits. The maximum calculated stresses are all less than the allowable limits. Therefore, the existing RV support column design is acceptable for operation at PUR conditions.

Section 5.1.2 Reactor Vessel Integrity

RV integrity is impacted by any changes in plant parameters including the effects of neutron fluence levels (see Section 7.5), RCS temperature, or pressure/temperature transients. The most critical area, in terms of RV integrity, is the beltline region of the RV. Therefore, the changes in neutron fluence resulting from the PUR were evaluated to determine the impact on RV integrity.

The evaluation shows that the heating rates, pressure/temperature transients, and neutron fluence estimates that were used to represent operation at 3800 MW_t bound the values at the PUR power level of 3990 MW_t. The neutron fluence projections on the RV for the PUR power level will not adversely affect RV integrity AOR (i.e., pressure/temperature limits and Pressurized Thermal Shock (PTS) screening limits) for operation at 3990 MW_t. Therefore, operation at PUR condition will have no detrimental impact on the RV integrity.

Section 5.2 Reactor Vessel Internals

The following loading conditions are considered in the design of the Reactor Vessel Internals (RVIs):

1. Normal Operating Temperature (NOT) differences,
2. Normal Operating Pressure (NOP) differences,
3. flow loads,
4. weights, reactions, and superimposed loads,
5. vibration loads,
6. shock loads (including OBE and SSE),
7. anticipated transient loadings not requiring forced shutdown,
8. handling loads (not combined with other loads above), and
9. loads resulting from postulated LOCAs.

The RVIs interface was assessed and the structural integrity of the RVIs was not adversely affected for PUR conditions. Section 5.2.3 evaluates the RVI components. In addition, thermal-hydraulic analyses were required to determine plant specific core bypass flows, pressure drops, and upper head temperatures in order to provide input to the LOCA and non-LOCA safety analyses, as well as to the NSSS performance evaluations. These issues are addressed in the following sections.

Section 5.2.1 Thermal/Hydraulic System Evaluations

Section 5.2.1.1 System Pressure Losses

A key area in evaluation of core performance is the determination of hydraulic behavior of RCS flow within the RVIs, i.e., RV pressure drops, core bypass flows, RV fluid temperatures, and hydraulic lift forces. The analyses for PUR concluded that the RVI component loads calculated at the mechanical design conditions of the original AOR are still bounding. The core pressure drop reflecting fuel assembly design and pressure loss coefficient correlations are less than those of the original fuel assembly in the AORs. Thus, no updates on the pressure and flow distribution within the RVI are required. The RV system pressure losses at PUR conditions are bounded by the AOR.

Section 5.2.1.2 Core Bypass Flow Analysis

Core bypass flow is the total amount of RCS flow bypassing the core region and is, therefore, not considered effective in the core heat transfer process. The AOR conservatively presents a design core bypass flow limit of 3% of the total RV flow. This value is used in thermal margin calculations. A lower bounding value of 2% is used in the AOR hydraulic load calculations since the higher core flow results in higher core pressure drops and, therefore, higher uplift and differential pressure loads.

The PUR analysis calculates a core bypass flow of 2.3%. Thus, the core bypass flow value under PUR conditions is appropriate relative to the upper and lower bound values. The total core bypass flow values (with uncertainties) at PUR conditions were determined to remain within the range of 2 to 3% used in the AOR. Therefore, the AOR remains bounding for operation at PUR conditions.

Section 5.2.1.3 Hydraulic Lift Forces

An evaluation of the AOR was performed to determine hydraulic lift forces on the various RVI components so that the RVI assembly would remain seated and stable.

This evaluation included analysis of hydraulic forces of a Reactor Coolant Pump (RCP) start. Before an RCP may be started, the secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG is ≤ 100°F above each of the RCS cold leg temperatures. Satisfying this bounding condition will preclude a large pressure surge in the RCS when the RCP is started.

The results of the hydraulic forces evaluation demonstrated that, with the PUR RCS conditions, the RVI assembly will remain seated and stable. Therefore, the AOR remains bounding for operation at PUR conditions.

Section 5.2.1.4 Reactor Trip Performance Evaluation

The Control Element Assemblies (CEAs) and CEDMs, including Reed Switch Position Transmitters (RSPTs) that indicate the positions of the CEAs, are designed to function during and after all normal plant transients/AOOs. The reactor trip criterion under these conditions is a maximum permissible CEA drop time for 90% insertion of 4 seconds (Reference 5-2, UFSAR, Section 3.9). Other criteria for CEA ability to scram under accident and SSE conditions involve fuel assembly structural integrity requirements (UFSAR, Section 4.2). These criteria can be summarized as follows:

- If the equivalent diameter pipe break for a LOCA does not exceed 0.5 ft^2; fuel assembly deformation shall be limited to a value that allows satisfactory insertion of the CEAs.
- If the equivalent diameter pipe break for a LOCA exceeds 0.5 ft^2; fuel assembly deformation shall be limited so that the fuel is maintained in a coolable array. CEA insertion is not required.

- During a SSE, fuel assembly deflections are such that permanent deformations are limited to a value allowing the CEAs to scram.

Related criteria for CEDMs and RSPTs ability to scram are based on maximum permissible CEDM deflection and curvature. CEDM evaluations were performed for PUR conditions, and indicated that for even the most severe accident case analysis results, the normal plant transient permissible drop time of 4 seconds is achievable for the CEAs. The analysis demonstrated that for the most severe seismic and LOCA conditions the above three criteria are met. See Section 5.3.1 for further discussion of these related evaluations. The RSPTs have been found adequate by comparing the maximum predicted CEDM deflections to those experienced in the original Seismic Qualification Test.

In addition to addressing the above criteria, the structural integrity of the CEAs was evaluated to demonstrate acceptable reactor trip performance. The parameters used to evaluate the CEA structural integrity were based on the 16 x 16 fuel assemblies. The analyses accounted for the fuel parameters associated with both the existing power level and the power level at PUR conditions.

The evaluation concluded that the existing 4-second maximum allowable drop time is maintained, based on CEA performance. Therefore, the CEA performance is acceptable for operation at PUR conditions.

<u>Section 5.2.1.5</u> <u>Control Element Assembly Structural Integrity</u>

With the introduction of the PUR, all reactor vessel components were reanalyzed. The objective was to assess the impact arising from possible changes in system dynamic response, due to differences in the mass distribution of the overall reactor-SG system, including different flowrates and temperatures.

CEA structural integrity during seismic and LOCA conditions depends on the design characteristics of the CEAs as well as those SSCs that comprise adjacent support structures and guide paths. Evaluations of adjacent support structures or other SSCs that comprise the guide paths for the CEAs are addressed in the RVI evaluations (see Section 5.2.3).

The CEA structural analysis used comparisons of overall fuel assembly response to the seismic and LOCA events in order to evaluate whether the AOR results remained bounding for CEA structural integrity. The assessment of the PUR effect concentrated on the guide tubes since those components, through their required deflected shapes, drive the results for the CEAs, i.e., lower guide tube stresses relate to lower component and CEA stresses. Therefore, the assessment of the CEA structural integrity under PUR conditions was based upon the results for the guide tubes in the fuel assembly evaluation.

Maximum stress intensities occur in the CEA fingers when the fuel guide tubes forcibly deflect them during postulated OBE, SSE, and LOCA conditions. The lateral deflections of the guide tubes and the resulting dynamic lateral loads and stresses under the PUR conditions are enveloped by the existing values in the AOR in all cases. It is therefore concluded that the CEA fingers within the guide tubes will also experience loading and stresses no more severe than those that have already been evaluated in the AOR. Therefore, the CEAs structural integrity is maintained for PUR.

Section 5.2.2 Mechanical System Evaluation

Since APS takes credit for Leak-Before-Break (LBB) methodology applied to the primary loop, the LOCA reanalysis of the reactor pressure vessel system for postulated ruptures of the primary loop piping is not required (Reference 5-1). The next limiting primary side breaks that required consideration were the branch line breaks in the:

1. Pressurizer surge line,
2. Safety Injection (SI) lines,
3. Shutdown Cooling (SCS) lines,
4. charging line, and
5. letdown line.

The bounding Branch Line Pipe Break (BLPB) among the NSSS and Balance of Plant (BOP) breaks analyzed for PUR conditions are:

1. Feedwater Line Breaks (FWLBs) - 122.7 in^2,
2. Main Steam Line Breaks (MSLBs) (terminal end and intermediate) - 510.71 in^2,
3. SCS line break - 136.3 in^2,
4. Surge line break - 81.9 in^2, and
5. SI line breaks (terminal end and intermediate) - 103.8 in^2.

For the AOR, the bounding primary side breaks (surge line, SI line, and SCS line) were analyzed along with the BOP breaks MSLB and FWLB.

The mechanical response of the RCS, subjected to branch line breaks, is determined in several steps. Initially, pressure changes were calculated in the annulus between the RV shell and RVI resulting from the rapid depressurization that occurs during a pipe break. These pressure time responses were converted into a set of applied loads acting on the RV shell and the RVI. The other loads resulting from the primary side LOCAs, the pipe tension release and jet impingement forces acting at the break location, and SG subcompartment pressurization forces on the major NSSS SSCs were also determined. Some secondary side pipe breaks (e.g., FWLBs) also produce SG subcompartment pressurization forces that are externally applied to the major NSSS SSCs.

These forces were then applied to a mathematical model of the reactor coolant loop that includes a model of the RVI. The results of this analysis were used to assess the structural integrity of the reactor coolant loop and component supports. In addition to system loads and motions, motions at the interface between the RV and the RVI models were determined.

The resulting RV motions were applied to a mathematical model of the RVIs, including fuel representation. The analysis of this RVI model produced LOCA motions at the core support plate that were then applied to a non-linear representation of the fuel assembly rows and the core shroud. The resulting loads, first for the RVIs and later for the fuel assemblies, were then evaluated as part of the component faulted load evaluations and compared with ASME Code and/or other design criteria. Thus, the effects of pipe break upon the RCS loop, the RV and RVIs, and the fuel were evaluated.

Section 5.2.2.1 Loss-of-Coolant Loads

As noted above, the LOCA loads applied to the pressure vessel system consist of:

1. RVI hydraulic loads (vertical and horizontal),
2. reactor coolant loop mechanical loads (vertical and horizontal), and
3. pressure loads acting on the RV core support barrel annulus.

From a structural standpoint, RVI hydraulic loads have a negligible effect on the RV shell and RV supports. Therefore, the shell was analyzed for reactor coolant loop mechanical loads, and the differential pressure loads discussed above. The RVI were analyzed for hydraulic and pressure loads, and the effects of reactor coolant loop mechanical loads. The motions developed at the RV/RVI interface represent the reactor loop mechanical effects. All loads were first determined individually, and then applied to the non-linear mathematical models in a combined, time-history manner.

With respect to the RVI, the severity of a postulated break is related to two factors:

- the distance from the RV to the break location, and
- the break size.

Pipe breaks farther away from the RV are less severe because the pressure wave attenuates as it propagates toward the RV. With the implementation of LBB, the controlling branch line breaks, such as the pressurizer surge line break, replace the previously analyzed RCS main line breaks, such as the RV inlet and outlet nozzle breaks. These smaller branch line breaks are less severe than the main line breaks, both in terms of distance from the RV, and break size.

The pressure loads acting on the RVI were calculated from the CEFLASH-4B computer code (Reference 5-8). The results of these analyses were then used as part of the input loads for analyses of the RCS structural model, which includes models of the RVI and fuel. Pressure differences in the RV were converted to loads and applied to this fuel model. The RCS structural model is analyzed with the ANSYS code (Reference 5-10). Interface motions determined in the RCS structural analysis are used as part of the input to a model of the RVI, which is analyzed with the CESHOCK code (Reference 5-12).

The CEFLASH-4B output was generated for the SI, SCS, and surge nozzle breaks. In addition to the terminal end breaks, intermediate breaks of the SI lines also required consideration. An enveloping set of loads was created from these results and used as input to the RVI structural integrity analyses.

LOCA evaluations demonstrate that the new RCS conditions will not adversely affect the response of the RVIs due to LOCA excitations.

Section 5.2.2.2 Flow Induced Vibrations

Flow-induced vibrations of pressurized water RVIs have been analyzed for PUR conditions. The objective of this analysis was to show the structural integrity and reliability of RVI components.

UFSAR Section 3.9.2.3.1 details flow-induced vibration of the RVIs components during normal operation that can be characterized as a forced response to both deterministic (periodic and transient) and random pressure fluctuations in the coolant. Methods have been developed to predict the various SSCs of the hydraulic forcing function and the response of the RVIs to such excitation.

For the PUR conditions, the flow-induced vibration loads on the RVIs did not increase since the RCS flow remains within the range of the original design. Therefore, the previous analyses of the RVIs remain bounding, and the structural integrity of the RVIs remains acceptable for flow-induced vibration.

Vibration evaluations demonstrate that the new RCS conditions will not adversely affect the response of the RVIs systems and components to flow induced vibrations.

Section 5.2.3 Structural Evaluation of Reactor Vessel Internal Components

Section 5.2.3.1 Introduction

In addition to supporting the core, a secondary design function of the RVIs assembly is to direct coolant flows within the RV. While directing the primary flow through the core, the RVIs also establish secondary flow paths for cooling the upper regions of the RV and for cooling the RVI's structural components. Some of the parameters influencing the mechanical design of the RVI's lower assembly are the pressure and temperature differentials across its component parts and the flowrate required to remove the heat generated within the structural components due to radiation (e.g., gamma heating, see Section 7.4). For PUR, configuration of the RVIs provides for adequate cooling capability. In addition, the thermal gradients resulting from gamma and alpha heating and core coolant temperature changes are maintained below acceptable limits within and between the various structural SSCs.

Structural evaluations were required to demonstrate that the structural integrity of the reactor components is not adversely affected by the PUR change in RCS conditions and transients and/or by secondary effects of the change on reactor thermal hydraulic or structural performance.

Since the RVIs were designed to ASME Code requirements, a Code compliance stress evaluation was performed. The evaluation of the RVIs considered the impact of revised thermal, hydraulic, seismic, and pipe break input data on Service Level A and B (normal and upset condition) and Service Level D (faulted condition) for the RVI in the AOR.

The process used for this evaluation was to compare the AOR input data to the revised thermal and hydraulic input data identified for the PUR, and recalculate RVIs stresses using the revised input data.

The results of the Service Level A, B, and D analyses meet the stress criteria given in Section III, Division 1 of the ASME Boiler and Pressure Vessel Code for PUR conditions.

<u>Section 5.2.3.2</u> <u>Methodology Used for the Reactor Vessel Internals Structural Evaluations</u>

The RVI components comprise both core support and other internal structures. Fuel support structures include lower fuel support plate, core shroud, and upper guide structure components. Other internal structures include core barrel, CEA shroud and in-core instrumentation support components. All RVI components were evaluated as core support structures in accordance with criteria defined in UFSAR Section 3.9.5.

The revised seismic and LOCA input loads were comprised of loads and moments on RVI components that reflected the larger SGs as well as seismic/LOCA modeling refinements per Regulatory Guide 1.61 and LBB and, as applicable, revised hydraulic input considering PUR and fuel management guidelines.

Critical Service Level A and B stress intensities in the RVI components were identified from the AOR. The revised thermal, hydraulic, and seismic (OBE) input data was compared with that used in the AOR, and stresses were recalculated using the existing methodology in combination with the revised input data. The calculations were performed consistent with the methodology outlined in Section 5.1. All Service Level A and B stress intensities were successfully demonstrated to meet the ASME Code, per UFSAR Section 3.9.5.

The fatigue evaluation of the RVI components was performed based on the fatigue curve provided in the ASME Code, and provides acceptable results in all cases.

Critical Service Level D stress intensities in the RVI components were identified from the previous design basis AOR. The revised seismic (SSE) and pipe break loads were combined and compared with the existing SSE and pipe break loads, and new stresses were determined by calculations consistent with the methodology outlined in Section 5.1. The resulting Service Level D stress intensities were successfully demonstrated to meet the criteria defined in the ASME Code, per UFSAR Section 3.9.5.

Section 5.2.3.3 Additional Components

The holddown ring required re-evaluation to demonstrate adequate capacity to resist the increased hydraulic loads and to consider relaxation. The holddown ring provides a vertical downward force on the core support barrel, Upper Guide Structure (UGS), and associated components. The holddown ring thus maintains these SSCs in a clamped configuration to prevent rocking and sliding of the core support barrel and UGS assemblies. The evaluation demonstrates adequate holddown capacity to resist rocking and sliding.

The flow skirt and its weld attachment to the RV are not affected by PUR since the new downcomer flowrates are bounded by the original design flow conditions.

Section 5.2.3.4 Summary of Conclusions for Reactor Vessel Internal Components

Analyses have been performed to assess the effect of the changes on the RVI components due to PUR. Evaluations demonstrate that the response of the RVIs components to seismic and pipe break excitations will not be adversely affected. Therefore, the structural integrity of the RVIs is maintained.

Section 5.3 Additional Reactor Coolant System Items

The following subsections summarize the evaluations performed for PUR conditions for the:

- CEDMs,
- Heated Junction Thermocouple (HJTC) cables and flanges, and
- ICIs,
- Head Lift Rig (HLR).

Section 5.3.1 Control Element Drive Mechanisms

The ASME Code structural considerations for the pressure boundary components of the CEDMs were considered for PUR conditions. The CEDMs are magnetic jack type drives used to vertically position the CEAs in the core. Each CEDM is capable of withdrawing, inserting, holding, or dropping the associated CEA from any point within its 153-inch stroke in response to operation signals.

The CEDM is designed to function during and after all normal plant transients, AOOs, and Design Basis Accidents (DBAs). The CEA drop time for 90% insertion is 4.0 seconds maximum. The drop time is defined as the interval between the time power is removed from the CEDM coils to the time the CEA has reached 90% of its fully inserted position.

The design and construction of the CEDM pressure housings fulfill the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class 1 vessels. The CEDM pressure housings are part of the reactor coolant pressure boundary, and they are designed to meet stress requirements consistent with those of the RV. The pressure housings are capable of withstanding all normal operating loads that include the steady state and transient operating conditions specified for the RV. Mechanical excitations are also defined and included as a normal operating load.

A review of the evaluations performed to address operation at PUR conditions, of the CEDM pressure boundary components for PUR conditions determined that the ASME Code criteria are being met in all instances.

CEDM evaluations consisted of linear response spectrum analyses, specific to the new configuration for both seismic and BLPB excitations; to calculate response loads in the CEDM nozzles and CEDM components as well as CEDM deflections. A new three-dimensional beam finite element ANSYS 5.5 model (Reference 5-10), with all spatial degrees of freedom, was developed and used for these analyses. This model uses a sufficient number of nodes to accurately represent the dynamic characteristics of the nozzle components, and to provide a detailed load response distribution throughout the CEDM structure.

The same mathematical model was used for dead weight, seismic and BLPB analyses. Seismic and BLPB loads were applied to the CEDMs with the longest and shortest nozzle lengths. The longest nozzle produces conservative results for seismic loads, while the shortest nozzle produces conservative results for BLPB loads. The governing load for the faulted condition was the combined worst seismic load and worst LOCA load determined for different CEDM locations.

Section 5.3.1.1 Control Element Drive Mechanism Evaluations

Calculated response loads for PUR conditions were used to evaluate the CEDM components for the upset condition (dead weight and thermal loading, and operating pressure results plus OBE results) and the faulted condition (dead weight, thermal loading and operating pressure results plus the Square-Root-of-the-Sum-of-the-Squares (SRSS) combination of SSE and BLPB results). The calculated response loads were also used to perform a CEDM operability check for dead weight, thermal loading, and operating pressure plus seismic conditions. Comparing the maximum CEDM deflections due to the combination of SSE and BLPB with those shown acceptable by static test verified the CEDM operability. A maximum calculated combined deflection was used in this comparison for conservatism.

Total CEDM response loads for Service Level B (upset condition) and Service Level D (faulted condition) were determined. The total CEDM dead weight was applied to the CEDM nozzle, but was not included as an axial load in the motor housing, in the upper pressure housing, and in the shroud. The response loads for all CEDM pressure boundary components (excluding the nozzle Service Level A and B response loads) were compared to the limiting loads. All loads were smaller than the limiting loads except for the axial load on the upper pressure housing. However, the excess load is not significant for two reasons:

1. The axial load causes a minimal stress.
2. The sizing analysis for the CEDMs included a conservative dead weight load that covers the difference in the axial load on the upper housing considered above.

This comparison of CEDM responses to limiting loads demonstrated that the calculated response loads for Service Level A, B and D conditions were within the allowable ASME Code limits.

Section 5.3.1.2 Evaluation of Control Element Drive Mechanism Deflections

Maximum calculated absolute deflections at the top of the CEDM are 4.56 inches for OBE and 5.41 inches for the SRSS of SSE and BLPB. Since the gap between adjacent CEDMs is 4.568 inches, the possibility of CEDM impact was evaluated. An analysis was performed for the RCS demonstrating that no impact occurred between adjacent CEDMs during Service Level A, B or D conditions. Applying these results plus a conservative factor to the deflections demonstrated that the maximum relative deflection between the longest CEDMs for any service condition was less than the 4.568-inch gap.

Section 5.3.1.3 Ability to Trip the Reactor

To demonstrate the ability of the CEDMs to trip the reactor for the PUR RCS configuration, two required criteria based on previous testing need to be satisfied:

1. the maximum displacement of the SRSS combination of SSE and BLPB at the top of the CEDM is less than 8 inches and
2. the calculated minimum radius of curvature of the CEDM at the 80-inch elevation was greater than 20.965 inches. The 80-inch elevation is measured from the top of the RV dome, excluding insulation, to the point of interest on the upper pressure housing.

Comparison of results for PUR to these criteria demonstrated that both criteria are satisfied and therefore, that the reactor will retain the ability to trip during seismic and BLPB events.

Section 5.3.1.4 Reed Switch Position Transmitter Operability

The 150-inch RSPT was also seismically qualified by test for a maximum CEDM deflection of 7.85 inches for OBE and 9.5 inches for SSE. Since the maximum CEDM combined calculated deflection due to SSE and BLPB was only 5.41 inches, the RSPT for the PUR configuration was demonstrated to be operable during seismic events and following a BLPB.

Section 5.3.1.5 Conclusions

For PUR, it was demonstrated that CEDM loads due to Service Levels A, B, and D are within the allowable limits. Furthermore, CEDM stresses for Service Levels A and B will not affect CEDM and RV head thermal stresses. For a discussion of CEDM nozzles Alloy 600 material see Section 5.8.

The following was also demonstrated for PUR:

- during seismic, LOCA, and BLPB, the maximum displacement at the top of the CEDM is within acceptable limits,
- the CEAs retain their ability to insert within the 4 second time period, and
- the CEDMs will not be impacted, and the RSPTs remain operable.

Section 5.3.2 Heated Junction Thermocouple Cables and Flange

HJTCs use existing CEDM nozzles. Previous test data of CEDMs were used to determine dynamic amplification factors at the HJTC flange elevation for the combined effects due to SSE and BLPB. The amplification factors corresponding to the natural frequencies of the HJTC flange assembly were applied to the RV closure head seismic excitation applicable to the PUR RCS configuration, and to the RV closure head seismic excitation applicable to the existing RCS configuration. The maximum increase in response of the HJTC flange to BLPB excitations for the existing versus the PUR configuration was determined from RCS pipe break analysis results. The amplified excitation levels of the HJTC flange were then compared to the excitation levels used to qualify the HJTC cables for operability and structural integrity.

Section 5.3.2.1 Heated Junction Thermocouple Cables Evaluation

The first two natural frequencies of the HJTC assembly were calculated to be 8 Hz and 81 Hz. RV closure head seismic spectra values for 2% damping for these two frequencies were obtained for the PUR and existing configurations. Based on amplification factors derived from previous test data and applied to the peak acceleration (g levels), maximum HJTC elevation accelerations due to OBE and SSE excitations at 8 Hz and 81 Hz were obtained for both RCS configurations.

The HJTC cables are string-like structures with relatively low natural frequencies. Analytical results indicate that RV head response spectra peaks due to BLPB occur at high frequencies (>30 Hz). Therefore, HJTC cables do not respond dynamically to BLPB excitations. The faulted condition loadings on HJTC cables therefore consist of only SSE effects.

A comparison of the SRSS g levels to those applied in the HJTC cable seismic qualification tests demonstrates that the required response spectrum values envelop the calculated values with significant margin.

Section 5.3.2.2 Heated Junction Thermocouple Instrumentation Flange Assembly

An evaluation of seismic loads along the length of an active CEDM demonstrated that the seismic loads are reduced at elevations above the motor housing. The seismic loads at the HJTC instrumentation flange assembly (HJTCIFA) are therefore less than similar loads for the upper pressure housing for the following reasons:

1. the HJTCIFA is located above the motor housing,
2. the HJTCIFA does not contain a long upper pressure housing structure,
3. the lowest HJTCIFA natural frequency is higher than that of an active CEDM, and
4. the HJTCIFA natural frequencies do not lie on the peaks of the head spectra.

For BLPB effects, it was demonstrated that the maximum ratio of RV head response spectra at HJTC natural frequencies for the existing and PUR configurations is 12.8%. Through a comparison of calculated loads for the PUR configuration to the existing design basis loads, it was determined that the original HJTC flange design analyses considered higher seismic loads than those calculated at the HJTC flange for the PUR configuration. The comparison also established that the HJTC flange design loads contained sufficient margin to accommodate the BLPB loads for both configurations.

Since the structural integrity of the HJTCIFA has been established for the existing configuration, and since the results of the original HJTC flange design analysis envelop the results for the HJTC flange for the PUR configuration, the integrity of both RCS configurations was confirmed.

Section 5.3.2.3 Conclusions

HJTC cable operability and structural integrity, and HJTC flange assembly structural integrity, were demonstrated for seismic and BLPB excitations for operation at PUR conditions.

Section 5.3.3 In-Core Instrumentation Tubes

The ICI tubes are attached to the lower head of the RV, and are supported at ten guide tube support plates before terminating at the ICI Seal Table. The original analyses included response loads and stress calculations due to:

- dead weight,
- thermal expansion,
- pressure,
- seismic,
- mechanical loading, and
- LOCA in the Main Coolant Loop (MCL) piping.

The AOR demonstrated that stress criteria were met for all loading conditions for the existing configuration. The evaluation method used to reconcile the results for the PUR configuration was to demonstrate that the input loads on the ICI tubes for the PUR configuration are less severe than those of the existing configuration.

The first comparison made was between response spectra. Where the spectra for the PUR configuration were enveloped by the corresponding spectra for the existing configuration, it was concluded that the existing configuration results for that load are applicable to the PUR configuration. Where the spectra for the PUR configuration were not enveloped by the spectra for the existing configuration, a second comparison was made to demonstrate that the response of the ICI tubes to the input load for the PUR configuration is enveloped by the response to the load for the existing configuration.

Section 5.3.3.1 Operating Basis Earthquake Evaluation

Review of the OBE spectra demonstrated that the existing configuration spectra envelop both the PUR configuration RV spectra, and the floor spectra for the X direction (horizontal and parallel to the hot leg) and the Z direction (horizontal and perpendicular to the hot leg). For the vertical (Y) direction, both the PUR configuration RV spectra and the containment basemat spectra exceed the existing configuration spectra at certain frequencies. Since total enveloping of the spectra could not be demonstrated, it was necessary to demonstrate that the response of the ICI tubes to the PUR configuration spectra is enveloped by the response to the existing configuration spectra.

The AOR provided the response for selected locations for each mode and each excitation direction. The response to the PUR configuration spectra was calculated by multiplying each AOR modal response by the ratio of the new to the original modal accelerations. The response for each excitation direction was calculated by combining the modal responses in accordance with Regulatory Guide 1.92 (Reference 5-3). The total response of the ICI tubes was then calculated using the SRSS of the response due to the X, Y, and Z directions. The load results demonstrated that the response of the ICI tubes to the OBE excitation for the PUR configuration is less than the response calculated for the existing configuration. Therefore, all ICI OBE results for the existing configuration are valid for the PUR configuration.

Section 5.3.3.2 Safe Shutdown Earthquake Evaluation

Review of the SSE spectra demonstrated that the original RCS configuration spectra envelop both the PUR configuration RV spectra and the containment basemat spectra for all three directions. Therefore, all ICI SSE results for the existing configuration are valid for the RCS configuration.

Section 5.3.3.3 Branch Line Pipe Break Evaluation

Verified RV shell response spectra due to Main Coolant Line Breaks (MCLBs) for the existing configuration were available for the RV flange at 2% damping. In order to make a one-to-one comparison of spectra, 2% damping envelope spectra of the nine BLPB cases analyzed for the RCS configuration were developed from flange acceleration time history responses. Comparing the BLPB envelope spectra to the Main Coolant Line Break (MCLB) spectra from the AOR demonstrated that the MCLB spectrum envelops the BLPB spectrum in the Y direction, but that for the X and Z directions, there are frequencies at which the BLPB spectra exceeds the MCLB spectra. It was therefore necessary to demonstrate that the response of the ICI tubes to the BLPB spectra is less than the response to the MCLB spectra.

In order to do this, the modal ratios of the MCLB-to-OBE and BLPB-to-OBE spectra were determined, and used to perform a comparison of ICI tube responses. Since the

ICI tube locations most affected by the change in RV motion are the ICI nozzles, the response moments at the modeled ICI nozzle were calculated for the envelope BLPB spectra, and for the MCLB spectra, using the previously calculated modal ratios. This method demonstrated that the response load on the ICI nozzles and tubes due to the PUR configuration BLPB loads is less than the response load calculated in the AOR for MCLBs. Therefore, all ICI pipe break results for the existing configuration are valid for the PUR configuration.

Section 5.3.3.4 Conclusions

Since the original design loads on the ICI tubes due to seismic excitation and pipe breaks were shown to be valid for the PUR configuration, and since the other original design loadings on the ICI tubes are unchanged, it was concluded that all original design results bound the results for the PUR configuration. Since all ICI tube stress criteria were previously met for the original design, all stress criteria are also met for the ICI tubes in the PUR configuration.

Section 5.3.4 Head Lift Rig

The permanent HLR structure was analyzed for PUR seismic and pipe break-input excitations to generate the HLR response loads. The evaluation demonstrated that HLR stresses for all Service Level conditions are within the respective allowable limits, thereby demonstrating the structural integrity of the HLR.

Specific areas of evaluation are described below:

- The stress in the HLR lifting lugs was evaluated and determined to be acceptable.
- Stresses in the seismic plate were calculated, and the structural integrity of the seismic plate was determined to be acceptable.
- Stresses in the platform beams were calculated. It was demonstrated that the combined SSE and pipe break bending stresses are within the allowable, thereby justifying the structural integrity of the platform beams.

Section 5.4 Reactor Coolant Loop Major Components and Component Supports

The PUR parameters were reviewed for impact on the existing design for the reactor coolant loop piping, primary equipment (i.e., RV, RCP and SG) nozzles, and primary equipment supports.

Sets of input parameters that were used in the evaluation of the reactor coolant piping and supports consist of NOP/NOT (dead weight, pressure, and thermal expansion), thermal design transients, seismic OBE/SSE, and BLPB type LOCA parameters. PUR resulted in a new set of externally applied loads (NOP/NOT, seismic and LOCA) for the main coolant loop piping, the major RCS SSCs, and the major RCS SSC supports. Thermal design transients for PUR are bounded by the currently specified transients and did not affect the structural evaluations of the RCS. The PUR analyses used the same methods and criteria as those of the existing AOR.

The NSSS performance parameters define the various temperature conditions associated with the potential full power operating conditions of the plant. All of the thermal expansion, seismic, and LOCA analyses performed on the NSSS piping systems reflect full power operating conditions. The thermal design transients are used in the evaluation of piping fatigue. The RV dynamic LOCA motions associated with defined postulated break cases were used in the analyses of the remainder of the RCS structures. Because of the implementation of LBB methodology, postulated guillotine breaks in the primary loop piping have been replaced with postulated guillotine breaks at the loop branch connections for the largest ASME Class 1 branch lines. The revised basis also includes the largest branch line pipe breaks on the secondary side.

The evaluation for the reactor coolant piping, the primary equipment nozzles and the primary equipment supports indicated that all SSCs meet appropriate allowables. Operation at PUR conditions has no adverse effect on the ability of these SSCs to perform their safety function. Evaluations for the remaining RCS SSCs are discussed in the following sections.

Section 5.4.1 Reactor Coolant System - Leak-Before-Break

The original structural design basis required consideration of dynamic effects resulting from main coolant loop pipe breaks. Therefore, protective measures for such breaks (i.e., pipe whip restraints) were incorporated into the design. In the 1980's, the use of LBB criteria was applied to RCS piping based on fracture toughness technology and material toughness. The application of LBB to primary loop piping analyses was documented in Reference 5-1.

The RCS was reanalyzed for the PUR conditions, and the new design basis pipe breaks resulting from the application of LBB are now included in the design basis.

Section 5.4.2 Use of ANSYS Computer Code

The existing structural AOR for RCS SSCs was performed using a group of computer codes. Specifically, the codes were MEC-21, STRUDL DYNAL, and CE-DAGS (dynamic analysis of gapped structure). MEC-21 was used for static analysis. STRUDL DYNAL performed dynamic seismic analyses. STRUDL DYNAL and CE-DAGS were used to dynamically analyze the structure for pipe breaks.

The new RCS seismic and pipe break structural analyses were accomplished using the ANSYS code (Reference 5-10). ANSYS performs all of the STRUDL DYNAL, MEC-21, and CE-DAGS functions. This is a change in modeling of structural loads and motions. ANSYS is a general purpose finite element program with structural and heat transfer capabilities. Performing a seismic analysis of the original RCS mathematical model and comparing the results (i.e., modal frequencies, loads, and motions) to original design basis results benchmarked the ANSYS code. The results in this benchmark demonstrate the equivalence of ANSYS to STRUDL DYNAL, MEC-21, and CE-DAGS for RCS seismic and pipe break structural analyses. ANSYS is identified in UFSAR Section 3.9 as an approved computer code for RCS SSCs.

Section 5.4.3 Reactor Coolant Model Changes

The analyzed RCS mathematical model reflects incorporation of the RSGs and PUR. The new SGs have a larger diameter lower hemispherical head. The RCS model was modified to consider the changes in the RCS due to the larger diameter SGs.

Section 5.4.4 Reactor Coolant System Main Loop Piping and Tributary Nozzles

The new loads affecting the RCS piping were incorporated in a revision to the RCS piping design specification. The structural evaluations for any SSCs impacted by the piping design specification revision are discussed below. Results of existing configuration design basis analyses that were unaffected by the latest revision of the piping design specification are not presented. It is noted that because of the implementation of LBB, the pipe whip restraints are no longer required. Thus, the pipe whip restraint AOR is no longer applicable.

With the installation of the RSGs, the RCS main coolant loop piping (cold leg) elbows are being replaced due to the associated change in RCS geometry.

The piping and tributary nozzle designs were shown acceptable in accordance with the ASME Code of record, for the remainder of the plant license.

Section 5.4.4.1 Main Loop Piping

The limiting main coolant piping components examined were:

- hot leg pipe,
- hot leg elbow,
- cold leg pipe, and
- cold leg elbow.

Section 5.4.4.1.1 Non-Faulted Conditions

For the cold leg, there is no increase in the non-faulted loads. Therefore, the results from the AOR remain bounding.

The loads contributing to the primary stresses for design conditions are dead weight and OBE. For the hot leg, the dead weight loads increased. The new design loads resulted in primary stress intensities less than ASME Code allowables and is acceptable for operation at PUR conditions.

The thermal loads on the hot leg piping increased from those used in the existing analysis. Subsequent analyses demonstrated that these increases produced a negligible impact on the primary plus secondary stresses and on the fatigue usage factor. Therefore, the AOR for normal conditions remains bounding for PUR.

Section 5.4.4.1.2 Faulted Conditions

The loads contributing to the faulted conditions have increased from those used in the AOR for the cold leg piping. The new accident loads resulted in the maximum primary stress intensity of less than the Code allowable and is acceptable for operation at PUR conditions.

For the hot leg, there is no increase in faulted loads. Therefore, the results from the AOR remain bounding for operation at PUR conditions.

Section 5.4.4.2 Tributary Lines and Nozzles

Section 5.4.4.2.1 Tributary Piping

Tributary piping to the RCS (SI, SDC, CVCS, pressurizer surge line, and pressurizer spray) was structurally analyzed for the effects of thermal, dead weight, seismic, Seismic Anchor Movement (SAM), and LOCA (due to BLPB, MSLB, and FWLB). There are no changes in the dead weight and thermal loads on these piping compared to the AOR. Seismic and LOCA loads were different from those reported in AOR. This difference was due to the change in the dynamic response of the RCS with a heavier SG. The resulting stresses were compared against the stresses in the AOR and it was determined not to affect the fatigue calculations for the AOR. The new structural analyses demonstrate that the tributary piping meets the requirements of the ASME Code (Reference 5-5). The fatigue analyses for the Class 1 portions of these systems are bounded by the AOR.

Section 5.4.4.2.2 Safety Injection Nozzles

The loads on the SI nozzles were revised for PUR. The primary stresses due to the specified loads were recalculated, and the limiting nozzle stress location was found to be the end of the nozzle safe end.

Concerning the maximum moment criteria defined for this nozzle, the total stress remains below the allowable stress.

The new analysis also demonstrated that the revised loads have a negligible impact on the nozzle stresses and, correspondingly, on the fatigue usage factor. Therefore, the AOR for normal conditions remains bounding.

Section 5.4.4.2.3 Surge Line Nozzle

The loads on the surge line nozzle contributing to the primary stresses in the nozzle body are consistent between the design specification and the AOR. Therefore, the AOR for this region of the nozzle remains bounding.

The stresses in the nozzle-pipe juncture are increased due to the effects of the run pipe loads on the nozzle. The addition of these hoop stresses results in maximum primary stress intensities for design and faulted conditions. These stresses compare favorably with the allowable for design and faulted conditions respectively.

Concerning the maximum moment criteria, the total stress remains below the allowable stress.

The SAM OBE loads that contribute to the fatigue evaluation of the surge nozzle are increased from the loads used in the AOR. The analysis performed for PUR conditions shows that these increases have a negligible impact on the nozzle stresses and, correspondingly, on the fatigue usage factor. Therefore, the AOR for normal conditions remains bounding.

Section 5.4.4.2.4 Charging Nozzle

The loads on the charging inlet nozzle were revised in the design specification. The primary stresses due to the specified loads were recalculated. The stresses at the limiting location were determined to be less than the corresponding ASME Code allowable stresses.

The maximum moment criteria total stress remains less than the ASME Code allowable stress.

The OBE loads contributing to the fatigue evaluation of the charging inlet nozzle are increased from the AOR. It was determined that these increases have a negligible impact on the nozzle's OBE load stresses and, correspondingly, on the fatigue usage factor. Therefore, the AOR for normal conditions remains bounding.

Section 5.4.4.2.5 Letdown/Drain Nozzles

The OBE and LOCA loads contributing to the design and faulted condition stresses on the letdown/drain nozzles were revised in the design specification. The primary stresses due to the specified loads were recalculated. The results of the analysis demonstrated that the stresses at the limiting location were less than the corresponding ASME Code allowable stresses.

The OBE loads on the nozzles, as well as the run pipe moment at normal operation conditions were revised. Thus, the AOR for primary plus secondary stresses and the fatigue usage factors were also reviewed. It was determined that increases in OBE loads on these nozzles, as well as the run pipe moment, produce a negligible impact on the nozzle's primary plus secondary and peak stress ranges. Therefore, the AOR for normal conditions, including the calculation of the fatigue usage factor, remains bounding.

The loads contributing to the maximum moment criteria analysis are not affected by the latest revision of the design specification. Therefore, the AOR remains bounding.

Section 5.4.4.2.6 Shutdown Cooling Nozzles

The OBE nozzle loads and the run pipe loads contributing to the design condition stresses on the SCS outlet nozzles were revised in the design specification. The primary stresses for design conditions due to the specified loads were reassessed. The results of the reassessment demonstrate that the calculated stresses are within the allowable limits of the ASME Code.

The primary plus secondary stresses and fatigue usage factors were examined. The conclusion was that the increase in OBE loads has a negligible impact on the nozzle's primary plus secondary, peak stress ranges, and the fatigue usage factor for the nozzle-pipe juncture. Therefore, the AOR for normal conditions in this region remains bounding.

Section 5.4.4.2.7 Spray Nozzles

The seismic and pipe rupture nozzle loads, as well as the run pipe accident loads contributing to the design and faulted condition stresses on the spray nozzles were revised in the design specification. The primary stresses, due to the specified loads, were recalculated and found to be less than the corresponding ASME Code allowable stresses.

As mentioned above, the OBE nozzle loads were revised in the design specification. The pipe run normal operation loads were also revised. Thus, the primary plus secondary stresses and fatigue usage factors were reexamined. It was demonstrated that increases in OBE nozzle loads and in the run pipe moments produce a negligible impact on the nozzle's primary plus secondary and peak stress ranges. Therefore, the AOR for normal conditions, including the calculation of the fatigue usage factor, remains bounding.

Section 5.4.4.2.8 Partial Penetration Nozzles

The partial penetration nozzles evaluated include:

- Resistance Temperature Detector (RTD) nozzles on the hot leg,
- RTD nozzles on the cold leg, and
- Pressure and sampling nozzles on the hot leg.

There was no change in loads contributing to the analysis of the step boundary region of the RTD nozzles. Therefore, the AOR for this region remain bounding.

The normal operation loads for the hot leg and faulted loads for the cold leg were revised in the design specification. The primary stresses due to the specified loads at the weld boundary region were considered. Results of the analysis demonstrate the stresses in all partial penetration nozzles are less than the ASME Code allowable stresses.

As mentioned above, the normal operational loads contributing to the fatigue evaluation of the partial penetration nozzles at the hot leg were revised in the design specification. Thus, the primary plus secondary stresses and fatigue usage factors were reassessed. The reassessment demonstrated that increases in the loads produce a negligible impact on primary plus secondary and peak stress ranges. Therefore, the AOR for normal conditions, including the calculation of the fatigue usage factor, remains bounding.

There are no external LOCA loads that apply to the small-bore nozzles. Thus, no evaluation was required.

Section 5.4.5 Reactor Coolant Pumps

The term RCP refers to the pumps assembly, which consists of the pump casing, pump skirt, motor stand, and all associated subcomponents (i.e., impellers, bearings, seal closure, pump shaft, etc). The RCP motor is mounted on top of the RCP. Therefore, the entire unit is referred to as the pump/motor combination. The RCP and the RCP motor structural evaluations are discussed below:

Section 5.4.5.1 Reactor Coolant Pump Structural Evaluations

The RCPs were designed and analyzed to meet the pump design specifications, and the ASME Code criteria. Since the design transients already considered in the AOR would bound any transients applicable to PUR, the original ASME Code thermal transient analyses performed in the AOR remain bounding. Specifics of the performed structural analyses/assessments due to PUR conditions are discussed below.

The methodology for the RCP assessments was as follows:

1. Initially, previous design basis RCP loads were assessed by:
   - Reviewing the stress margin survey generated for the PUR to determine those pump locations having stress margins below a threshold of 10%.
   - Comparing the new dead weight and thermal, seismic, and faulted loads acting on the RCPs to the loads used as input to the AOR for the RCPs.
2. Those pump locations where new stresses might exceed allowables (i.e., those locations with low stress margins and relatively high loads for PUR) were selected for closer examination. Any location with increases in one or more of the loads that comprised a loading condition was at least considered.
3. Further load reconciliation or stress analysis was performed for the selected locations to demonstrate that any increased loads were acceptable.

Of these locations, the motor stand shell/lower window locations required further examination, due to an increase in the torsional moment. The controlling stress intensity is at the lower window section, due to loads acting at the top of the lower window. This stress intensity was recalculated for the appropriate PUR motor stand section loads, and compared to the allowable limit.

The stress intensity at the critical location was determined to be less than the allowable in the ASME Code.

The higher PUR design basis loads at other RCP locations were reconciled by performing quantitative and qualitative assessments. In these cases, the structural integrity of the RCP was demonstrated without redetermining stresses. Instead, the assessments considered the impact of relative changes in load component magnitudes and, in some cases, the impact of changes in the overall direction of the loading.

In conclusion, the new loads resulting from PUR are considered acceptable for the RCPs, and the pump pressure boundary components were demonstrated to still be within the RCP design specification and the ASME Code.

Section 5.4.5.2 Reactor Coolant Pump Motor Structural Evaluations

The "worst case" loads for the RCP motors, in the form of RCP motor peak accelerations, were calculated for PUR conditions and compared with original design specification values. In addition, three areas where parameter changes affect performance are:

1. continuous operation at the revised hot and cold loop ratings,
2. pump start-up, and
3. loads on thrust bearings.

From a structural integrity analysis standpoint, continuous operation at rated conditions, and loads on the thrust bearings are considered the most critical areas. These critical areas were determined to be acceptable for operation at PUR conditions by demonstrating that the new RCP motor peak accelerations are bounded by the limits defined in the RCP design specification and the ASME Code.

The limits defined in the design specification are curves of peak acceleration versus pump motor elevation. There are both horizontal and vertical direction curves for each type of excitation, OBE, SSE, and LOCA. The PUR calculated maximum accelerations due to earthquake and LOCA at the uppermost elevation on the pump motors were each found to be less than the specified limits. In all cases, the PUR calculated values were less than the design limits by significant margins.

Section 5.5 Steam Generators

The RSGs are being designed and fabricated to operate at PUR conditions. PUR will be implemented during plant startup following SG replacement. The analyses and evaluations for PUR were performed at conditions associated with the RSGs. Generally, the RSGs differ from the Original Steam Generators (OSGs) as follows:

- The number of tubes is increased by 10%.
- Primary and secondary water volumes are increased.
- The RSG dry weight is increased.

- The RSGs are taller, resulting in an increase to the main steam nozzle elevation.
- The upper level indication nozzle tap elevations are higher.
- The main feedwater (FW) nozzle elevation is higher.
- The snubber lugs are at the same elevation but now project from the shell cone.
- A new recirculation nozzle is added.
- A new upper blowdown nozzle is added.
- SG tube material is changed from Inconel 600 to Inconel 690.
- New computer programs used for RSG stress analyses are discussed in Section 5.5.2 of this report.

The RSGs are being designed and analyzed in accordance with the ASME Code for structural acceptability, thermal-hydraulic, U-bend fatigue, tube degradation, tube plugging, and repair requirements (Reference 5-6). The maximum allowable RSG tube wall degradation has been analyzed according to the requirements of Regulatory Guide 1.121 (Reference 5-9).

<u>Section 5.5.1</u> <u>Steam Generator Supports</u>

The existing SG upper supports, with any necessary readjustments at the building interface and the sliding base and at the pedestal flange/sliding base interface will be re-used with the RSGs.

<u>Section 5.5.1.1</u> <u>Steam Generator Upper Supports</u>

The upper part of the SG is restrained by means of shear keys and snubbers that are attached to the refueling canal walls and secondary shield walls. Loads on the SG snubber lugs and upper Z keys on the SG snubber hardware (i.e., the snubber arrangements), and on the portions of the upper support structure were calculated for the PUR conditions.

Subsequent to this, an assessment of the snubber lugs, snubber arrangements (i.e., the snubbers, link, lever, lever bracket, and pins), upper Z keys, and upper Z key supporting structures (i.e., expansion plates, shims and wall brackets) was performed which considered the impact of PUR on normal and accident loadings that are used to size the SG upper supports. The conclusion of this assessment was that the original calculated design loads on the existing SG upper supports bound the new RCS configuration. The design specifications for the SG snubbers and snubber hardware were revised to acknowledge the review of the impact of PUR on these components.

The existing design load on each SG snubber lug is 3,600 kips. The snubber/lever arrangements, which attach to the SG snubber lugs, were qualified for loads of up to 2,134 kips each. The corresponding calculated load on this equipment for PUR conditions was 1,430 kips.

The existing design loads for the SG upper Z keys and the supporting structures as defined above are 2,330 kips and 900 kips, respectively. The corresponding calculated loads for PUR conditions are 1,941 kips and 481 kips, respectively.

Therefore, the existing limiting loads bound the new loads on the upper support systems. This confirms that the existing snubber/lever systems and upper Z key expansion plates, shims and wall brackets are more than adequate for PUR conditions. Thus, all of the design load requirements for the SG upper support system subcomponents will continue to be met after PUR. The design specifications for the SG snubbers and snubber hardware were revised to acknowledge the review of the impact of the PUR condition on these components.

<u>Section 5.5.1.2 Steam Generator Sliding Base and Skirt Studs</u>

The design specification for the sliding base was revised to acknowledge the review of the impact of PUR on the component. Since the sliding base is essentially at ambient temperature, PUR does not affect thermal stresses. The changes in sliding base support loads were considered relative to the original analysis. As a result of load distribution changes, the stresses in the baseplate and a number of sliding base subcomponents were reanalyzed, using the methodology defined in the original analysis. The reanalysis remains consistent with the requirements of the ASME Code.

The SG skirt has been redesigned and, therefore, an assessment of the effects of the skirt redesign and the PUR loads on the SG skirt bolting was required.

An axisymmetric finite element model of the new skirt geometry and skirt studs was used to analyze the new loads due to PUR. The finite element analysis considered a maximum initial preload of 530 kips/stud during installation and a minimum residual preload of 320 kips/stud that remains in effect thereafter for all loading conditions. Effects of interactions at the critical interfaces (i.e., skirt/sliding base bolt circle and skirt/stud/nut) were accounted for with boundary conditions. Additionally, an analysis was performed to determine the maximum allowable bolt loads, based on the allowable stress limits for the upset and accident conditions. The results of the bolting analysis found bolt loads for the critical bolts to be less than the allowable.

Based on the results of the analyses and assessments, it is concluded that all design changes made to the skirt and skirt flange bolting, including the addition of shims, are acceptable for operation at PUR conditions.

<u>Section 5.5.2 Computer Codes Used in Steam Generator Structural Analysis</u>

The SG thermal, mechanical, and vibration analyses have been performed with ANSYS computer program (Reference 5-10). In conjunction with ANSYS, the following special purpose post-processing procedures have been used to compute the stress intensity ranges and fatigue usage factors required by the ASME Code: RANGE, RANGETS and FATIGTS. These codes are written in Fortran 77 language and are a post-processing procedure according to the ASME Section III rules of stress outputs as generated by ANSYS. Specifically:

- RANGE takes the six stress components ($P_m + P_b$) given as output by ANSYS and computes the max stress intensity range based on Subsection NB, Article NB-3222-2 of ASME III;
- RANGETS takes the stress components ($P_m + P_b$) given as output by ANSYS in the perforated region of a tubesheet and computes the stress intensity range according to the rules outlined in ASME III, Appendix A-8000 Article A8142-2-a;
- FATIGTS takes the total stress components as calculated by ANSYS and evaluates the peak stresses in a perforated plate according to the rules of ASME III, Appendix A-8000, Article A8142-2-b; then computes the fatigue usage factor based on the rules of Subsection NB, Article NB-3222-4 of ASME III.

All these procedures have been verified and bounded for PUR by means of comparison with hand-calculated results.

<u>Section 5.6</u> <u>Pressurizer</u>

The design requirements for this component remain bounding for operation at PUR conditions. Therefore, the AOR remains bounding.

<u>Section 5.7</u> <u>Nuclear Steam Supply System Auxiliary Equipment</u>

NSSS auxiliary equipment was evaluated for impact due to PUR. The existing configuration parameters, manufacturing/quality assurance requirements, and transients for the auxiliary valves, auxiliary pumps, and heat exchangers are defined in the equipment design specifications. No tank volume, pump hydraulics, or heat exchanger performance were affected by the PUR.

Based on the PUR parameters for the NSSS auxiliary equipment, the following conclusions were made:

- Comparison of the maximum system operating temperatures and pressures at PUR conditions and original system design conditions shows that all maximum operating temperatures and pressures for systems within the NSSS scope are bounded by the existing design basis. Since the design of all auxiliary equipment was consistent with the system design requirements, the auxiliary equipment is acceptable for the maximum operating temperatures and pressures resulting from the PUR.
- In addition, the auxiliary equipment thermal and hydraulic transients resulting from the PUR are bounded by the original design parameters. Therefore, the auxiliary equipment designs remain acceptable for the thermal and hydraulic transients.

<u>Section 5.8</u> <u>Alloy 600 Material Evaluation</u>

An issue facing all Pressurized Water Reactor (PWR) owners is the Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 material exposed to primary coolant, especially at elevated temperatures. The SGs are constructed with Alloy 690 tubes that are more resistant to most types of corrosion, but there are other applications of Alloy 600 in the RCS. These include the CEDM nozzles, in-core instrumentation nozzles and vent lines in the RV head; sampling and RTD nozzles in the RCS piping; and heater sleeves in the pressurizer. The pressurizer instrument nozzles have been replaced with Alloy 690. APS is currently in the process of replacing the sampling and RTD nozzles in the RCS hot leg piping with Alloy 690 nozzles.

APS is aggressively managing Alloy 600 issues. The pressurizer will continue to operate at the same temperatures and thus the potential for PWSCC for the pressurizer heater sleeves will not increase with PUR. The cold leg nozzles are not affected since Alloy 600 nozzles in PWR applications become more susceptible to PWSCC at temperatures exceeding 600 °F. The increase in susceptibility for the Alloy 600 components in the hot legs and the RV head is acceptable. APS has been following industry recommendations through the EPRI Materials Reliability Project (MRP). Visual inspections are performed per Generic Letter 88-05 (Reference 5-7).

Section 5.9 References

Reference 5-1 Letter A. E. Scherer of Combustion Engineering, Inc., to Darrell G. Eisenhut, Docket No. STN 50-470F, December 23, 1983, with enclosure, "Leak Before Break Evaluation of the Main Loop Piping of a CE Reactor Coolant System," Revision 1, November 1983.

Reference 5-2 PVNGS Units 1, 2, and 3 Updated Final Safety Analysis Report (UFSAR), Revision 11, dated June 2001.

Reference 5-3 Regulatory Guide 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Revision 1, February 1976.

Reference 5-4 ASME Code, specifically the 1974 edition of Section III, up to and including the Summer 1974 addenda (original piping).

Reference 5-5 ASME Code, specifically the 1974 edition of Section III, up to and including the Winter 1975 addenda (tributary piping).

Reference 5-6 ASME Code, specifically the 1989 edition of Section III (RSG piping).

Reference 5-7 NRC Generic Letter 88-05, dated, 3/17/1988, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.”

Reference 5-8 Combustion Engineering, Inc., “Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel,” CENPD-252-P-A, July 1979.

Reference 5-9 Regulatory Guide 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” Revision 0, August 1976.

Reference 5-10 De Salvo, G. P. and Swanson, J. A., ANSYS-Engineering Analysis System Swanson Analysis Systems, Inc., Elizabeth, Pa., 1972. SW V&V Report MISC-ME-C-274, Revision 0, “Software Verification and Validation Report ANSYS Version 5.5.2 on HP9000/800 Series Machines with the HP-UX 10.20 Operating System,” June 8, 1999.

Reference 5-11 Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 0, October 1973.

Reference 5-12 Appendix B of Report CENPD-42, Revision 1, “Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Applications of Analyses to CE 800 MW_t Class Reactors,” February 1986.

Section 6 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSIS

Section 6.1 Emergency Core Cooling System Performance Analysis

Section 6.1.1 Introduction

The existing ECCS performance Analyses of Record (AOR) were performed for a rated core power of 4070 MW_t (3990 MW_t with a 2% power measurement uncertainty), as described in UFSAR Sections 6.3 and 15.6.5. The AOR was evaluated for the larger Replacement Steam Generators (RSGs) and the results are bounding. ECCS performance was not reanalyzed for Power Uprate (PUR).

The AOR is revised as a part of the application of Reload Process Improvement (RPI). In support of RPI, Section 6.3.3.7 of the UFSAR, a bounding ECCS performance analysis is performed and the plant design data used in the bounding analysis are documented in an ECCS performance analysis comprehensive checklist. Changes to plant design data are evaluated and, if necessary, incorporated into the AOR. All of these revisions are typically implemented under the provisions of 10 CFR Part 50.59.

The ECCS acceptance criteria are contained in 10 CFR Part 50.46(b) (Reference 6-1). To determine the conformance with the criteria, ECCS performance analyses consist of three parts:

- Large Break Loss-of-Coolant Accident (LBLOCA),
- Small Break Loss-of-Coolant Accident (SBLOCA), and
- post-Loss-of-Coolant Accident (LOCA) Long-Term Cooling.

The following sections address these three parts.

Section 6.1.2 Large Break Loss-of-Coolant Accident

The AOR for LBLOCA has a peak cladding temperature of 2174 ^{o}F at a Peak Linear Heat Generation Rate (PLHGR) of 13.1 kW/ft. These results are based on the core power of 4070 MW_t (3990 MW_t with 2% power measurement uncertainty), and are applicable to the PUR.

Section 6.1.3 Small Break Loss-of-Coolant Accident

For SBLOCA ECCS performance the AOR has a peak cladding temperature of 1907 ^{o}F at a PLHGR of 13.5 kW/ft. These results are based on a core power of 4070 MW_t (3990 MW_t with 2% power measurement uncertainty), and are applicable to the PUR.

Section 6.1.4 Post-Loss-of-Coolant Accident Long-Term Cooling

The AOR for post-LOCA Long-Term Cooling ECCS performance is based on a core power of 4070 MW_t (3990 MW_t with 2% power measurement uncertainty), and remains applicable to the PUR.

Section 6.2 Containment Response Analysis

The containment building is the final barrier against the release of significant amounts of radioactive fission products. The containment structure must be capable of withstanding the pressure and temperature conditions resulting from a postulated LOCA and maintain a leaktight barrier. In addition, the structure must be capable of withstanding a Main Steam Line Break (MSLB). The containment response analysis is performed per requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 16 and 50 (Reference 6-1), to demonstrate that the design pressure and temperature conditions for the containment structure are not exceeded. These analyses also define environmental envelopes for mechanical/electrical equipment located within the containment.

Containment is designed to a pressure of 60 psig and maximum liner temperature of 300 °F and is described in detail in UFSAR Section 6.2.1 (Reference 6-13).

Section 6.2.1 Introduction and Background

Containment Structure Accident Conditions. The postulated accidents considered in determining design containment peak pressure (and temperature) and external pressures are summarized in UFSAR Table 6.2.1-1. Containment analyses were performed at 102% of requested licensed power of 3990 MW_t (4070 MW_t core power). For the containment structure peak pressure analysis, it is assumed that each postulated accident is concurrent with the most limiting single active failure in systems required to mitigate the consequence of the accident or to shutdown the plant. In addition, if the postulated accident causes a turbine or reactor trip, Loss of Offsite Power (LOP) is also assumed if the results are limiting. MSLBs are analyzed with offsite power available since secondary side breaks are more severe with no LOP. No two accidents are postulated to occur simultaneously or consecutively. For conservatism, containment leakage is not incorporated into the containment peak pressure analysis. The planned power increase from existing licensed power has an effect on the containment response to a Design Basis Accident (DBA). The following list identifies those changes that have the most impact:

1. The power increase would result in an increase in the RCS average temperature (T_{ave}) and the decay heat, which results in more energy being transferred to the containment via the LOCA break flow.
2. The additional RCS inventory due to the larger SGs increases the mass being transferred to the containment during the RCS LOCA blowdown.
3. The additional SG mass inventory, larger heat transfer area, and a higher secondary operating pressure result in more energy being transferred to the containment for an MSLB.

Section 6.2.2 Loss-of-Coolant Accident Containment Analysis

Section 6.2.2.1 Introduction and Background

A LOCA is characterized by the rapid discharge of the RCS inventory into the containment. The analytical simulation of the LOCA event is initiated from 102% of 3990 MW_t (4070 MW_t) and is characterized by four distinct phases. These are blowdown, reflood, post-reflood, and long-term cooldown. The LOCA Mass and Energy (M&E) and containment response analysis has been performed in accordance with Sections 6.2.1 of the Standard Review Plan (SRP), Reference 6-6.

Section 6.2.2.2 Description of Loss-of-Coolant Accident Containment Analysis

Three break types are investigated,

- the RCS Reactor Coolant Pump (RCP) Double-Ended Discharge Leg Slot Break (DEDLSB),
- the RCS RCP Double-Ended Suction Leg Slot Break (DESLSB), and
- the Double-Ended Hot Leg Slot Break (DEHLSB).

All three-break locations are analyzed assuming both minimum and maximum Safety Injection (SI) pump flows. SI pump flows are shown in Table 6.2-2. Limiting single failure for these analyses is a loss of one train of Containment Spray System (CSS).

The containment pressure/temperature profiles were calculated for each break location using computer program COPATTA (Reference 6-8).

Section 6.2.2.3 Methodology Used for Loss-of-Coolant Accident Containment Analysis

Section 6.2.2.3.1 Loss-of-Coolant Accident Mass and Energy Release Calculations

The blowdown phase of the LOCA was simulated with CEFLASH-4A code (Reference 6-4) and conforms to the requirements of 10 CFR Part 50 Appendix K. The base deck and input data used for the Appendix K ECCS analysis are biased toward conservatively maximizing fuel rod temperature. Some input and nodalization modifications are made to maximize the M&E releases to the containment. Nodalization and input data changes are needed to assure that bounding parameters and models are employed to maximize the mass and energy release to the containment. This approach is consistent with SRP 6.2.1.1.a, which states in part, "…accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis." The reflood and post-reflood phases of the LOCA are simulated using FLOOD3 code.

The FLOOD3 simulation of the reflood and post-reflood phases is only used for the cold leg breaks. The reflood and post-reflood phases are not simulated for hot leg breaks. For hot leg breaks, most of the reflood fluid does not pass through a SG before being released to the containment; hence, in contrast to a cold leg break, there is no physical mechanisms to rapidly remove the residual SG secondary energy during or after reflood.

These three phases comprise the short-term LOCA response for the cold leg breaks. For the hot leg break, the short-term response terminates at the end of the blowdown phase with the CEFLASH-4A code. The M&E release data was utilized to calculate the containment pressure and temperature response with the COPATTA code.

Some of the significant inputs and assumptions for the M&E release calculations are outlined in Table 6.2-1. SI flows are listed in Table 6.2-2.

Section 6.2.2.3.2 Loss-of-Coolant Accident Long-Term Response

The long-term phase of the LOCA completes the transient simulation of this event. In this phase, the analysis accounts for all residual heat, including decay heat, in the primary and secondary systems. CONTRANS2 containment code (Reference 6-7) was run for this long-term phase to calculate the time dependent energy addition due to this residual heat.

Utilizing the long-term, M&E release data as well as RCS residual heat (energy) data at the end of blowdown for hot leg breaks and end of post-reflood for cold leg breaks, the containment pressure and temperature response was calculated.

Section 6.2.2.4 Results of Loss-of-Coolant Accident Containment Analysis

M&E releases were calculated for the LOCA events. This information was utilized as input to calculate the transient containment pressure and temperature response. Figure 6.2-1 provides a comparison plot, of current versus proposed power, time history total energy discharged to containment for the limiting LOCA case, DEDLSB.

The containment pressure and temperature response and sump water temperature response versus time for the DEDLSB are given in Figure 6.2-2 and Figure 6.2-3. The calculated condensing heat transfer coefficient based on Tagami or Uchida correlation as discussed in UFSAR Section 6.2.1 are shown in Figure 6.2-4. Figure 6.2-5 provides the containment liner temperature profile based on COPATTA heat sink output data. Table 6.2-3 presents a summary of pressures and temperatures for all RCS breaks. Based on the results presented in this table, the DEDLSB LOCA with maximum ECCS was identified as the pipe break with the highest peak pressure. The resulting peak pressure is less than 58 psig (73 psia), which remains below the design pressure value of 60 psig. Table 6.2-4 provides a summary of results for this event.

In order to reflect the increased peak containment pressure, this submittal includes a request to change the containment leakage rate test program, Technical Specification 5.5.16. The peak calculated containment internal pressure for the design basis LOCA (P_a) will be changed from 52.0 to 58.0 psig.

Section 6.2.2.4.1 Long-Term Containment Performance

Long-term analyses of the worst case DEDLSB, and the worst-case pump suction leg break were performed to verify the ability of the Containment Heat Removal System (CHRS) to maintain the containment below the design conditions. These evaluations were based upon conservatively assumed performance of the engineered safety features. The CHRS long-term operating mode is assumed to include one CSS train. This analysis shows that within 24 hours the containment pressure is reduced to less than one half of peak containment pressure.

Table 6.2-1
Summary of Initial Condition Parameters for LOCA Analysis

Parameter	Value
Core power (MW_t) (102% of 3990 MW_t)	4070
RCS T_{cold} (°F)	568.2
Maximum RCS pressure (psia)	2325
Minimum RCS flow (lb_m/hr)	161.0E+06
Minimum CSS pump flowrate (gpm)	3500
Maximum Refueling Water Tank (RWT): temperature (°F) inventory (gal)	 120 400,000
Number of SG tubes plugged	Zero
SG nominal initial mass (liquid + steam), (lb_m)	192,591
Initial SG dome pressure (psia)	1051.4
Limiting initial containment parameters temperature (°F) relative humidity (%) pressure (psia)	 120 Zero% 16.7 (14.2+2.5)
Conservative volumetric expansion multiplier due to temperature and pressure (%)	2

Table 6.2-2
Safety Injection Flows

Minimum SI flow per pump as a function of RCS pressure (injection phase)		
RCS cold leg pressure (psig)	High Pressure Safety Injection (HPSI) (gpm)	Low Pressure Safety Injection (LPSI) (gpm)
0	950	3744
50	936	3020
100	921	2025
130	912	667
200	891	0
310	856	0
605	756	0
782	688	0
993	597	0
1199	494	0
1349	405	0
1483	307	0
1581	205	0
1700	2	0
Maximum SI flow per pump as a function of RCS pressure (injection phase)		
RCS cold leg pressure (psig)	HPSI (gpm)	LPSI (gpm)
0	1374	5760
50	1353	5080
100	1331	4280
150	1310	3240
175	1299	2560
200	1288	1720
400	1198	0
600	1105	0
800	1006	0
1000	900	0
1200	785	0
1400	657	0
1600	506	0
1700	416	0
1800	309	0
SI flow per pump (post-Recirculation Actuation Signal (RAS)		
RCS cold leg pressure (psig)	Maximum HPSI (gpm)	Minimum HPSI (gpm)
0	1441	950
50	1421	936

Table 6.2-3
Summary of Containment Pressure and Temperatures for RCS Breaks

LOCA result	DEHLSB 19.2 ft^2 maximum ECCS		DESLSB 9.82 ft^2 maximum ECCS		DESLSB 9.82 ft^2 minimum ECCS		DEDLSB 9.82 ft^2 maximum ECCS		DEDLSB 9.82 ft^2 minimum ECCS	
	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t
Peak Pres., psia	62.43	67.06	62.8	68.65	62.85	68.31	66.11	72.05	65.16	69.62
Peak Pres., psig	48.23	52.86	48.6	54.45	48.65	54.11	51.91	57.85	50.96	55.92
Peak Vapor Temp., °F	271.01	276.6	291.93	300.55	292.4	299.64	293.5	308.41	293	304.07
Time to Reach Peak Pres., sec.	20.4	11.4	92.6	95	92.6	90.4	253.6	204	454	340

Table 6.2-4
Summary of Results for the DEDLSB LOCA

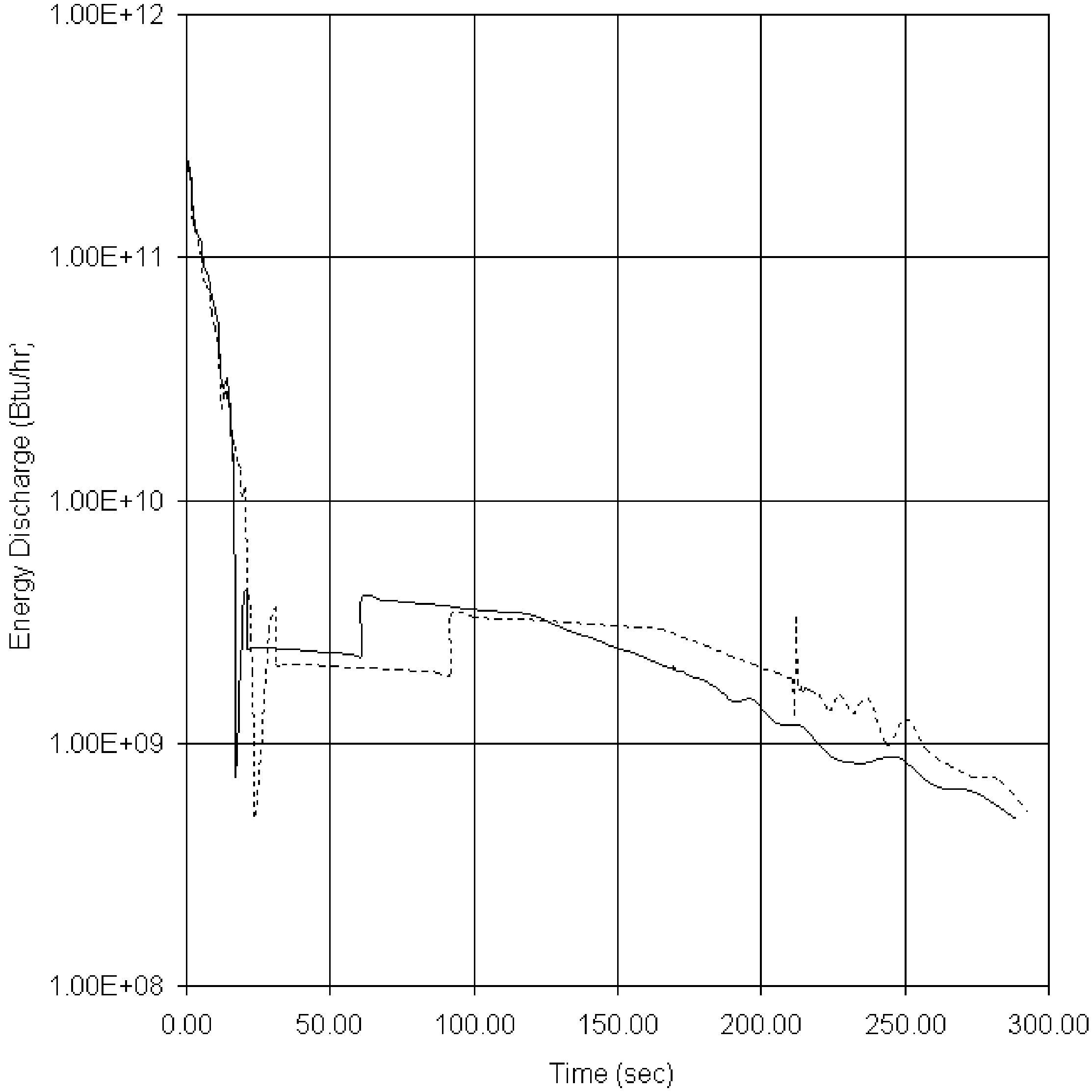
Title	3876 MW_t	3990 MW_t
Peak pressure, psia	66.11	72.05
Peak pressure, psig	51.91	57.85
Elapse time to peak pressure, sec	253.6	204
Peak containment vapor temperature, °F	293.5	308.41
Elapse time to peak temperature, sec	93	95
Peak sump temperature, °F	251	242.47
Elapse time to peak sump temperature, sec	17,500	15,500
Peak liner temperature, °F	<293.5(1)	270.97
Elapse time to peak liner temperature, sec	N/A	460
Initiation of recirculation, sec	1438	1403.13
Pressure at 24 hr, psia	38.81	35.59

Note: (1) Containment vapor temperature.

Section 6.2.2.5 Conclusion

The impact of PUR on LOCA was reviewed. It is concluded that the peak containment liner temperature is predicted to be 270.97 °F. This temperature is below the containment building liner design peak temperature of 300 °F.

Figure 6.2-1
DEDLSB LOCA RCS Energy Discharge Comparison



RCS RCP DEDLS Break With Maximum ECCS Flow
Break Type: DEDLS
Pipe ID: 30 inches
Break area: 9.82 square feet
ECCS: Maximum

—— 4070 MWt ------ 3954 MWt

Figure 6.2-2
DEDLSB LOCA Pressure Profile for Containment Design

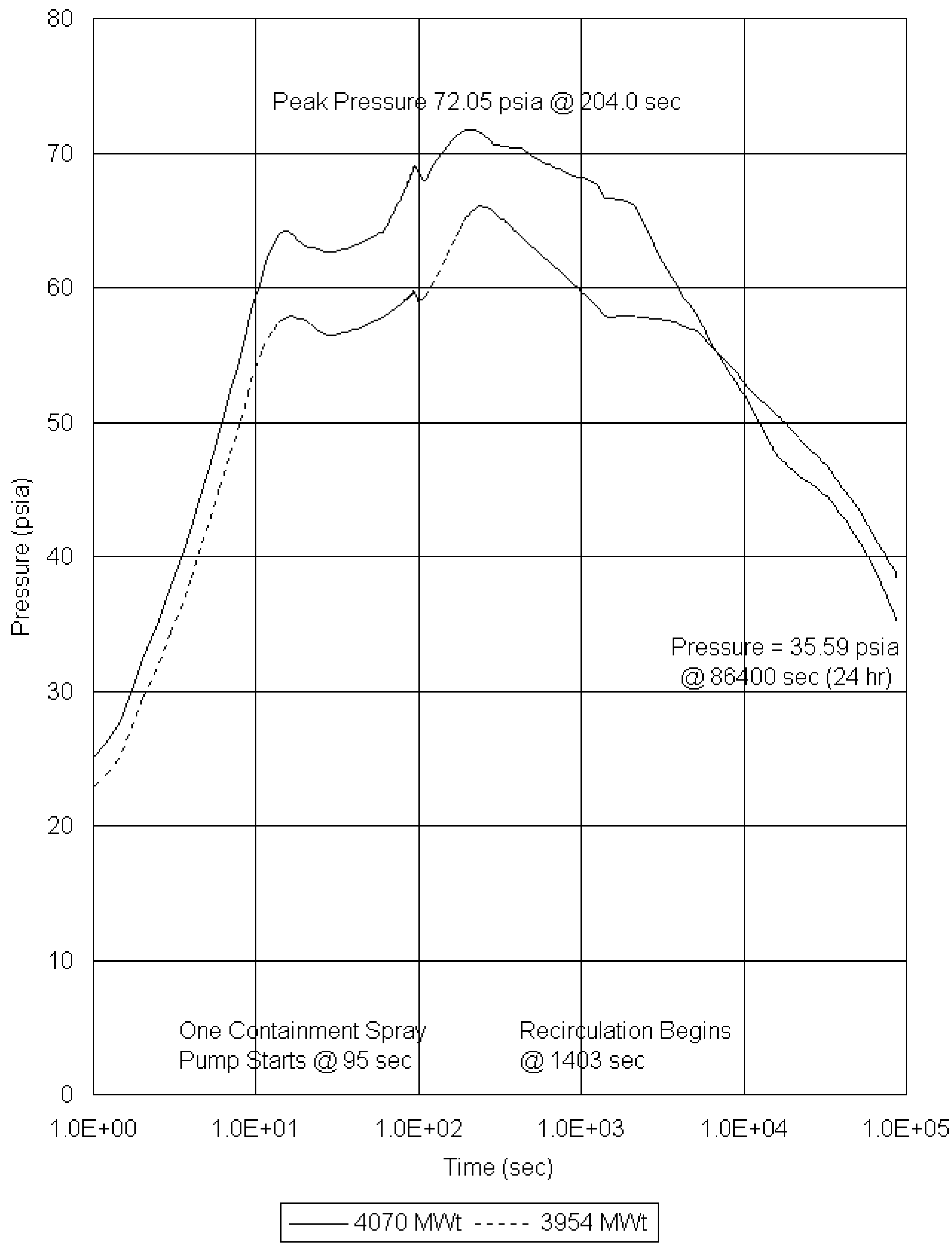


Figure 6.2-3
DEDLSB LOCA Temperature Profiles for Containment Design

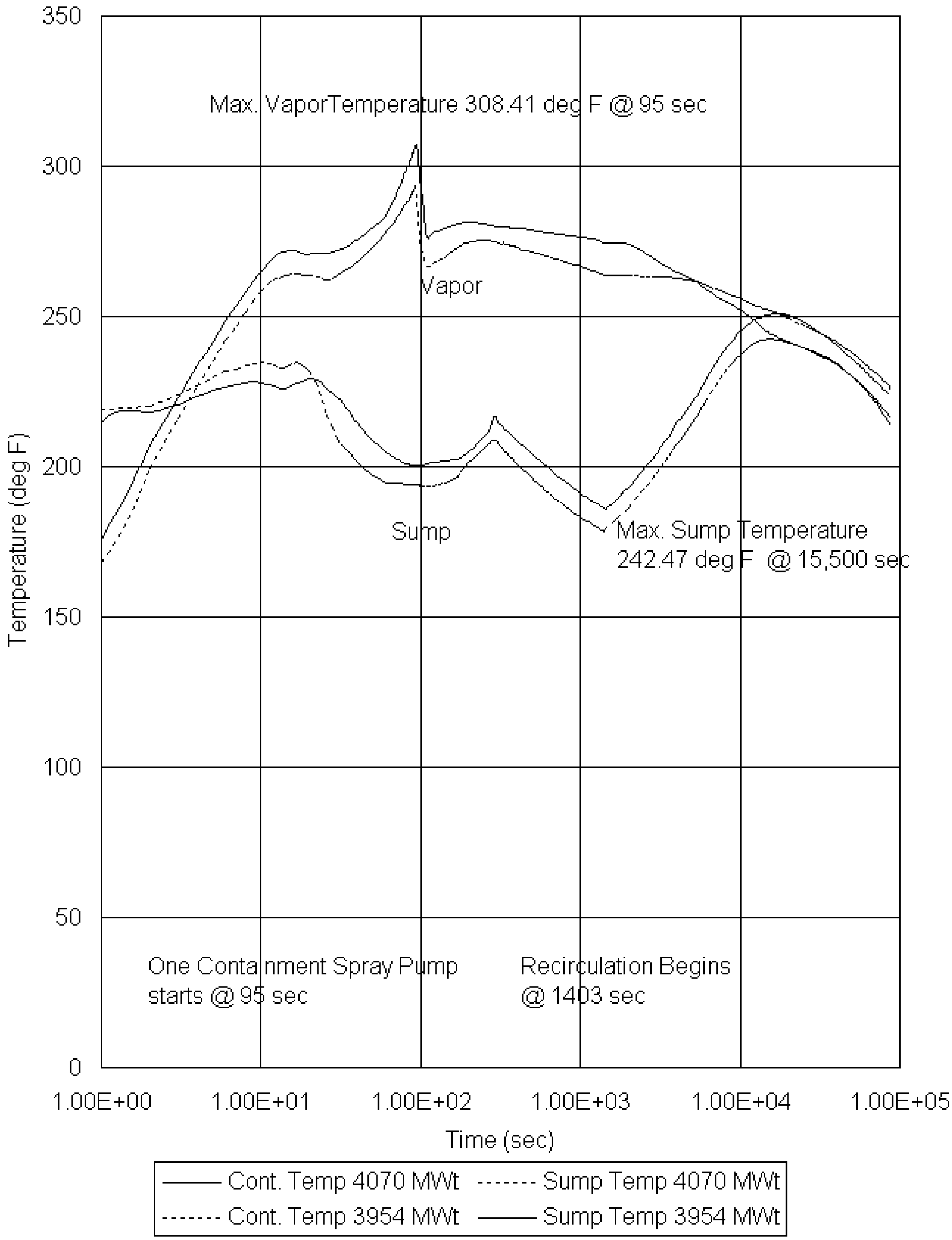


Figure 6.2-4
DEDLSB LOCA Condensing Heat Transfer Coefficient with Maximum ECCS

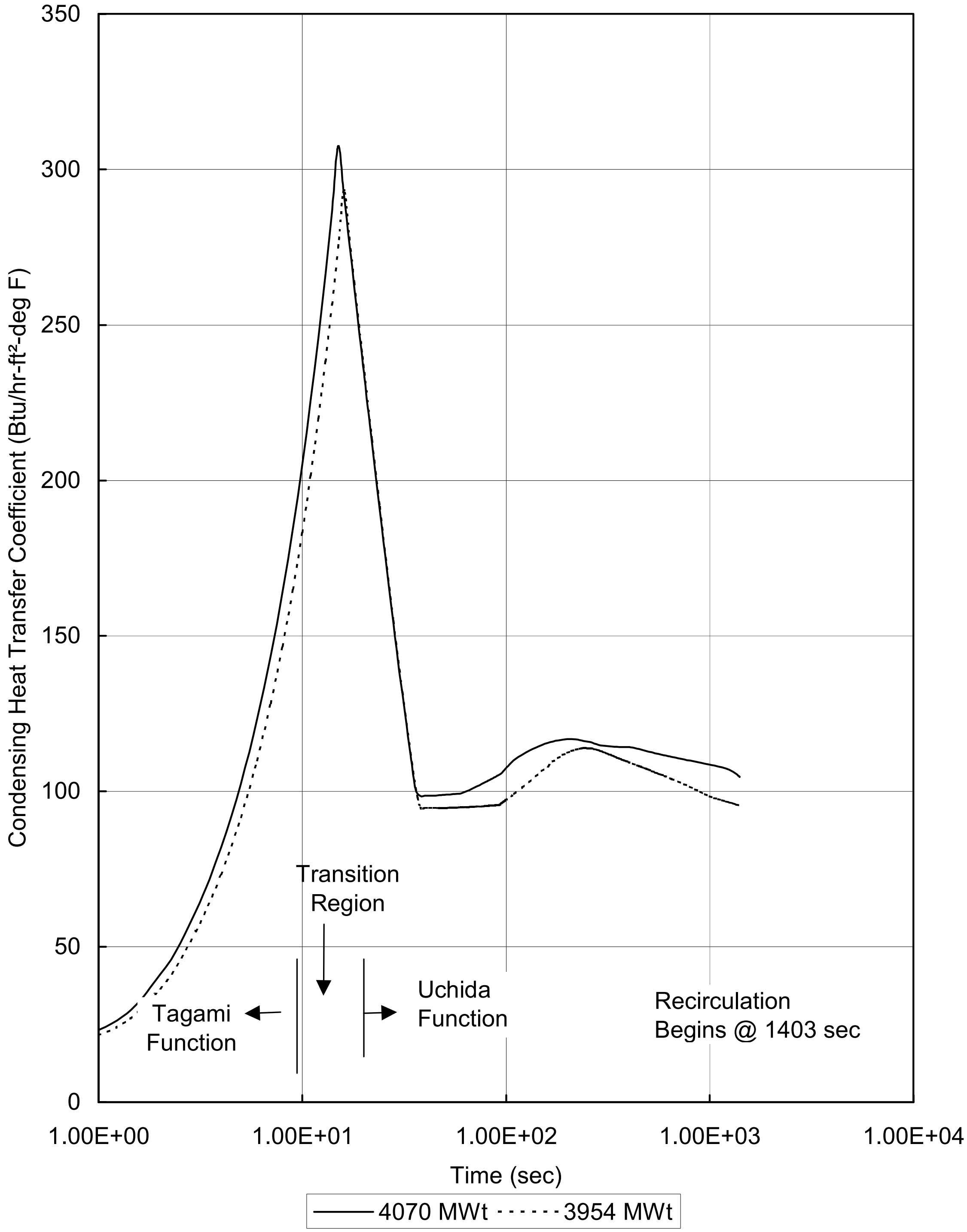
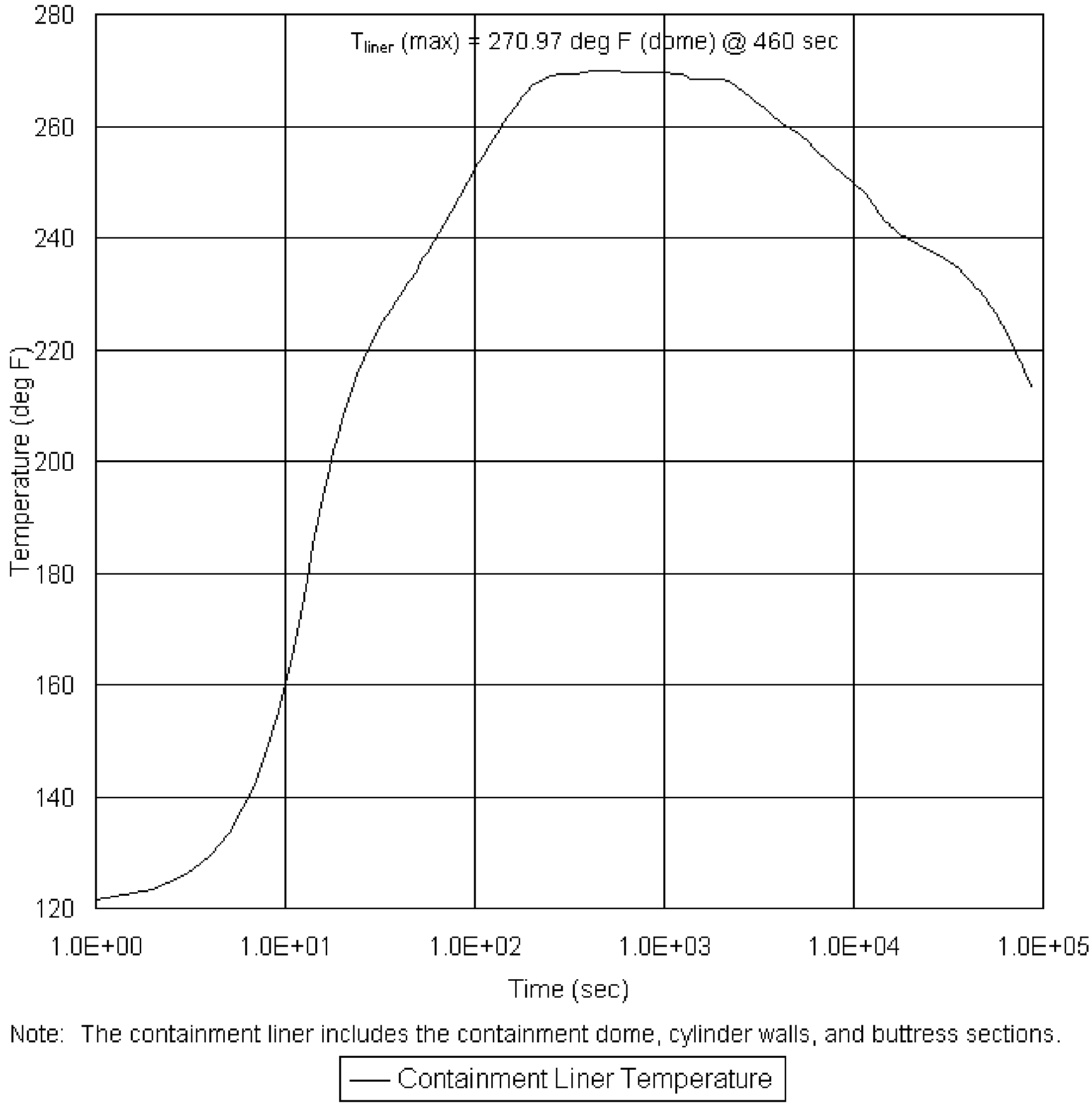


Figure 6.2-5
DEDLSB LOCA Containment Liner Temperature Profiles at 4070 MW_t



Section 6.2.3 Main Steam Line Break Containment Analysis

Section 6.2.3.1 Introduction and Background

The MSLB containment event is characterized by the rapid blowdown of steam into the containment due to a rupture in the main steam line.

The PUR and the larger SG volume and larger heat transfer area have a significant effect on the containment response to a MSLB. The following list identifies those changes that have the most impact:

1. The higher core thermal power results in higher RCS average temperature and higher decay heat, resulting in more energy being transferred to the SG, and increases the severity of the MSLB blowdown to the containment.
2. The additional SG secondary inventory and a higher secondary operating pressure result in increased mass and additional energy being transferred to the containment.
3. The increased power results in more Feedwater (FW), at a higher enthalpy, being delivered to the SGs for eventual release to the containment.

The MSLB M&E and containment response analysis have been performed in accordance with Sections 6.2.1 of the SRP. Additional NRC Bulletins and Information Notices have also been considered and/or implemented as part of this analysis. These include Bulletin 80-04 (Reference 6-28) for the treatment of main FW addition, and Information Notice 84-90 (Reference 6-11) and NUREG-0588 (Reference 6-12) for the treatment of Equipment Qualification (EQ) cases.

Section 6.2.3.2 Description of Main Steam Line Break Containment Analysis

The transient containment pressure and temperature response to a MSLB is determined utilizing M&E release data for a spectrum of cases for this event. The criteria of the overall containment analysis are the containment design pressure and EQ temperature profile.

These analyses focus on a matrix of cases that included three power levels (i.e., 102%, 75%, and 0% power).

For the EQ analysis, the calculated peak temperature case was then modified to include EQ related assumptions. These included additional super-heating upon U-tube uncovery as specified in the NRC Information Notice 84-90 (Reference 6-11) and NUREG-0588 8% of the containment wall condensate revaporization.

Section 6.2.3.3 Methodology Used for Main Steam Line Break Containment Analysis

Section 6.2.3.3.1 Main Steam Line Break Mass and Energy Release Calculations

The blowdown phase of the MSLB was simulated with SGNIII code (Reference 6-9 and Reference 6-10) to generate the M&E release data. These analyses conservatively assume the availability of non-emergency power, since it allows the continuation of RCP operation. This maximizes the rate of heat transfer to the affected SG that maximizes the rate of mass/energy release. With non-emergency power, an Emergency Diesel Generator (EDG) failure is not limiting and need not be postulated.

There is a Main Steam Isolation Valve (MSIV) in each of the four main steam lines. The MSIV's have been designed to close based on a conservative calculation that maximizes the dynamic pressure loading on the valve for all possible flowrates and qualities. Each valve has dual solenoid valves to assure closure even with a single failure in the control system. Single failure of the actuation signal will not prevent valve closure since both trains of Main Steam Isolation Signal (MSIS) actuation are provided to each MSIV. Any failure would result in the valve going to the closed position. The other MSIV isolates the unaffected SG. However, conservatively, a random failure is assumed as a failure of an MSIV in the broken steam line. This break would maximize the forward and reverse flow to the break and will maximize the consequences of the event.

There are two Feedwater Isolation Valves (FWIVs) in series in each of the two main FW lines. If one FWIV fails, the second FWIV will provide isolation. All cases were analyzed considering the flashing of fluid in the lines from the FWIVs to the affected SG; therefore, there is no need to do a separate analysis assuming FWIV failure. Some other significant assumptions for the M&E release calculations are:

- conservative volumetric expansion multiplier due to temperature and pressure,
- maximum RCS flow,
- maximum SG pressure, liquid, and steam inventories,
- no SG tubes plugged,
- maximum FW flow and enthalpy, and
- limiting initial containment pressure, temperature, and relative humidity.

Table 6.2-5 provides a summary of significant input parameters for MSLB analysis.

Section 6.2.3.3.2 Containment Response Analysis

The containment pressure and temperature responses to a MSLB are calculated using the computer program COPATTA (Reference 6-8). The event parameters are biased to maximize containment vapor temperature. The bounding event for containment peak pressure is the RCS DEDLSB, and maximized containment vapor temperature is characteristic of this postulated event. The primary differences between the LOCA analysis and the MSLB analysis are:

1. for the MSLB, the M&E release calculation terminates when the affected SG dries out and
2. the Uchida correlation is used for the heat transfer coefficient to the structural heat sinks in the MSLB, while the Tagami correlation is used for the LOCA (see Figure 6.2-4).

In addition to the parameters established above, total time for CSS actuation was reanalyzed. The existing analysis conservatively bounded the time to reach the containment high high-pressure setpoint of 10 psig. Total CSS actuation time is based on as-built plant conditions using the following approach:

Spray delay time for the Containment Spray Actuation Signal (CSAS) includes transmitter to Engineered Safety Features Actuation System (ESFAS) relay time, load sequencer time, CSS pump response time, and the time to fill the spray headers and establish full spray flow.

$$\text{spray delay time} = t_r + t_f$$

where:
t_r = total equipment response time with offsite power available = 23 seconds
(t_r is composed of 1 sec transmitter to ESFAS relay time, 16 sec load sequencer time, and 6 sec spray pump response time)
t_f = time to spray headers filled and full spray flow established = 59 seconds.

Thus, the spray delay time (total instrument and equipment response and fill time after pressure set point is reached) is 23 + 59 = 82 seconds.

For the 102% power case with PUR configuration, the time to reach the pressure setpoint of 10 psig is calculated by the COPATTA code to be approximately 8 seconds. Given an instrument and equipment delay time of 82 seconds, the CSS would start quenching the containment environment at approximately 90 seconds, or about 11.5 seconds earlier than the time for the existing power limit AOR.

Section 6.2.3.4 Results of Main Steam Line Break Containment Analysis

The M&E releases for the MSLB event inside containment were calculated and used to calculate the transient containment pressure and temperature response. The MSLB blowdown at 102% rated thermal power is the limiting case for both containment design and for equipment qualification. Figure 6.2-6 and Figure 6.2-7 provide a comparison plot, of current versus proposed power, of time dependent energy release rate to the containment environment for existing analysis conditions of 3954 MW_t (102% of 3876 MW_t) and the new core power of 4070 MW_t (102% of 3990 MW_t). Containment peak pressures and temperatures, and times to peak pressure, are summarized in Table 6.2-6 for analyzed MSLBs.

No long-term analysis is performed for the MSLB since after isolation and blowdown, there is no further energy input to containment. The maximum peak containment vapor temperature of 405.55 °F occurs at 90 seconds following initiation of the 102% power MSLB containment design case. The containment pressure, vapor and liner temperature profiles are presented in Figure 6.2-8 through Figure 6.2-11 for DBA MSLBs (containment design and EQ case). It is concluded that the cumulative effect of increased blowdown due to PUR is offset by a reduction in total time for ESFAS (CSAS). Therefore, as shown in Figure 6.2-8 through Figure 6.2-11, the existing temperature profiles for containment design and EQ due to MSLB remain bounding.

Table 6.2-5
Summary of Initial Condition Parameters for MSLB Analysis

Parameter	Value
Range of analyses reactor power level, MW_t	4070 (102% of 3990 MW_t) zero%
RCP heat input to RCS, MW_t	26
Range of T_{cold}, °F	568.2 @ 102% power, 572 @ zero% power
Initial RCS pressure, (psia)	2325
Total RCS flowrate, lb_m/hr (110% of unplugged SGs)	192.3E+06
RCS expansion multiplier	2%
Containment initial pressure, psia (minimum Technical Specification)[(1)]	13.2
Containment initial temperature, °F (maximum normal operating)	120
Containment initial relative humidity, %	20
Containment net free volume, ft^3	2.62E+06
Containment outside air temperature, °F	130
Refueling water temperature, °F (maximum)	120
RWT, gal (analytical minimum)	400,000
Condensate Storage Tank (CST) temperature, °F	120
Secondary expansion multiplier, %	2
Range of SG dome initial pressure, psia	1051.4 @ 102% power 1220 @ zero% power
Range of SG nominal inventory, lb_m	192,591 @ 102% power 282,064 @ zero% power
Range of FW temperature, °F	450.0 @ 102% power 120.0 @ zero% power
Auxiliary Feedwater (AFW) flow, gpm (2 pumps at run out)	3200

Note: (1) Selected values to maximize containment temperature since DEDLSB LOCA bounds all events for peak containment pressure.

Table 6.2-6
Summary of MSLB Results

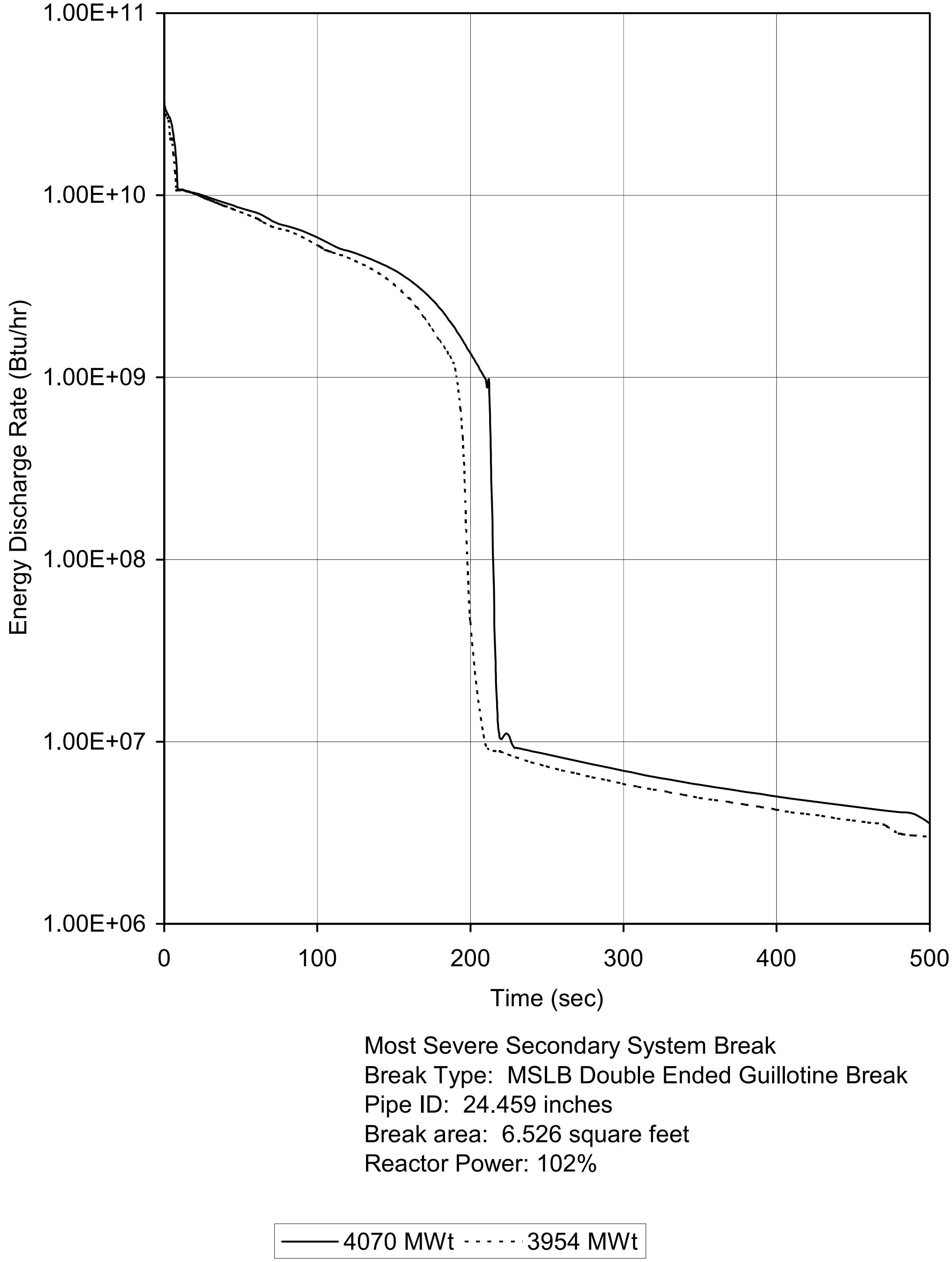
Parameters	Containment Design						EQ	
	102%		75%		0%		102%	
	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t
Peak containment pressure (psia)	51.82	55.19	49.94	55.54	51.74	55.27	51.84	55.95
Peak containment pressure (psig)	37.62	40.99	35.74	41.29	37.54	41.07	37.64	41.75
@ seconds	102	199.5	102	210.5	183	219.0	186.5	199
Peak containment vapor temperature (°F)	405.65	405.55	402.45	404.03	401.43	400.11	387.33	383.03
@ seconds	101.5	90.0	101.5	89.4	99	88.4	101.5	90.1
Peak sump temperature (°F)	249.45.	255.39	247.25	255.70	249.77	255.17	250.53	256.40
@ seconds	320	360.0	340	380.0	320	400.0	320	360.0
Blowdown time (seconds)	500	500	500	500	500	500	500	500
Peak containment building cylinder wall liner plate temperature (°F) (2)	(1)	252.16	-	-	-	-	-	-
@ seconds	(1)	252.0	-	-	-	-	-	-

Notes: (1) Not specifically extracted from the COPATTA output.
(2) Calculated for the limiting 102% power case.

Section 6.2.3.5 Conclusion

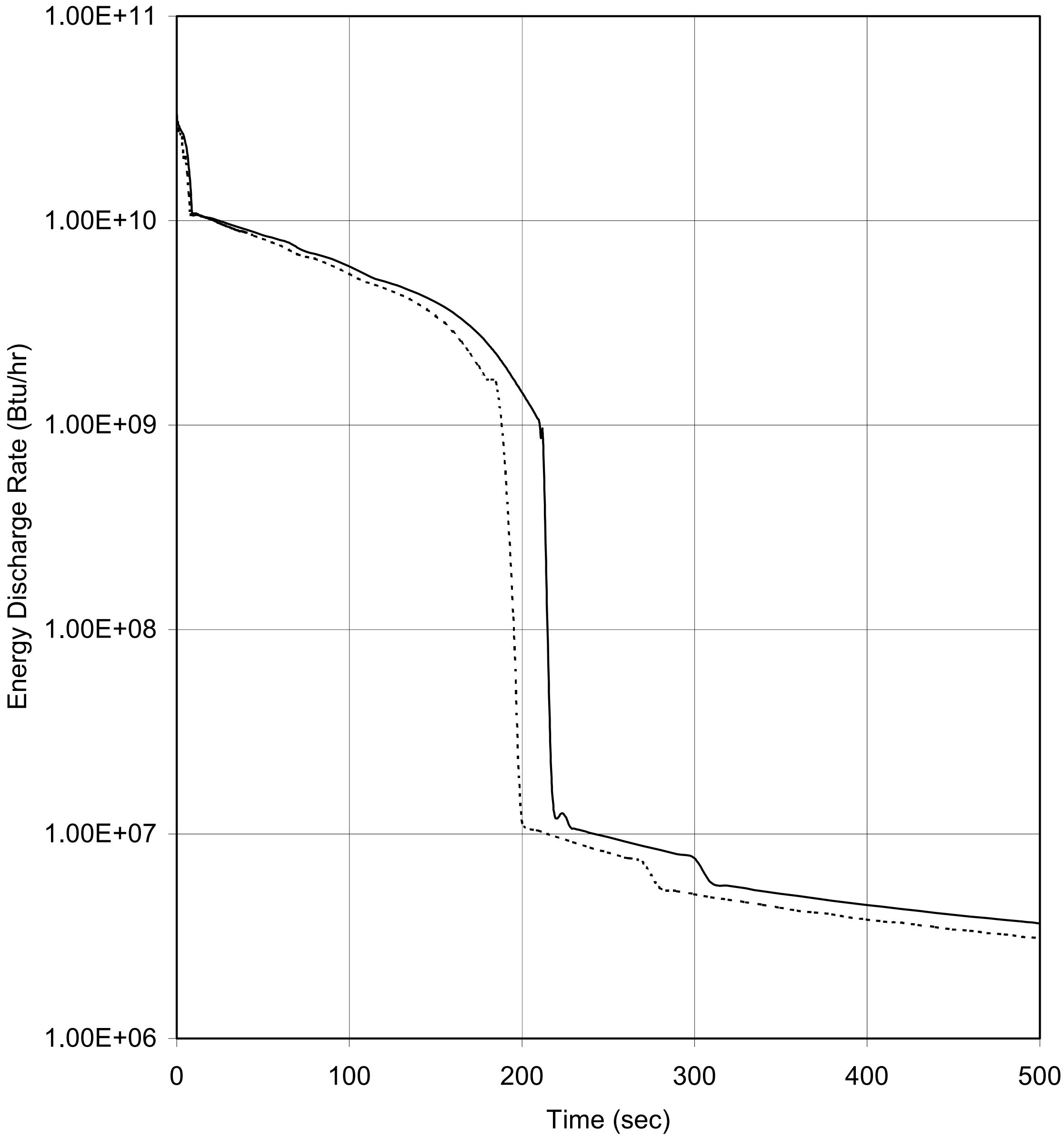
The impact of PUR on double-ended MSLB inside containment was analyzed. It is concluded that the pressure in the containment building is predicted to be 41.29 psig and 41.75 psig for EQ. This pressure is below the containment building design peak pressure of 60 psig.

Figure 6.2-6
MSLB Energy Discharge Comparison for Containment Design



Most Severe Secondary System Break
Break Type: MSLB Double Ended Guillotine Break
Pipe ID: 24.459 inches
Break area: 6.526 square feet
Reactor Power: 102%

Figure 6.2-7
MSLB Energy Discharge Comparison for EQ Design



Most Severe Secondary System Break
Break Type: MSLB Double Ended Guillotine Break
Pipe ID: 24.459 inches
Break area: 6.526 square feet
Reactor Power: 102%

4070 MWt 3954 MWt

Figure 6.2-8
MSLB Pressure Profile for Containment Design at 102% Core Power

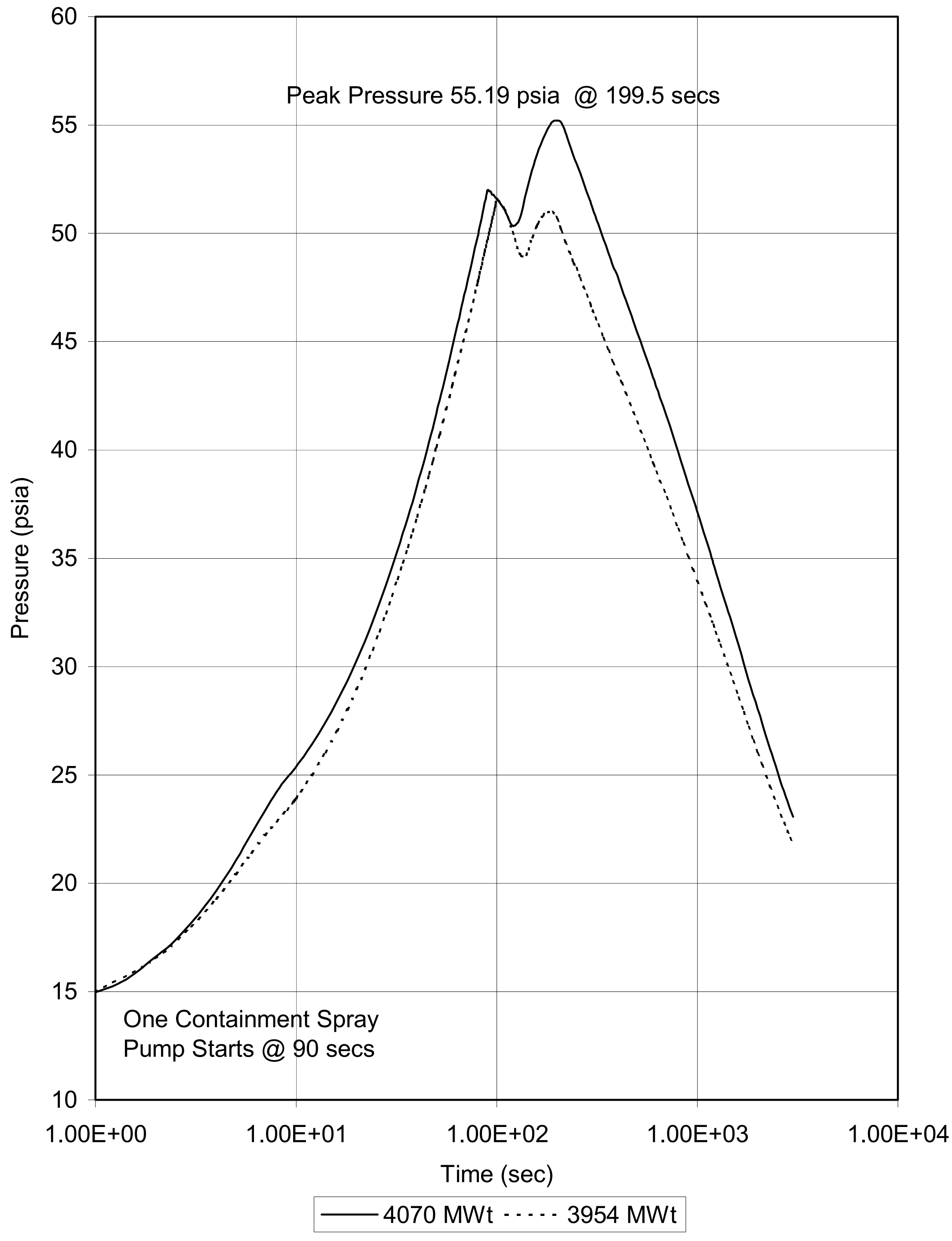


Figure 6.2-9
MSLB Containment Vapor and Sump Temperature Profile

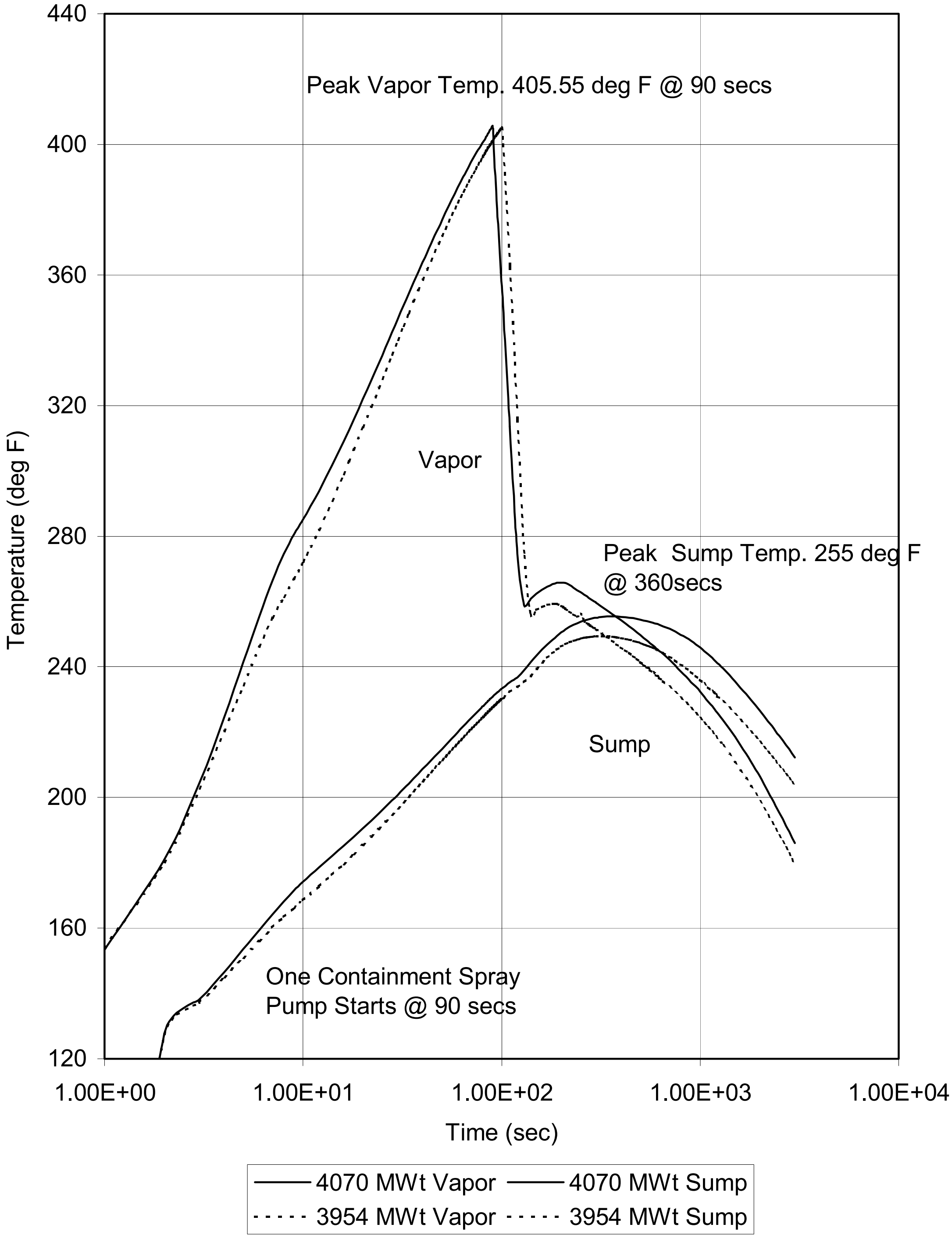


Figure 6.2-10
MSLB Containment Liner Temperature Profile at 4070 MW_t

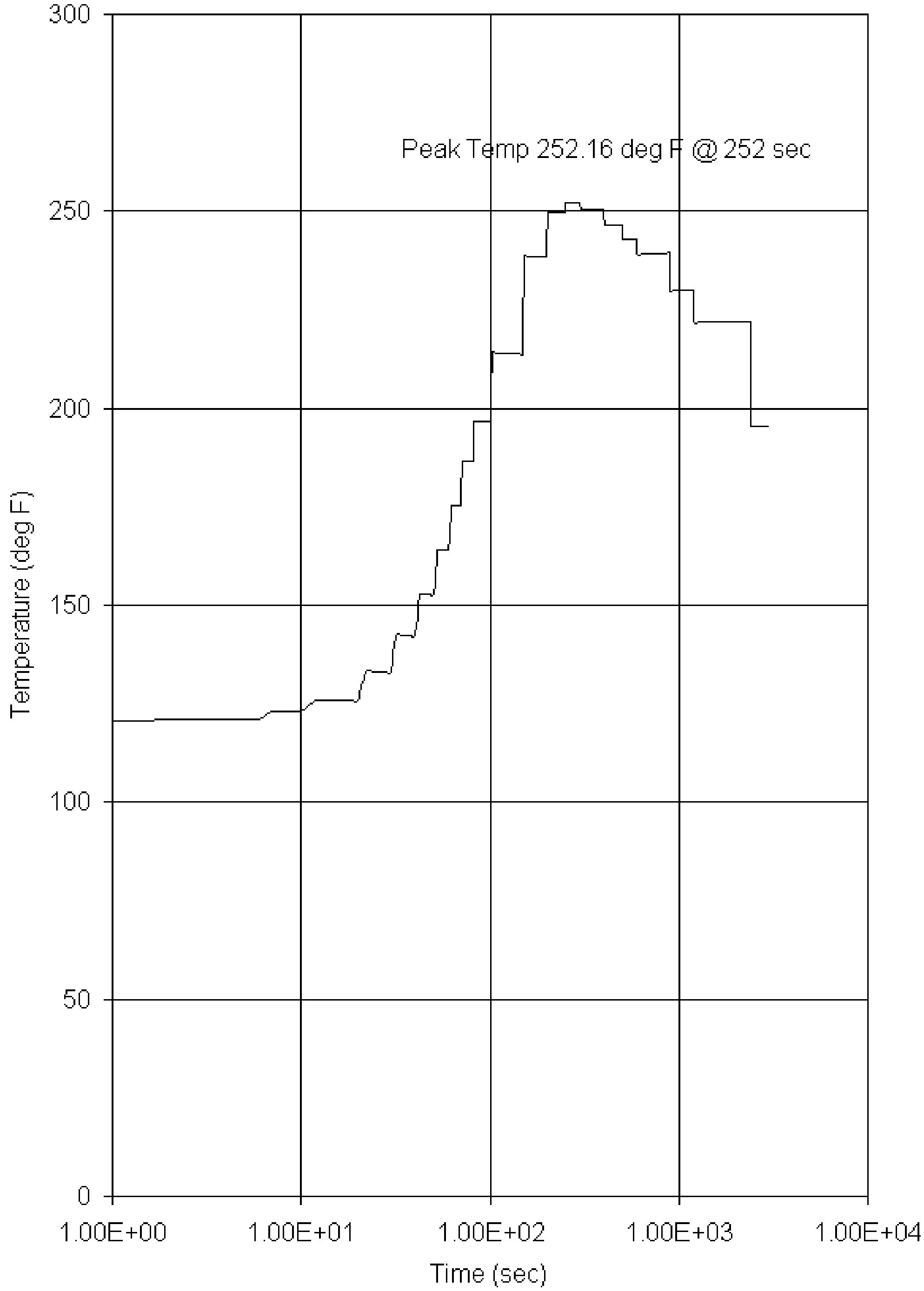
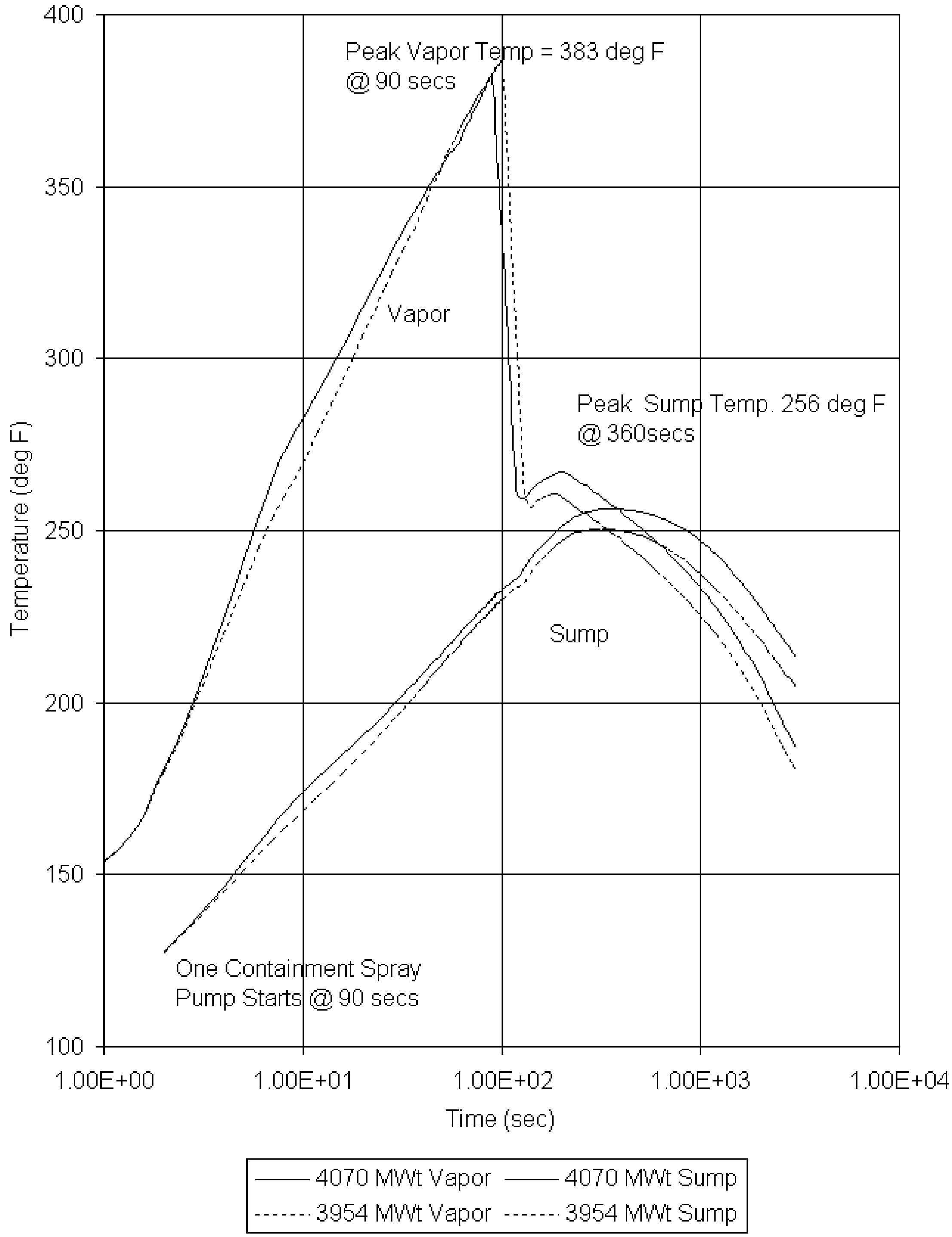


Figure 6.2-11
MSLB Temperature Profile for EQ Design at 102% Core Power



Section 6.2.4 Main Steam Line Break Outside Containment Analysis

Section 6.2.4.1 Introduction and Background

The planned PUR and the larger SG volume and larger heat transfer area have the potential to affect the outside containment response to a MSLB for the same reasons discussed in Section 6.2.3.1. This analysis is required to maintain EQ in the Main Steam Support Structure (MSSS).

In order to quantify the effect PUR has on the limiting event, a 1 ft^2 non-mechanistic steam line break (SLB) was analyzed. Analyses were performed in accordance with Section 3.6 of the SRP. Additional NRC Bulletins and Information Notices have also been considered and/or implemented as part of these analyses. These include Bulletin 80-04 for the treatment of main FW addition, and Information Notice 84-90 (Reference 6-11) and NUREG-0588 (Reference 6-12) for the treatment of EQ cases.

Section 6.2.4.2 Description of Main Steam Line Break Outside Containment Analysis

The breaks outside containment are assumed to be at the first weld outside containment. This assumption minimizes the flow resistance between the break and the affected SG and increases the calculated M&E release. The M&E release quantities are generated for EQ cases for this event. The analyses focus on M&E releases at 102% power and 0% power. The outside containment cases include the assumption that the MSIV in the steam line with the least flow resistance fails to close following the MSIS. This assumption maximizes the M&E release during a MSLB outside of the containment.

The outside containment EQ case assumes 1 ft^2 breaks. Super-heating within the SG initiates upon U-tube uncovery as specified in the NRC Information Notice 84-90.

The turbine stop valves are assumed to close instantaneously at the time of the reactor trip. This assumption is conservative for a MSLB event because the entire steam inventory at the time of reactor trip is assumed to be forced out the break.

No leakage is assumed through the MSIVs or Main Feedwater Isolation Valves (MFIVs). The AFW logic is assumed to function properly and to isolate all AFW to the affected SG.

Section 6.2.4.3 Methodology Used for 1 ft^2 Main Steam Line Break Outside Containment Equipment Qualification Analysis

Section 6.2.4.3.1 Mass and Energy Release Calculations

The blowdown phase of the MSLB was simulated with SGNIII code and Sections 6.2.1 of the SRP, to generate the M&E releases.

In addition, the inputs and assumptions listed in Section 6.2.3.3, additional significant inputs and assumptions for the outside containment MSLB are outlined below:

The outside containment EQ M&E release calculations assume:

- 1 ft^2 breaks;
- super-heating attributed to SG tube uncovery (required by NRC Information Notice 84-90); and
- nominal values for the moderator temperature entering the core.

Section 6.2.4.3.2 Change in SGNIII Code Methodology

The peak temperature pf the MSSS results based on the M&E releases calculated for PUR are bounded by the peak temperature values generated in the AOR. The M&E released during the 0% PUR case is bounded by the 0% power AOR. The 102% PUR case has a lower peak temperature because of a revision in the SGNIII code that better represents the secondary side. The revised code provides a more detailed modeling of the four main steam lines versus the original analysis which modeled only two main steam lines, the closing of the MSIVs and the steam flow through the main steam line cross header path following the closure of MSIVs. As a result, after 300 seconds into the event, the PUR M&E release rate begins to decrease faster than the M&E release rate in the AOR. The temperature in the AOR continues to increase while the temperature in the PUR analysis is decreasing. The peak temperature for the PUR case occurs at 290 seconds and is based on about 44E+06 Btu's of released energy. The temperature for the AOR does not peak until 400 seconds and is based on about 49E+06 Btu's of released energy. The improved secondary model provides a better representation of the post-MSIS M&E release.

There are also differences between the methodology used for the new PUR analysis and the AOR methodology. These methodology differences include a reduction in some conservative input values selected in the AOR and a revised reactor trip methodology. Based on detailed analyses, it has been concluded that reactor trip on Core Protection Calculator (CPC) Variable Over-Power Trip (VOPT) and low SG pressure can be credited if the effects of the Moderator Temperature Coefficient (MTC) are accounted for. The analysis submitted here evaluates all reasonable reactor trips and it identifies the most conservative trip. The methodology applied for selection of reactor trip is as follows:

> To select the appropriate trip sequence, successive cases are run with successively less negative MTC values until the plant trips on low SG pressure (rather than high power). A less negative MTC will result in a quicker trip on low SG pressure while a more negative MTC will result in a quicker trip on high reactor power. The high power trip is used if the less negative MTC allowed by the Technical Specifications is reached.

The changes in methodology have allowed eliminating the need to credit a reactor trip on low SG pressure when evaluating MSLBs outside of the containment. The computer code changes have resulted in producing M&E for a MSLB outside containment that have reduced the predicted MSSS building temperature below the existing AOR (See Figure 6.2-14).

Section 6.2.4.3.3 Main Steam Support Structure MSLB Pressure and Temperature Response Analysis

This section describes the sub-compartment pressure/temperature model used for PUR. This model is identical to the existing AOR. The MSSS is modeled by breaking the structure up into eight sub-compartments. Compartment boundaries are established at locations with restrictions to flow such as gratings and piping. The boundaries of each compartment are made up of the exterior concrete walls, the center concrete wall, and the grating at elevations 120'-0", 132'-0", and 140'-0". The peak blowdown temperatures and peak pressures are developed using this multiple volume model.

The long-term cooldown profile is developed using a single volume model. This model treats the four compartments on the south side of the MSSS center wall as a single thermal volume.

Computer code PCFLUD (Reference 6-34) was used to evaluate this multi nodal model. PCFLUD is designed to analyze a variety of problems dealing with the thermodynamics of one-dimensional ideal gas flows in a system of interconnected compartments and an external environment. Basic equations of M&E conservation are used along with either an explicit quasi steady state or a unique implicit, finite difference flow methodology to calculate the transfer of M&E among the various compartments and Structures, Systems, and Components (SSCs) that comprise the system. PCFLUD is used to calculate pressure and temperature transients to evaluate structure impact during a hypothetical pipe break accidents. It is used to calculate long-term temperature transients resulting from pipe breaks or LOCAs.

Significant inputs and assumptions for MSSS pressure/temperature model outside containment MSLB are outlined below. For the calculation of peak temperatures during the blowdown phase of the profile, a multiple volume MSSS model is developed with the following conditions:

- ambient conditions: 122°F, 90% humidity, 14.2 psia,
- doors in center wall between compartments are open to maximize the MSSS internal equipment temperature,
- free flow to turbine building/corridor out through the main steam and FW penetrations,
- MSLB blowdown flow split between adjacent compartment to the break location,
- a value of 1*Uchida for the condensation heat transfer coefficient,
- the MSSS missile doors are assumed to be closed and remain closed throughout the transient analysis to maximize the internal building temperature. These doors are designed to withstand high differential pressures,

- a constant leakage rate of 5,000 lb_m/hr (1.389 lb_m/sec) is added to the steam mass flowrate in the PCFLUD code blowdown data. This value accounts for the back-leakage through the AFW steam supply check valve from the unaffected SG and continues for 13.5 hours. It is conservatively assumed that the unaffected SG is maintained at the 0% power nominal operating pressure of 1170 psig. The corresponding blowdown enthalpy, 1185.5 BTU/lb_m, is the saturated enthalpy of 1170 psig, and
- a constant convective heat transfer coefficient is assumed of 1 BTU/hr/ft^2/F. This value accounts for natural convection from the vapor/air mixture in the room to the surface of heat sinks.

In addition, calculation of the temperature envelope during the long-term cooldown phase, a single volume MSSS model is developed with the following conditions:

- constant flow from the turbine building/corridor into the main steam and FW penetrations to model the buoyancy induced flow and
- a value of 4*Uchida for the condensation heat transfer coefficient.

Section 6.2.4.4 Results of this Analysis

The M&E release for a 1 ft^2 MSLBs outside of the containment was calculated using the SGNIII code. The results were used to calculate the potential effects of a MSLB on essential equipment outside of containment using code PCFLUD (Reference 6-34). Figure 6.2-12 provides a comparison plot, of current versus proposed power, of mass release rate versus time. Figure 6.2-13 provides comparison plot, of current versus proposed power, of rate of energy discharge to compartments containing breaks at zero and 4070 MW_t (102% of 3990 MW_t). Pressure and temperature profiles for the most limiting compartment are presented in Figure 6.2-14. Table 6.2-7 provides a comparison of result of a 1 ft^2 SLB break outside containment. As it can be deduced, the change in code modeling results in a more realistic pressure and temperature profiles. The peak temperature and pressure for the event remain bounded by the existing analysis.

Table 6.2-7
Summary of Results for 1 Sq. ft. MSLB in the MSSS for EQ

	MSLB at zero% power		MSLB at 102% license power	
	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t
Peak Temperature (°F) [(1)]	383	367	373	357
Time to reach peak temperature (sec)	289	290	400	290
Peak pressure (psia) [(2)]	15.2	15	15	14.8
Time to reach peak pressure (sec)	<1	<1	<1	<1

Notes: (1) Initial temperature in the MSSS is assumed 122 °F.
(2) Initial pressure in the MSSS is assumed to be 14.2 psia.

Section 6.2.4.5 Conclusion

The impact of PUR on double-ended MSLB and Feedwater Line Break (FWLB) outside containment in the MSSS was analyzed. It is concluded that the existing AOR would remain bounding for pressure profiles. The peak pressure in the MSSS building is predicted to be 30.2 psia (16 psig). This pressure is below the MSSS building design pressure of 35.1 psia (20.9 psig).

Figure 6.2-12
MSLB Outside Containment Mass Release Comparison

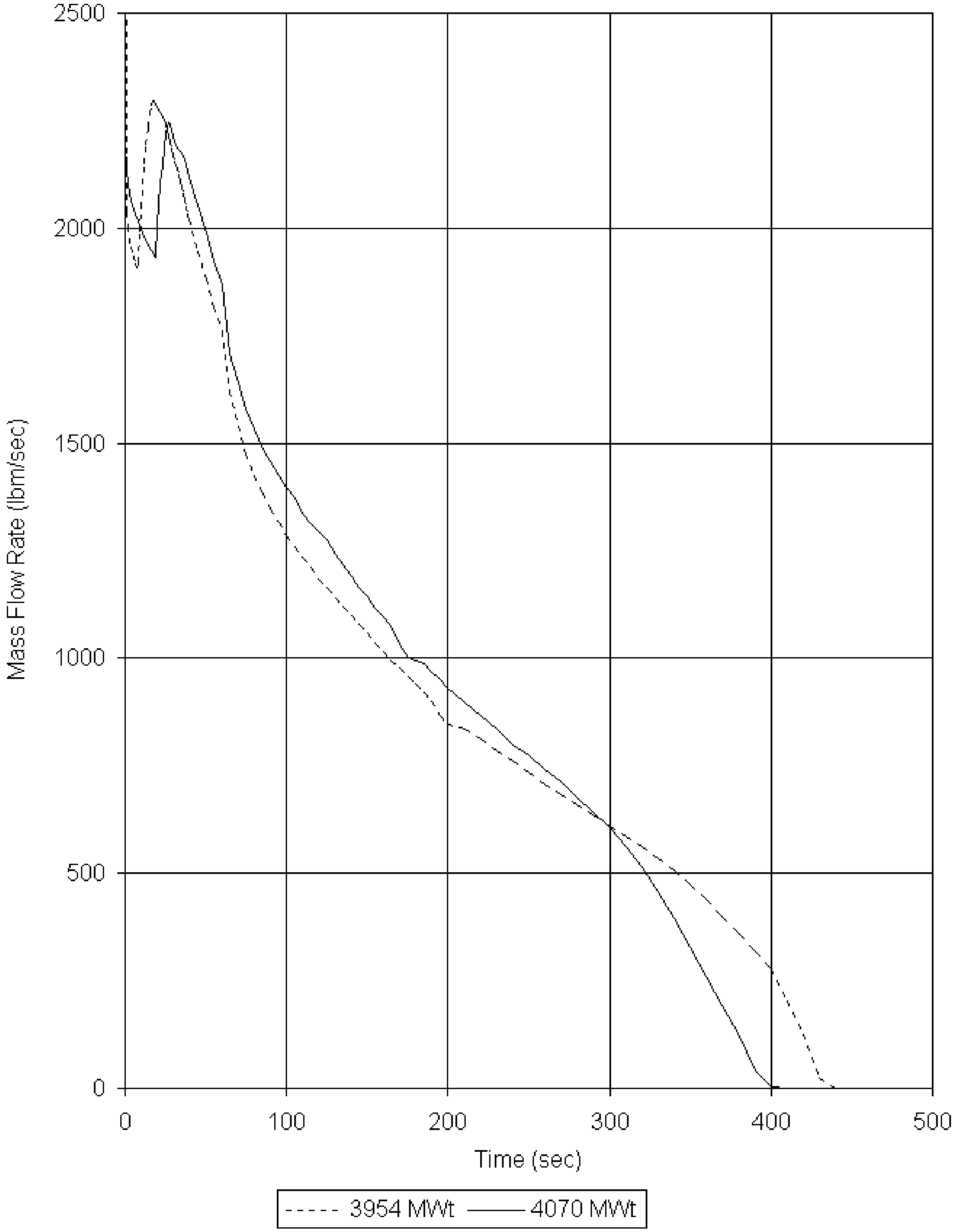


Figure 6.2-13
MSLB Outside Containment Energy Discharge Comparison

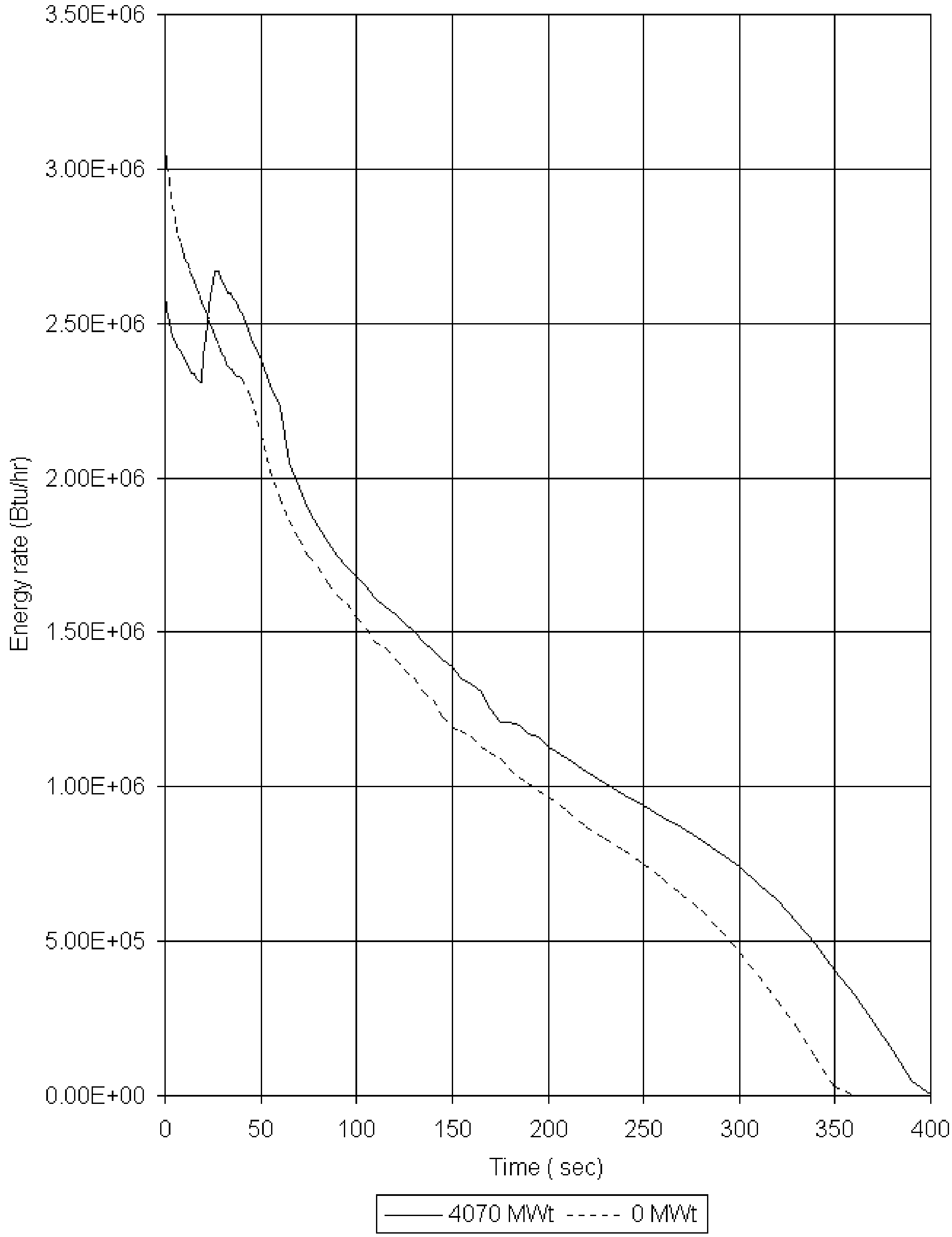
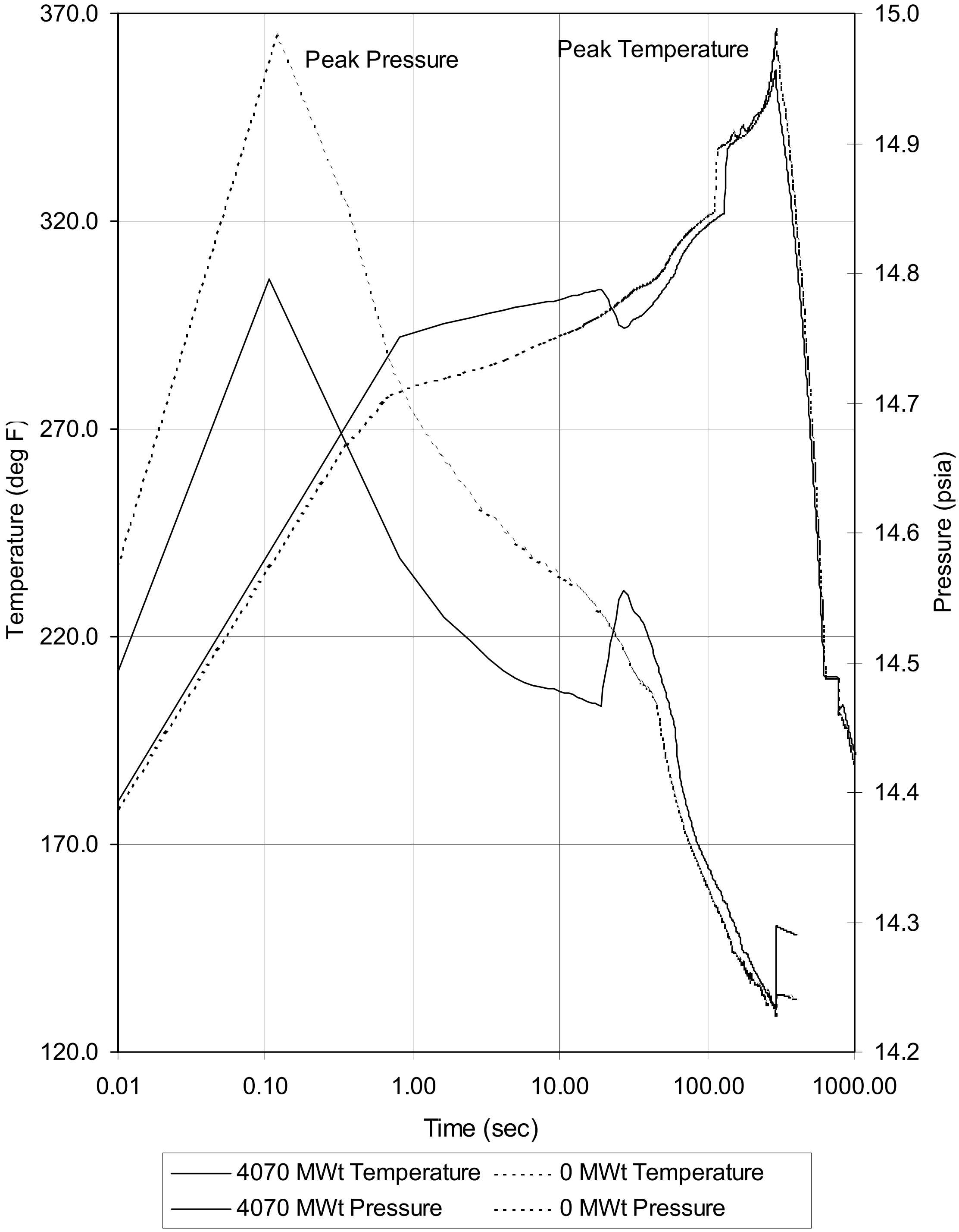


Figure 6.2-14
MSLB Outside Containment Pressure and Temperature Profile



Section 6.3 Non-Loss-of-Coolant Accident Transient Analysis

Summary

UFSAR Chapter 15 Non-LOCA transient events were evaluated for PUR. The evaluations for 4070 MW_t (3990 MW_t with 2% power uncertainty) were performed at conditions associated with the RSGs. The results of the evaluation demonstrate that the transients continue to be within the acceptance criteria. The Combustion Engineering Nuclear Transient Simulator (CENTS) code was employed to simulate the NSSS response to the transients. The CENTS code has been generically approved by the NRC for the calculation of transient response in Pressurized Water Reactors (PWRs) with limitations. APS has recieved amendment 137 to Technical Specification 5.6.5.b to add the CENTS code to the list of codes approved for use in determining core operating limits. APS has used the CENTS code in place of the CESEC III code in the calculation of NSSS response to Non-LOCA events. In addition to the CENTS-based evaluation of Non-LOCA transient events for the PUR and RSG, this report also includes the CENTS-based evaluation of the existing plant configuration. Comparison of these two evaluations illustrates the PUR and RSG impact, eliminating the impact of code change.

Examination of the NSSS response curves and sequence of events tables for all of the Chapter 15 Non-LOCA transient events reveals:

- PUR does not significantly affect dose consequences of the transient events. The approximate 3% increase in core source term due to PUR yields a proportional increase in offsite dose.
- PUR maintains current fuel design limits, which are measured by the amount of fuel failures and the approach to Specified Acceptable Fuel Design Limits (SAFDLs).
- Peak RCS and secondary system pressures, experienced during the decreased heat removal by the secondary system events, increase because of PUR operating point. Calculated peak pressures remain below acceptable limits.
- Shutdown margin requirements for the MSLB events increase because of the larger RSGs. With PUR, there is sufficient shutdown margin and acceptable consequences for these events.
- The overall NSSS response during non-LOCA transients is similar to between the existing plant configuration and PUR.

Section 6.3.0 Introduction

All UFSAR Chapter 15 non-LOCA transient analyses were evaluated to demonstrate acceptability at PUR conditions. Analyses supporting both existing plant configuration and PUR with RSGs are presented in this section. These analyses are labeled "3876 MW_t" and "3990 MW_t" respectively.

The Design Basis Events (DBEs) presented in this section are categorized, based on frequency of occurrence, into three groups:

- Anticipated Operational Occurrences (AOOs),
- infrequent events, and
- limiting faults.

Table 6.3-1 lists the non-LOCA transient events by category and defines the level of evaluation included for PUR. All UFSAR events received some level of evaluation to ensure acceptable consequences following PUR. The levels of evaluation are:

- The analysis "remains bounded" for PUR by an existing UFSAR analyses.
- The analysis has been "reanalyzed" as part of this submittal.
- The analysis has been "evaluated" for PUR, the UFSAR results remain bounding, and a complete "reanalysis" was not required.

Table 6.3-1
Non-LOCA Transient Events
(Page 1 of 4)

UFSAR	Transient Event	Category	PUR Assessment
15.1	Increase in Heat Removal by the Secondary System		
15.1.1	Decrease in FW Temperature	AOO	Remains bounded by the increase in main steam flow events in Section 6.3.1.1.
15.1.2	Increase in FW Flow	AOO	Remains bounded by the increase in main steam flow events in Section 6.3.1.2.
15.1.3	Increase in Main Steam Flow	AOO	Reanalyzed in Section 6.3.0.3.1 and Section 6.3.1.3 as part of the CPC signal filters analysis.
	Steam Bypass Control System (SBCS) Malfunction	AOO	Reanalyzed in Section 6.3.1.3.

Table 6.3-1
Non-LOCA Transient Events

(Page 2 of 4)

UFSAR	Transient Event	Category	PUR Assessment
15.1.4	Inadvertent Opening of an Atmospheric Dump Valve (ADV) (IOSGADV) with a LOP	Infrequent Event	Reanalyzed in Section 6.3.1.4.
15.1.5	MSLB - Modes 1 and 2, Post-Trip Return-to-Power (R-t-P), Pre-Trip Power Excursion	Limiting Fault	Reanalyzed in Section 6.3.1.5 and Section 6.3.1.7.
15.1.6	MSLB - Mode 3 Post-Trip R-t-P	Limiting Fault	Reanalyzed in Section 6.3.1.6.
15.2	Decrease in Heat Removal by the Secondary System		
15.2.1	Loss of External Load	AOO	Reanalyzed in Section 6.3.2.1 and Section 6.3.0.3.1 as part of CPC signal filters analysis.
15.2.2	Turbine Trip	AOO	Remains bounded by the Loss of Condenser Vacuum (LOCV) in Section 6.3.2.2 and Section 6.3.2.3.
15.2.3	LOCV	AOO	Reanalyzed in Section 6.3.2.3.
15.2.4	MSIVs Closure	AOO	Remains bounded by the LOCV in Section 6.3.2.3 and Section 6.3.2.4.
15.2.5	Steam Pressure Regulator Failure	AOO	N/A (Section 6.3.2.5)
15.2.6	Loss of Non-Emergency AC Power	AOO	Remains bounded by the LOCV in Section 6.3.2.3 and Section 6.3.2.6.
15.2.7	Loss of Normal FW Flow	AOO	Remains bounded by the LOCV in Section 6.3.2.3 and Section 6.3.2.7.
15.2.8	Feedwater Line Breaks (FWLBs)	Limiting Fault	Reanalyzed in Section 6.3.2.8.
15.3	Decrease in RCS Flowrate		
15.3.1	Total Loss of RCS Flowrate	AOO	Reanalyzed in Section 6.3.3.1.
15.3.2	Flow Controller Malfunction	AOO	N/A (Section 6.3.3.2).
15.3.3	Single RCP Rotor Seizure with a LOP	Limiting Fault	Remains bounded by the RCP shaft break event in Section 6.3.3.3.

Table 6.3-1
Non-LOCA Transient Events

(Page 3 of 4)

UFSAR	Transient Event	Category	PUR Assessment
15.3.4	Single RCP Shaft Break with LOP	Limiting Fault	Reanalyzed in Section 6.3.3.4.
15.4	Reactivity and Power Distribution Anomalies		
15.4.1	Uncontrolled Control Element Assembly (CEA) Withdrawal (CEAW) - Subcritical and Hot Zero Power (HZP)	AOO	Reanalyzed in Section 6.3.4.1.
15.4.2	Uncontrolled CEAW at Power	AOO	Reanalyzed in Section 6.3.0.3.1 as part of CPC signal filters analysis. Long-term NSSS response detailed in Section 6.3.4.2.
15.4.3	Single Full-Length CEA Drop	AOO	Reanalyzed in Section 6.3.4.3.
15.4.4	Startup of an Inactive RCP	AOO	Evaluated in Section 6.3.4.4.
15.4.5	Flow Controller Malfunction Causing an Increase in BWR Core Flow	AOO	N/A (Section 6.3.4.5).
15.4.6	Inadvertent Deboration (ID)	AOO	Evaluated in Section 6.3.4.6.
15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Location	AOO	Evaluated in Section 6.3.4.7.
15.4.8	CEA Ejection	Limiting Fault	Reanalyzed in Section 6.3.4.8. Fuel failure is evaluated on a cycle-by-cycle basis.
15.5	Increase in RCS Inventory		
15.5.1	Inadvertent Operation of ECCS	AOO	Evaluated in Section 6.3.5.1.
15.5.2	Chemical and Volume Control System (CVCS) Malfunction - Pressurizer Level Control System (PLCS) Malfunction with LOP	Infrequent Event	Evaluated in Section 6.3.5.2.
15.6	Decrease in RCS Inventory		
15.6.1	Inadvertent Opening of a Pressurizer Safety Valve (PSV)	AOO	Evaluated as part of ECCS performance analysis in Section 6.1 and Section 6.3.6.1.

Table 6.3-1
Non-LOCA Transient Events

(Page 4 of 4)

UFSAR	Transient Event	Category	PUR Assessment
15.6.2	Double-Ended Break of a Letdown Line Outside Containment of the letdown line control valve (DBLLOCUS)	Limiting Fault	Reanalyzed in Section 6.3.6.2.
15.6.3	Steam Generator Tube Rupture (SGTR) with LOP	Infrequent Event	Reanalyzed in Section 6.3.6.3.3.
	SGTR with LOP and single failure	Limiting Fault	Reanalyzed in Section 6.3.6.3.2.
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Limiting Fault	N/A (Section 6.3.6.4).
15.6.5	LOCAs	Limiting Fault	Evaluated as part of ECCS performance analysis Section 6.1 and Section 6.3.6.5.
15.7	Radioactive Material Release from a Subsystem or Component		
15.7.1	Waste Gas System Failure	Limiting Fault	Evaluated in Section 6.3.7.
15.7.2	Radioactive Liquid Waste System Leak or Failure	Limiting Fault	Evaluated in Section 6.3.7.
15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Limiting Fault	Evaluated in Section 6.3.7.
15.7.4	Radioactive Consequences of Fuel Handling Accidents	Limiting Fault	Evaluated in Section 6.3.7.

Section 6.3.0.1 Methodology and Computer Codes

The transient analysis methodology used for PUR analyses is the same as the methodologies documented in the UFSAR, except where noted. The transient simulation code CENTS (Reference 6-17) was employed for a majority of the Chapter 15 events. The CENTS code calculates the decay heat using:

1. ANS/ANSI 5.1-1979 (Reference 6-36) including two-sigma uncertainty and representing heavy element decay and fission product neutron capture, or

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.

Section 6.3.0.2 Initial Conditions

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.

The consequences of a given transient may be insensitive to certain initial conditions. In these instances, a nominal value within the range specified in Table 6.3-2 was selected.

Before the transient event, non-safety grade control systems were selected to be in their most adverse configuration (i.e., manual or automatic) consistent with steady state plant operations. For example, letdown and charging flow will be determined by either an automatic response to the PLCS or manually set to a constant and equal value. Initial system settings that are not consistent with steady state operation and introduce a system perturbation (i.e., minimum letdown along with maximum charging) were not considered. Non-safety grade control systems (i.e., Reactor Power Cutback System (RPCS), SBCS, Reactor Regulating System (RRS) that would act to mitigate the severity of transients were not credited.

The Core Operating Limits Supervisory System (COLSS) monitors various instruments and alerts operators on approach to many of the Technical Specification Limiting Conditions of Operation (LCOs). In addition, COLSS preserves DNB and Linear Heat Rate (LHR) margin that is important in maintaining the initial conditions assumed by the safety analyses. COLSS displays the available thermal margin via DNB-Power Operating Limit (POL) and LHR-POL meters and alerts operators on approach to the prescribed POL. Thus, COLSS ensures that the initial conditions of the safety analyses are not exceeded.

Table 6.3-2
Range of Initial Conditions Evaluated in the Non-LOCA Transient Analyses

(page 1 of 3)

Parameter	Units	Lower Limit	Upper Limit
Steady-State Conditions			
Core power	MW_t	0	3990 (3876) (2)
Core inlet temperature HZP HFP (Hot Full Power)	°F	548 548	572 566 (562) (2)
RCS mass flowrate	% of design E+06 lb_m/hr	95 155.80	116 190.24
Pressurizer pressure	psia	2100	2325
Pressurizer level	% feet	24 11.4	59 23.9
SG level	% Narrow Range (NR) feet	4 32.8 (31.2) (2)	92 45.3 (42.3) (2)
SG tube plugging	% of tubes per SG	0	10 (16) (2)

Table 6.3-2
Range of Initial Conditions Evaluated in the Non-LOCA Transient Analyses

(page 2 of 3)

Parameter	Units	Lower Limit	Upper Limit
Safety Valve Lift Setpoints			
Main Steam Safety Valves (MSSVs), lift bank #1 bank #2 bank #3	psia	 1227 1266 1290	 1303 1344 1370
PSVs, lift	psia	2450	2550
Core Physics Conditions (1)			
Axial power distribution ≥50% power <50% power	Axial Shape Index (ASI)	 -0.20 -0.30	 +0.20 +0.20
Azimuthal tilt ≥20% power <20% power	%	 0 0	 5 10
MTC HZP HFP	E+04 $\Delta\rho/°F$	EOC -3.20 -4.00	BOC +0.50 -0.20
Fuel temperature coefficient	$\Delta\rho/°K^{1/2}$	EOC -0.00240	BOC -0.00131
Kinetics - beta group 1 group 2 group 3 group 4 group 5 group 6 TOTAL	- -	EOC 0.000137 0.000982 0.000871 0.001818 0.000714 0.000178 0.004700	BOC 0.000255 0.001617 0.001458 0.003113 0.001152 0.000284 0.007879
Kinetics - lambda group 1 group 2 group 3 group 4 group 5 group 6	Sec^{-1}	EOC 0.01280 0.03140 0.12385 0.32809 1.40800 3.78820	BOC 0.01276 0.03160 0.12070 0.32180 1.40040 3.84780

Table 6.3-2
Range of Initial Conditions Evaluated in the Non-LOCA Transient Analyses

(page 3 of 3)

Parameter	Units	Lower Limit	Upper Limit
Prompt neutron lifetime	E-06 sec	EOC 35	BOC 12
HFP CEA scram worth N-1 N-2 (CEA ejection)	$\%\Delta\rho$	EOC -8.5 -5.5	BOC -8.0 -5.5

Notes: (1) The burnup dependence of the bounding physics parameters is denoted by Beginning of Cycle (BOC) and End of Cycle (EOC).
(2) Value in parenthesis represents limit for the existing plant configuration (i.e., 3876 MW_t).

Section 6.3.0.3 Reactor Protection Systems

The Reactor Protection Systems (RPS) provides the necessary reactor trips required to mitigate the consequences of non-LOCA transient events. Table 6.3-3 lists the analytical trip setpoints credited in the transient analyses. The analytical setpoints have been calculated for both normal and harsh environments. The analytical setpoints include instrument uncertainties that were applied to delay the response of the RPS. The manner in which the RPS responds to each transient event is detailed in each event section.

The Core Protection Calculator System (CPCS) calculates DNBR and Local Power Density (LPD). DNBR and LPD trips assure that the SAFDLs on DNB and centerline fuel melting are not exceeded during AOOs and assist the ESFAS in limiting the consequences of certain postulated accidents. In addition to DNB and LPD trip functions, CPCs provide reactor trips generated on VOPT, low RCP shaft speed, hot leg saturation temperature, cold leg differential temperature, and various out-of-range trips.

To ensure a conservative CPC response during transient conditions, certain algorithms have been included to dynamically adjust processed sensor inputs and power values to compensate for signal and processing delays. The magnitude of dynamic compensation is affected by certain DBEs. These DBEs were evaluated to assess the impact of PUR.

Table 6.3-3
RPS Analytical Setpoints Credited in the Transient Analysis
(Page 1 of 2)

Parameter	Units	Analytical Setpoint	Response Time
High pressurizer pressure normal harsh	psia	 2415 2450	0.50 sec
Low pressurizer pressure normal harsh	psia	 1750 1670	1.15 sec
High containment pressure normal harsh	psig	 N/A 5.0	1.15 sec
High Log Power Trip (HLPT) normal harsh	%	 0.060 N/A	0.50 sec
VOPT normal harsh	% (band)	 11.0 N/A	0.45 sec
Low RCS flow (SG ΔP)	psid	Note (1)	Note (1)
LSGP normal harsh	psia	 915 (850) 875 (810) (2)	1.15 sec
High SG level normal harsh	% Wide Range (WR)	 97.0 N/A	1.15 sec
Low SG level normal harsh	%WR	 40.0 35.0	1.15 sec
CPC DNBR normal harsh	—	 1.34 N/A	0.30 - 1.35 sec depends on sequence
CPC LPD normal harsh	kW/ft	 21.0 N/A	0.75 - 1.35 sec depends on sequence
CPC low RCP shaft speed normal harsh	Fraction	 0.95 0.95	0.30 sec

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

(Page 2 of 2)

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

Notes: (1) SG ΔP low RCS flow trip setpoints (i.e., ramp, step, and floor) selected to ensure breakers open at 2.5 seconds for a sheared shaft event. This is the only event that credits this trip function.
(2) Value in parenthesis represents the analysis setpoint for the existing plant thermal power output (i.e., 3876 MW_t).

Section 6.3.0.3.1 CPC Dynamic Signal Filter Coefficients

The CPC power and temperature signal filters, along with any associated correction factors, were evaluated to demonstrate their continued acceptability at PUR conditions. These signal filters were reviewed for the set of AOOs that provide the greatest challenge to the SAFDLs. Included in this evaluation was a power correction factor to compensate for pressure transients. Ensuring that the CPC signal filter responses (and corresponding correction factors) remain conservative will ensure a CPC reactor trip before the DNB and centerline-to-melt SAFDLs are violated.

Section 6.3.0.3.1.1 Increasing Power Signal Filters

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

To ensure that the CPC calibrated core heat flux conservatively leads actual core heat flux, a thermal margin calculation is performed to ensure that there is enough margin available at time of the trip to ensure that the SAFDLs are not violated. The CEAW events were initiated at different power levels along with conservative initial conditions (i.e., bank worth, MTC, trip response, etc). Section 6.3.4.2 details the CEAW event including the long-term NSSS response.

The subcritical and HZP CEA bank withdrawal events (see Section 6.3.4.1) rely on the HLPT and Plant Protection System (PPS) VOPT respectively. Therefore, these events were not used to tune the CPC neutron flux power signal filters at low powers.

The increasing LPD signal filter is not included in this analysis since none of the events would have a trip on LPD before receiving a trip on minimum DNBR.

Section 6.3.0.3.1.2 Increasing Reactor Coolant System Temperature Signal Filters

Increasing temperature, in the RCS, that is not driven by a power increase can be caused by decreased heat removal events. Several potential decreasing heat removal AOOs are part of the CPC's DBEs. A LOFW or a loss of load each decreases the rate of heat removal. The loss of load is the more adverse heatup scenario.

The CPC signal filters provide two RCS temperature indications, T_{CMIN} and T_{CMAX} that are used in the protective calculations.

T_{CMAX} is used as the core inlet temperature for the CPC's DNBR calculation. To be conservative, the value of T_{CMAX} should be larger than the actual core inlet temperature.

T_{CMIN} is used to adjust the raw neutron power signal for reactor vessel downcomer temperature shadowing. To be conservative, the value of T_{CMIN} should be less than the actual RCS temperature existing in the downcomer region of the reactor vessel at any given time in the transient.

The CPC's lead and lag temperature signal filters take the cold leg Resistance Temperature Detector (RTD) signals and adjust them to obtain the desired conservative direction of T_{CMIN} and T_{CMAX}. The loss of load was selected as the CPC's DBE used to examine these signal filters for transients resulting in RCS temperature increases.

Section 6.3.0.3.1.3 Decreasing Reactor Coolant System Temperature Signal Filters

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

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

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

Section 6.3.0.3.1.4 Decreasing Pressure Penalty

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

The RCS depressurization associated with the spray malfunction event was simulated using the CENTS code. Output from the CENTS code was then analyzed and a power equivalent of pressure decrease was calculated. In accordance with the existing reload design process, this power equivalent penalty is verified and applied in each reload.

Section 6.3.0.3.1.5 Results

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

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

The CPC transient signal filter analysis verifies that the CPC's adjusted process parameters are conservative for the expected values for a given transient event. The CPC coefficients are adjusted as necessary to assure the CPC's action prevents SAFDL violation during these AOOs. This analysis included parametric studies on RCS flow and tube plugging to determine the limiting values of these inputs.

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

Section 6.3.0.4 Engineered Safety Features

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

Table 6.3-4
ESF Analytical Setpoints

Parameter	Units	Analytical Setpoint	Response Time
Safety Injection Actuation Signal (SIAS) on high containment pressure: normal harsh	psig	 N/A 5.0	30 sec
SIAS on low pressurizer pressure: normal harsh	psia	 1750 1670	30 sec
MSIS on LSGP: normal harsh	psia	 915 (850) 875 (810) (1)	5.6 sec
MSIS on high SG level: normal harsh	%NR	 97.0 99.9	5.6 sec
MSIS on high containment pressure: normal harsh	psig	N/A 5.0	5.6 sec
Auxiliary Feedwater Actuation Signal (AFAS) on low SG level: normal harsh	%WR	 20.0 10.0	 46 sec (LOP) 23 sec (no LOP)
AFAS lock-out on SG ΔP: normal harsh	psid	 240 270	16 sec

Note: (1) Value in parenthesis represents the analysis setpoint for the existing plant configuration (i.e., 3876 MW_t). The LSGP limits in Technical Specification 3.3.1 and 3.3.2 (Reference 6-2), currently 890 psia, will be changed to 955 psia with the implementation of the PUR.

Section 6.3.1 Increase In Heat Removal By The Secondary System

Section 6.3.1.1 Decrease in Feedwater Temperature

As described in UFSAR Section 15.1.1, a decrease in FW temperature may result from a loss of one or more FW heaters. Loss of one of two FW heater drain tank pumps interrupts the steam extraction from the high pressure turbine to one of two parallel FW trains and results in the loss of three of six high pressure heaters. No other single failure would result in the loss of more heaters.

The decrease in FW temperature event is classified as an AOO. This event was reviewed for the impacts of PUR and remains bounded by the limiting increase in the main steam flow event, SBCS misoperation, discussed in Section 6.3.1.3.

Section 6.3.1.2 Increase in Feedwater Flow

As described in UFSAR Section 15.1.2, an increase in the opening of a Feedwater Control Valve (FWCV) or an increase in FW pump speed causes an increase in FW flow. The maximum possible increase at full power is approximately 25% above nominal for the normal FW system.

The increase in FW flow event is classified as an AOO. This event was reviewed for the impacts of PUR and remains bounded by the limiting increase in main steam flow event, SBCS misoperation, discussed in Section 6.3.1.3.

Section 6.3.1.3 Increased Main Steam Flow

Section 6.3.1.3.1 Identification of Event and Causes

As described in UFSAR Section 15.1.3, inadvertent increased opening of a TAV or a malfunction of the SBCS can cause an increase in main steam flow. These events will result in up to an 88% increase of the nominal full power steam flowrate (eight TBVs at 11% steam flow each). This increase in main steam flow event bounds the inadvertent opening of a single TBV or ADV.

An increase in main steam flow causes a decrease in RCS temperature, an increase in core power and heat flux, and a decrease in RCS and SG pressures. These conditions will initiate a reactor trip on high reactor power, low RCS pressure, or LSGP, as required. If the transient were to result in an approach to SAFDLs, trip signals generated by the CPC would assure that DNBR or LPD limits are not exceeded.

Section 6.3.1.3.2 Acceptance Criteria

The increased main steam flow event is classified as an AOO. As defined in the SRP Section 15.1.1, the specific acceptance criteria for this event are:

a. Pressures in the RCS and main steam system should be maintained below 110% of the design.
b. Fuel cladding integrity should be maintained by ensuring that Acceptance Criterion 1 of SRP Section 4.4 is satisfied throughout the transient.
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Section 6.3.1.3.3 Description of Analysis

The NSSS response to several increased main steam flow events were simulated using the CENTS code. The initial and transient DNBR was calculated using the CETOP-D code which uses the CE-1 Critical Heat Flux (CHF) correlation.

The limiting increase in main steam flow AOO is the malfunction of the SBCS. This event results in 88% excess steam demand, and in the presence of a negative MTC, produces a rapid power excursion. The limiting increase in main steam flow event with infrequent classification is detailed in Section 6.3.1.4.

Both the existing configuration and PUR were analyzed to compare the NSSS response to the SBCS malfunction event. Input parameters and initial conditions were selected to maximize the DNBR degradation and demonstrate that fuel cladding integrity is maintained throughout the event.

Section 6.3.1.3.3.1 Transient Simulation

The system was initialized at a POL using the most limiting initial parameters. At time equal zero, a malfunction of the SBCS results in a quick open of all eight TBVs. The mismatch of main steam demand and core power results in a decrease in core inlet temperature. In the presence of a negative MTC, this leads to a rapid power excursion. The excess steam demand is terminated by manual closure of the MSIVs.

A CPC VOPT produces a reactor trip and prevents the minimum DNBR from violating the SAFDL. The CPC FORTRAN code determined the timing of the VOPT.

Section 6.3.1.3.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-5 lists the initial conditions used for the SBCS malfunction event.

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 operation including the instrument uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.

2. There is no operator action for the first 30 minutes of the event.

Table 6.3-5
Parameters Used for SBCS Malfunction Event

PARAMETER	Value	
	$3876\ MW_t$	$3990\ MW_t$
Initial core power (% of rated)	100	100
Initial core inlet temperature (°F)	562	568
Initial pressurizer pressure (psia)	2315	2315
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
Fuel Temperature Coefficient (FTC)	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip - Worst Rod Stuck Out (WRSO) (%$\Delta\rho$)	-8.0	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6100	6100
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.1.3.5 Results

Table 6.3-6 presents a sequence of events that occur following the SBCS malfunction until operator action is initiated at 30 minutes. Figure 6.3-1 through Figure 6.3-12 present the behavior of NSSS parameters following the event.

The increased main steam flow caused by the SBCS malfunction leads to a reduction in core inlet temperature. The resulting power excursion causes a reactor trip on CPC VOPT that terminates the DNBR degradation. Closure of the MSIVs terminates main steam flow. The minimum DNBR remains above the SAFDL limit throughout the transient. RCS and secondary system pressures decrease because of the increased main steam flow and both remain below 110% of design.

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 DNBR degradation experienced during this transient is identical for the two configurations.

Table 6.3-6
Sequence of Events for SBCS Malfunction Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	SBCS malfunction, all 8 TBVs quick open	---	---
4.75	4.95	CPC VOPT setpoint (% of power)	110	110
5.50	5.70	Reactor trip breakers open	---	---
6.10	6.30	Scram CEAs begin falling	---	---
6.50	6.70	Turbine trip occurs	---	---
6.60	6.80	Minimum DNBR	1.40	1.40
23.9	25.0	SG pressure reaches MSIS setpoint (psia)	850	915
26.4	24.3	RCS pressure reaches SIAS setpoint (psia)	1750	1750
29.5	30.6	MSIVs close	---	---
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.1.3.6 Conclusions

For the limiting increase in main steam flow AOO, SBCS malfunction, all acceptance criteria are met. The peak primary and secondary pressures remain below 110% of design thus ensuring the integrity of the RCS and the main steam system. The minimum DNBR remains above the safety analysis limit ensuring fuel cladding integrity. The infrequent increases in main steam flow events are detailed in Section 6.3.1.4.

Figure 6.3-1
SBCS Malfunction Event - Core Power vs. Time

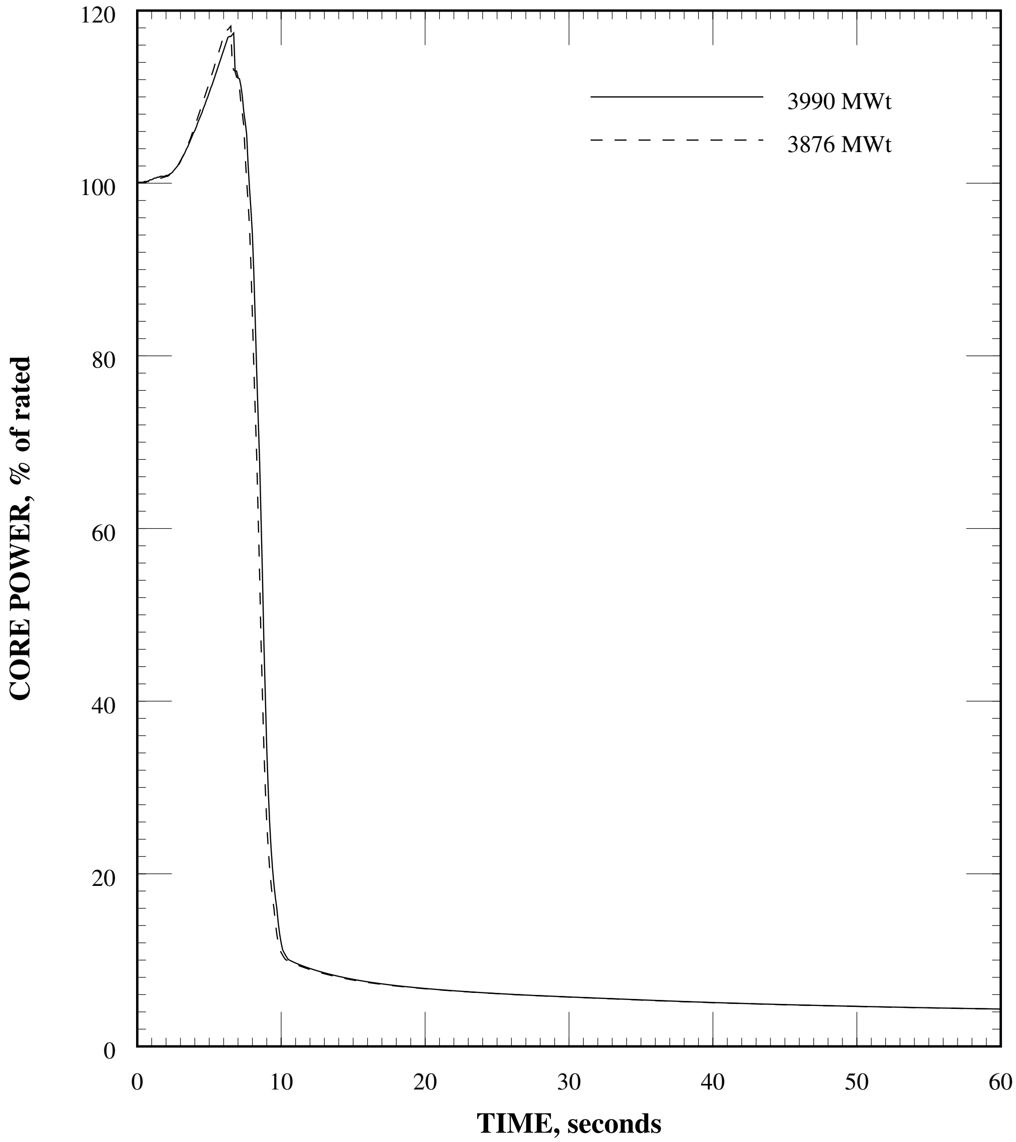


Figure 6.3-2
SBCS Malfunction Event - Core Average Heat Flux vs. Time

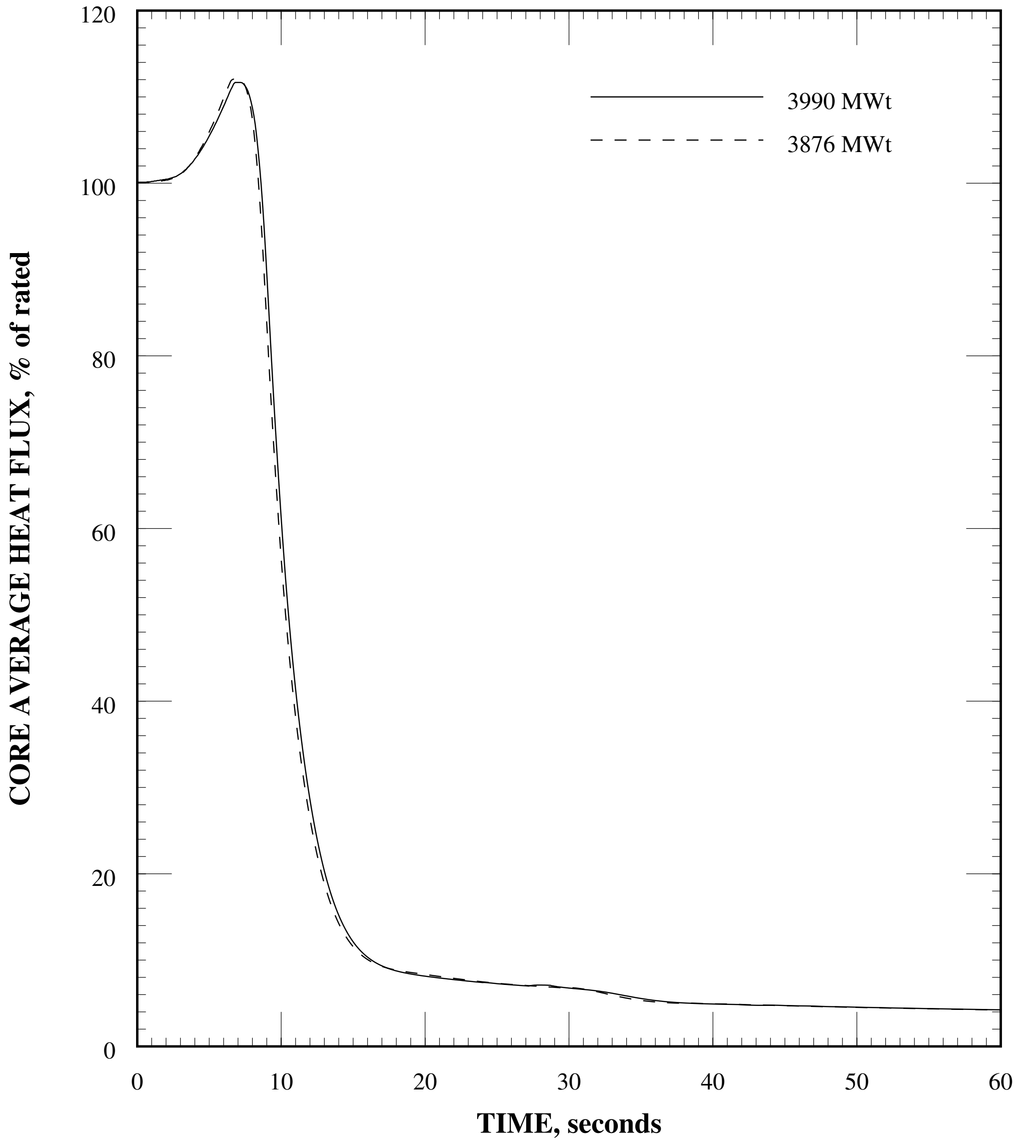


Figure 6.3-3
SBCS Malfunction Event - Minimum DNBR vs. Time

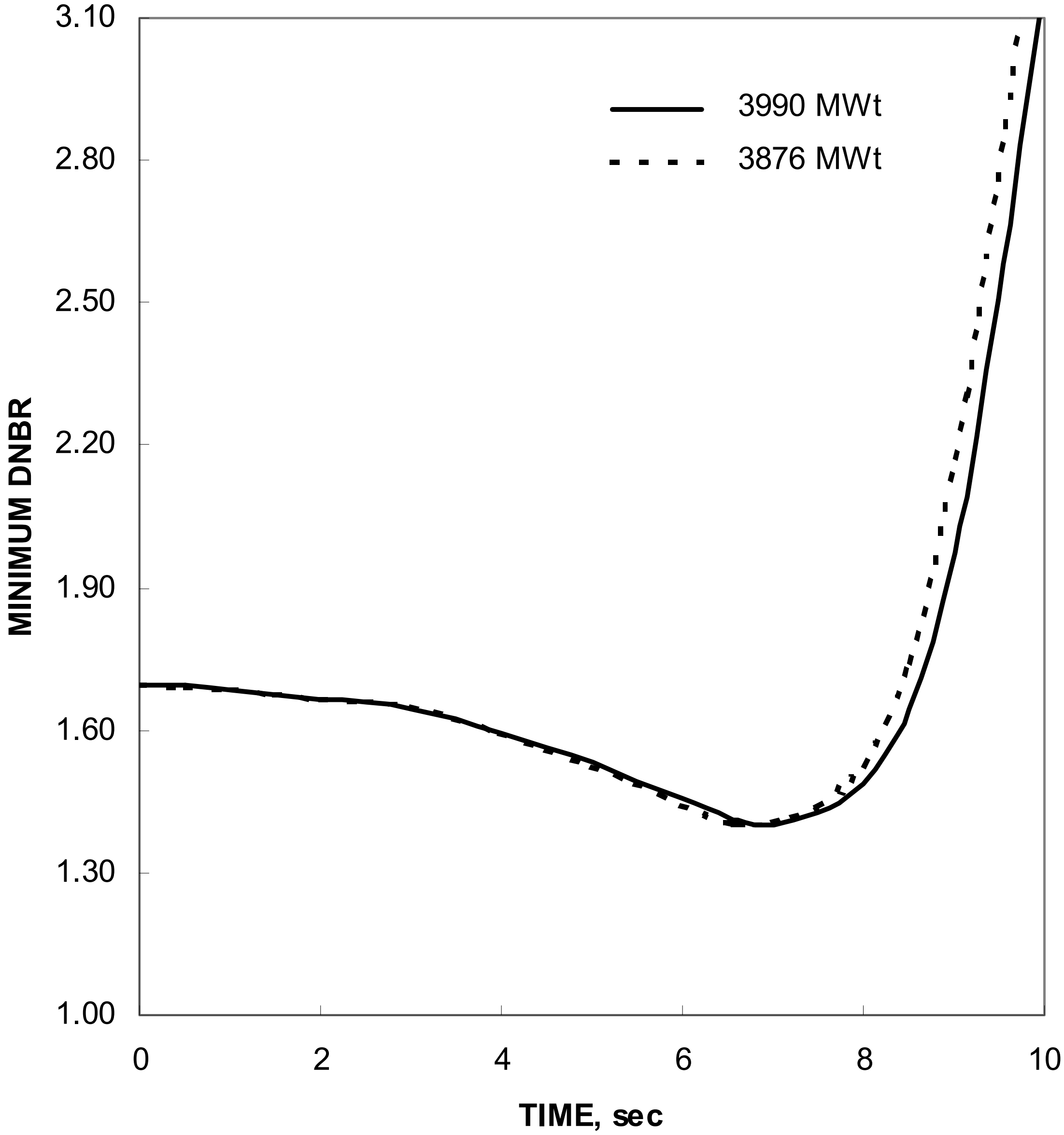


Figure 6.3-4
SBCS Malfunction Event - Core Reactivity vs. Time

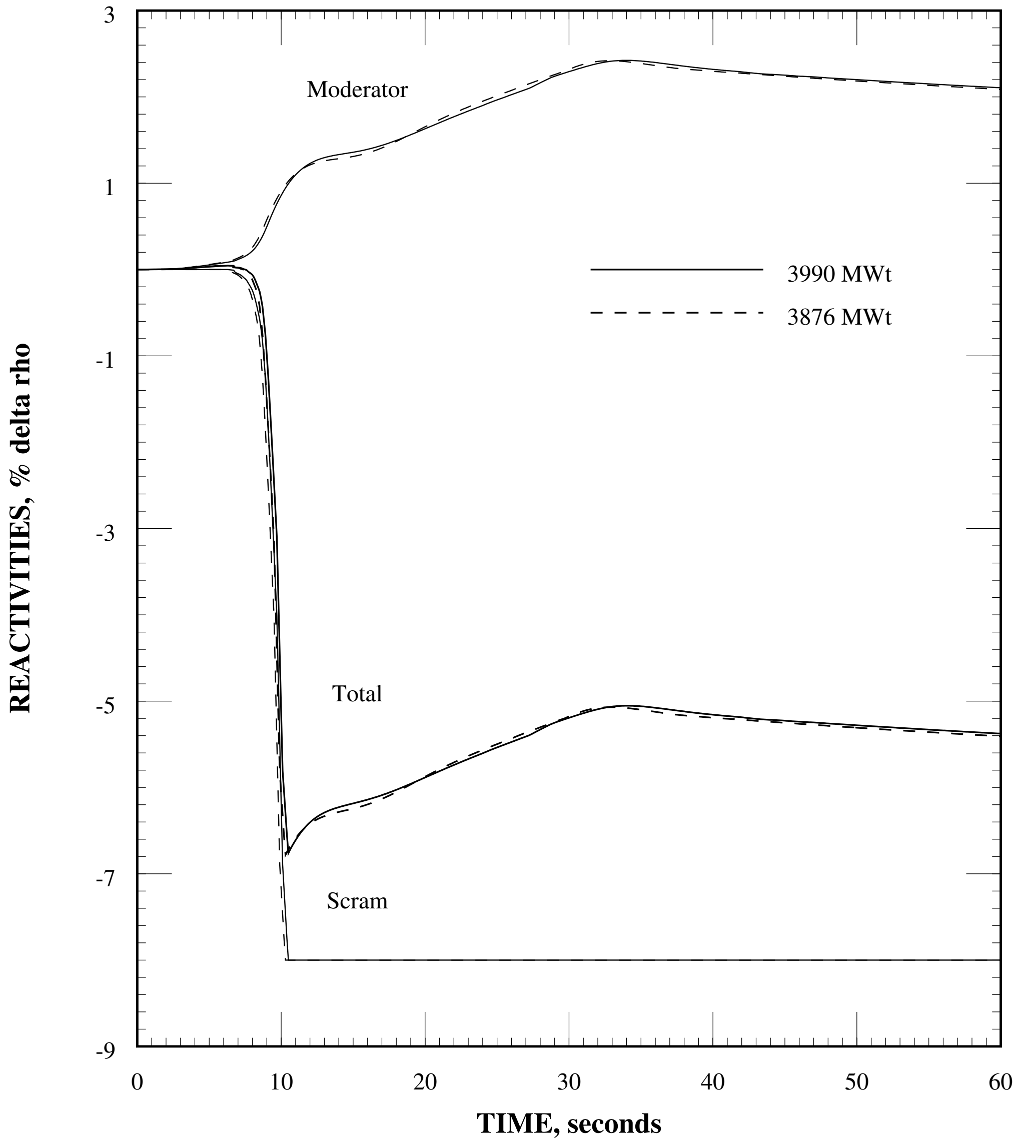


Figure 6.3-5
SBCS Malfunction Event - RCS Temperatures vs. Time

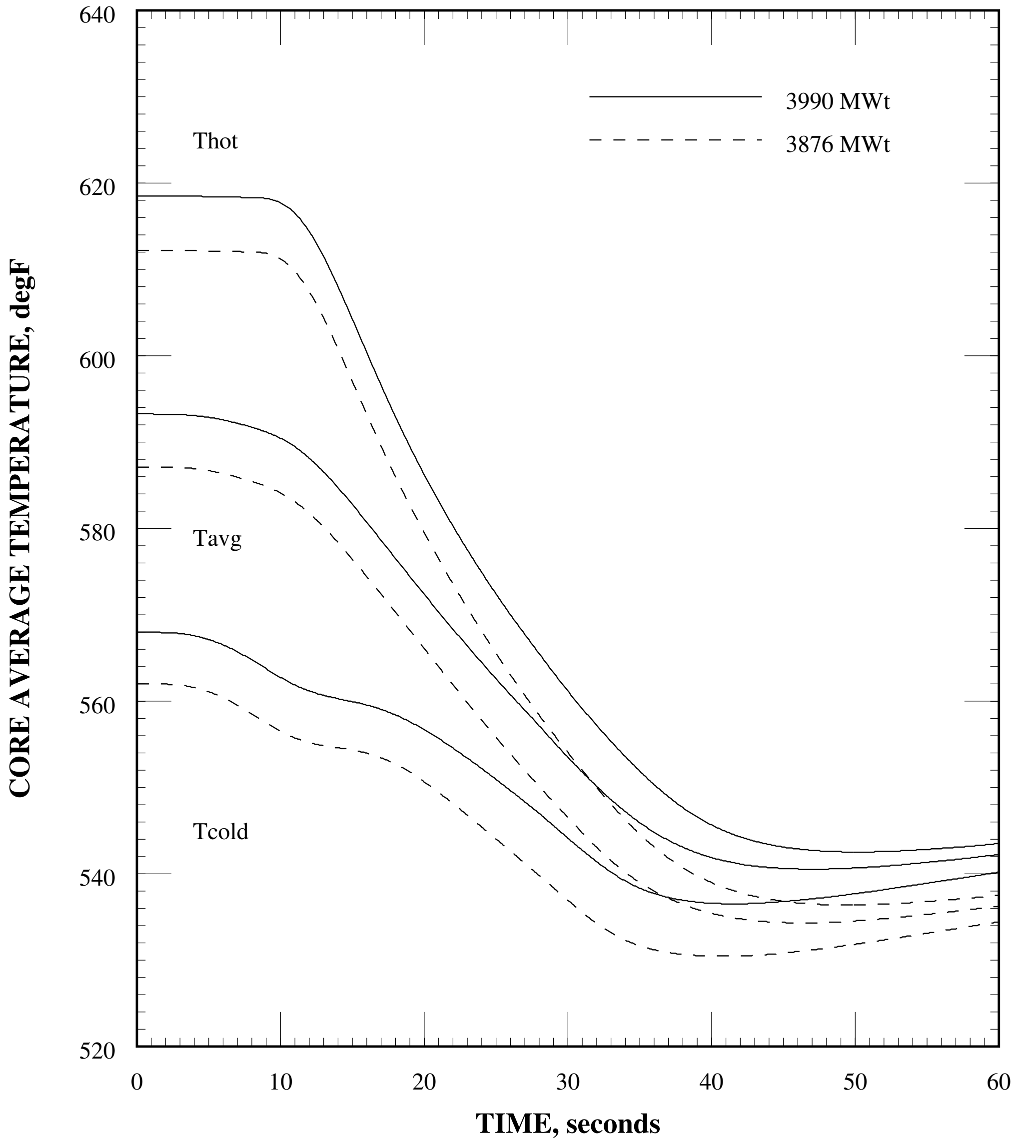


Figure 6.3-6
SBCS Malfunction Event - RCS Pressure vs. Time

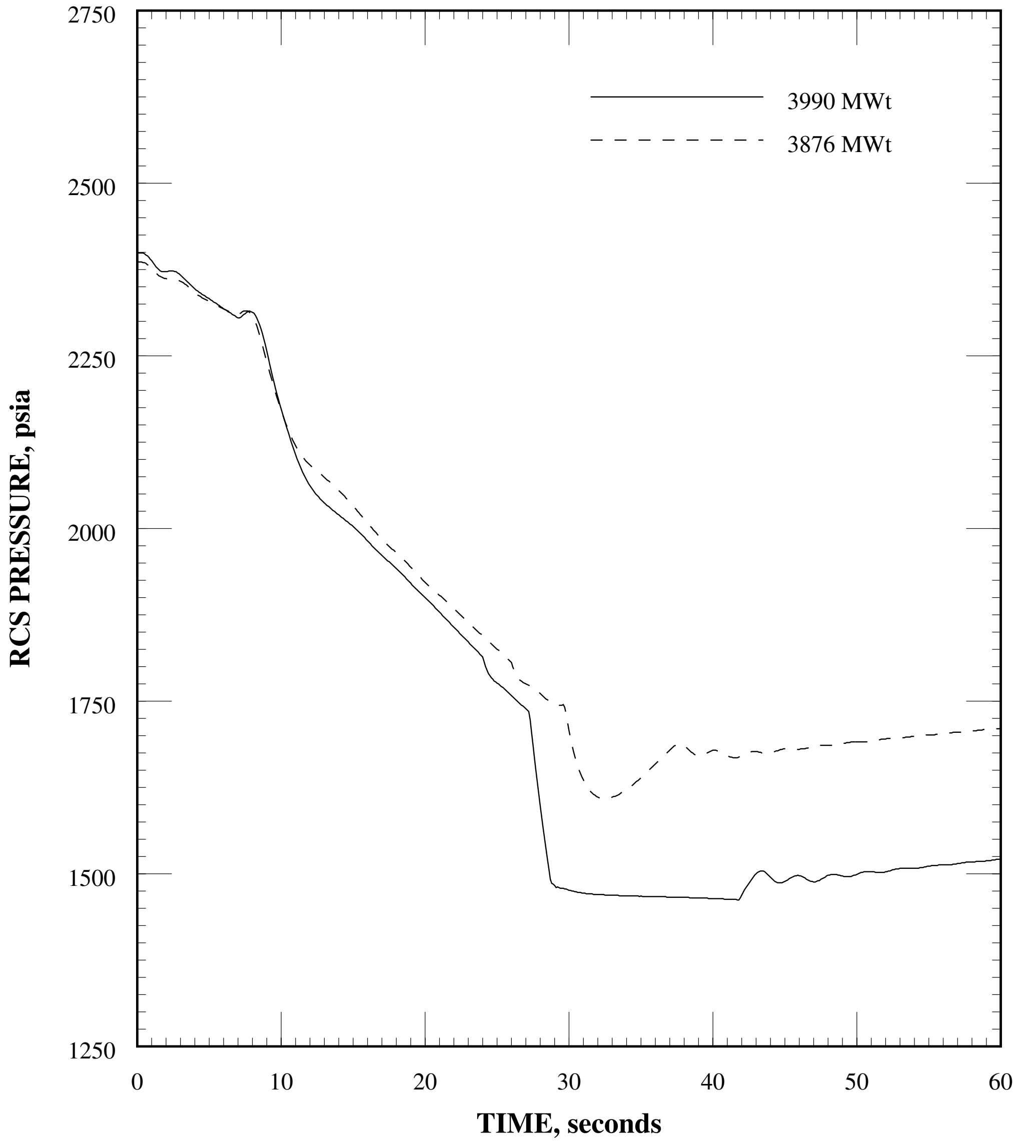


Figure 6.3-7
SBCS Malfunction Event - Pressurizer Water Volume vs. Time

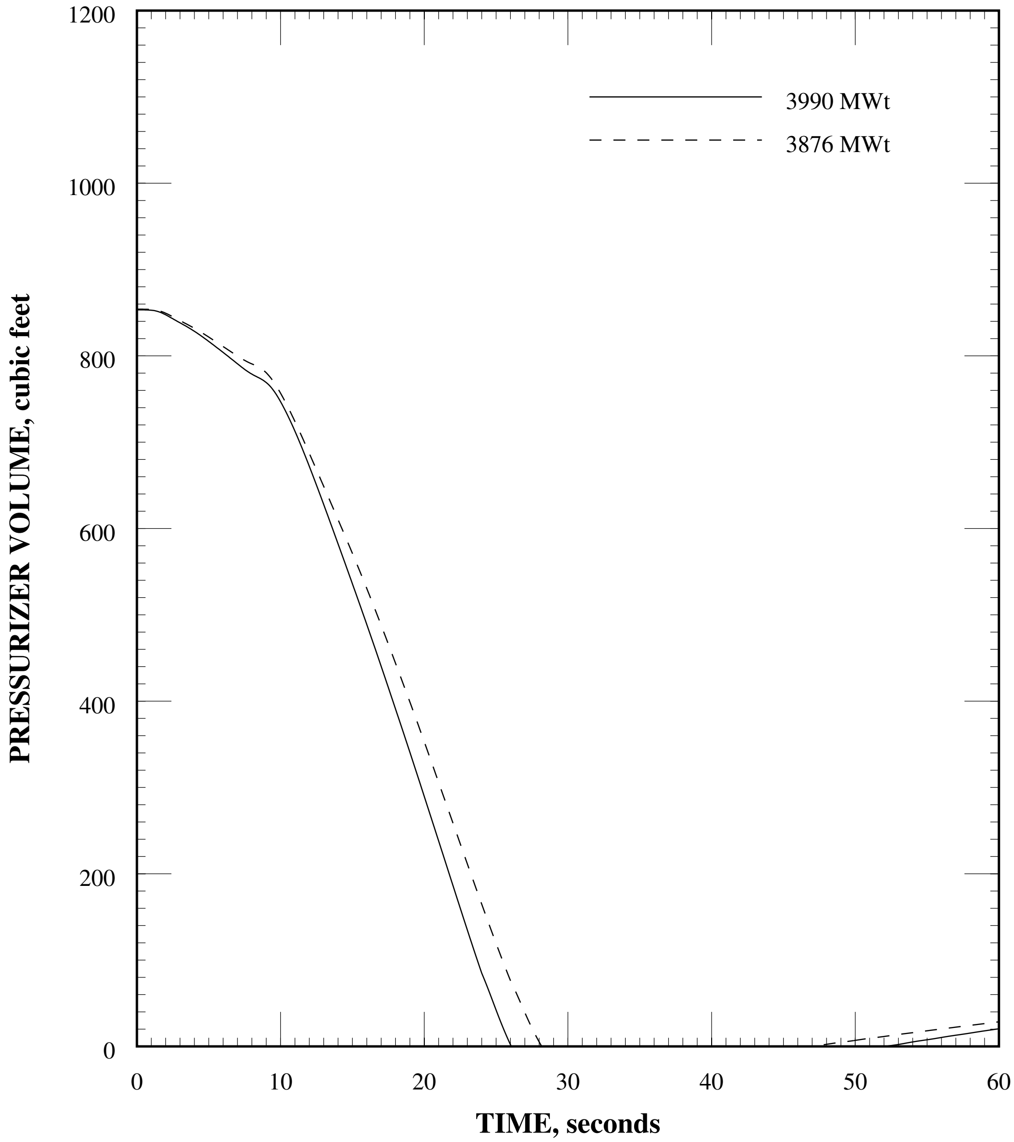


Figure 6.3-8
SBCS Malfunction Event - Steam Flow vs. Time

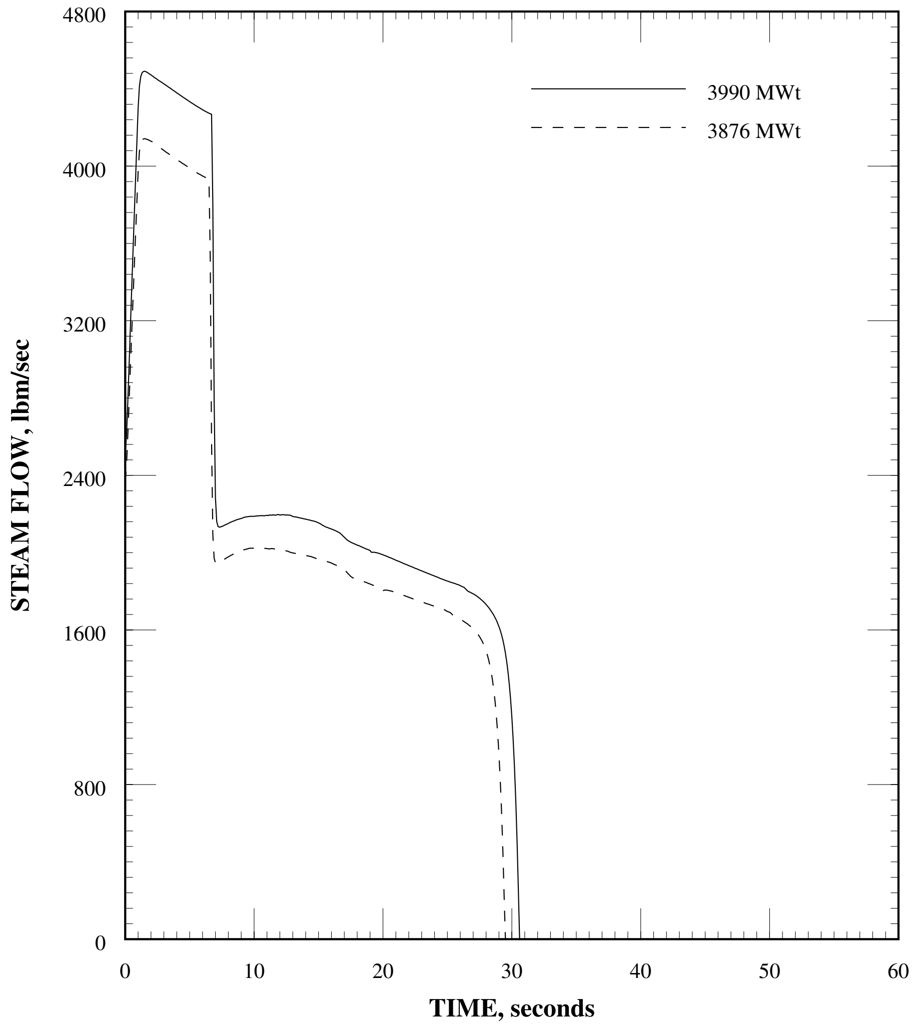


Figure 6.3-9
SBCS Malfunction Event - SG Pressure vs. Time

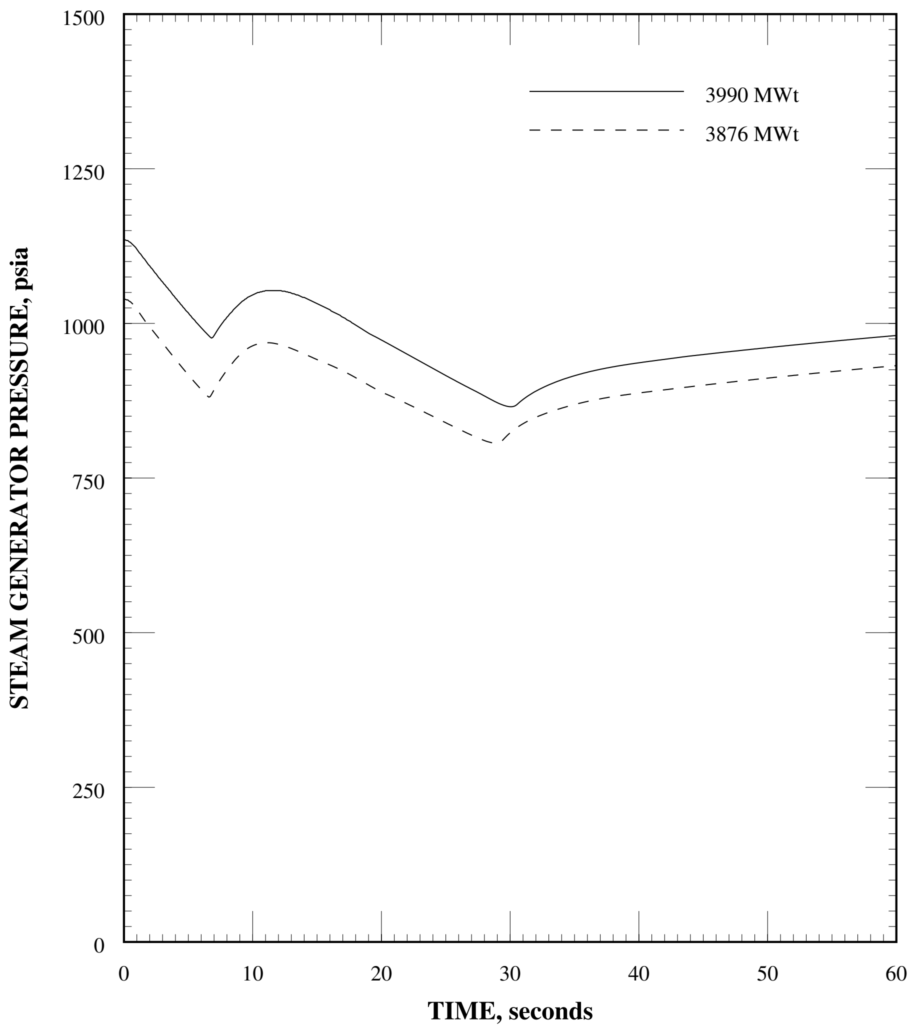


Figure 6.3-10
SBCS Malfunction Event - SG Level vs. Time

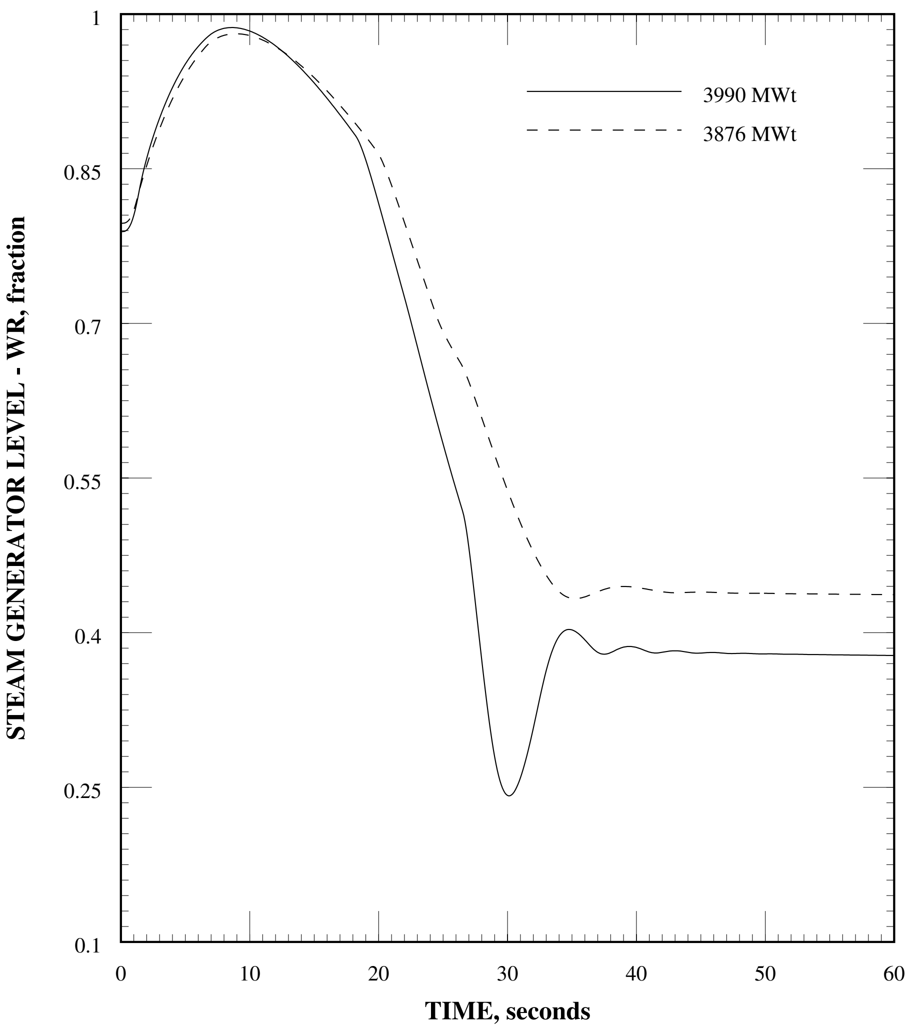


Figure 6.3-11
SBCS Malfunction Event - SG Liquid Mass vs. Time

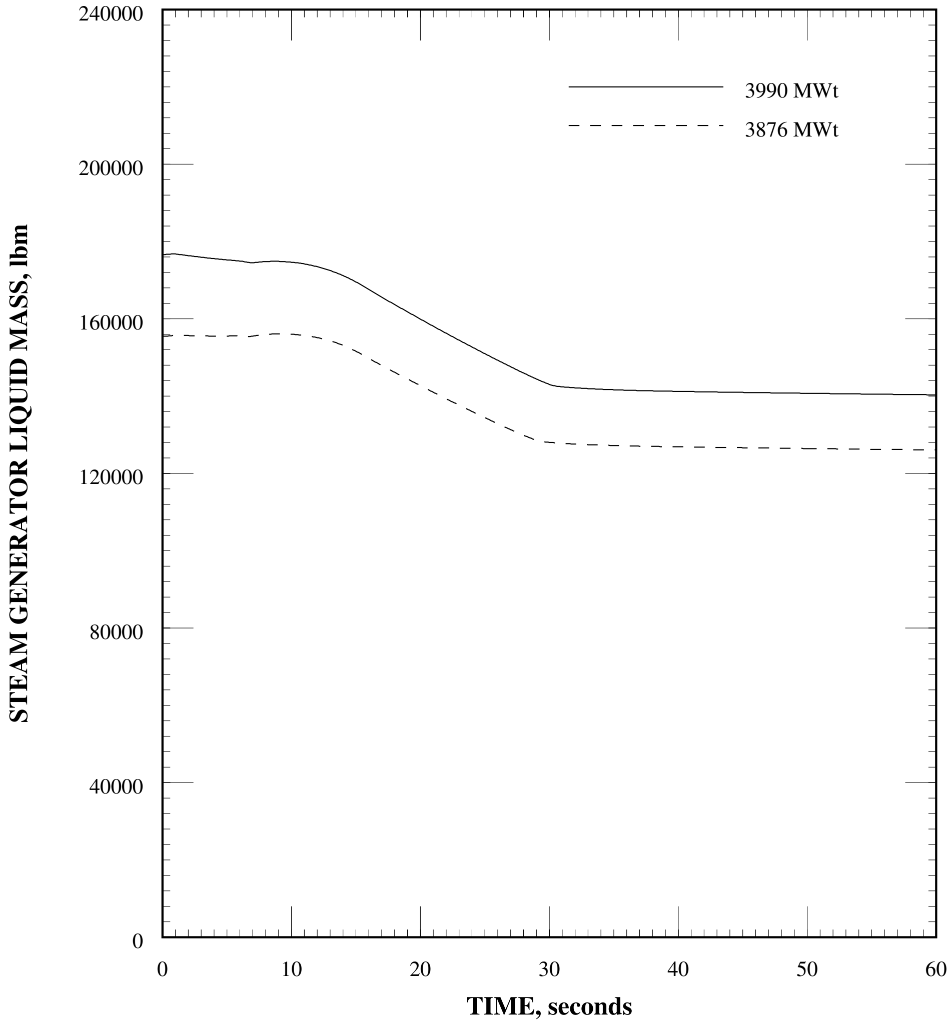
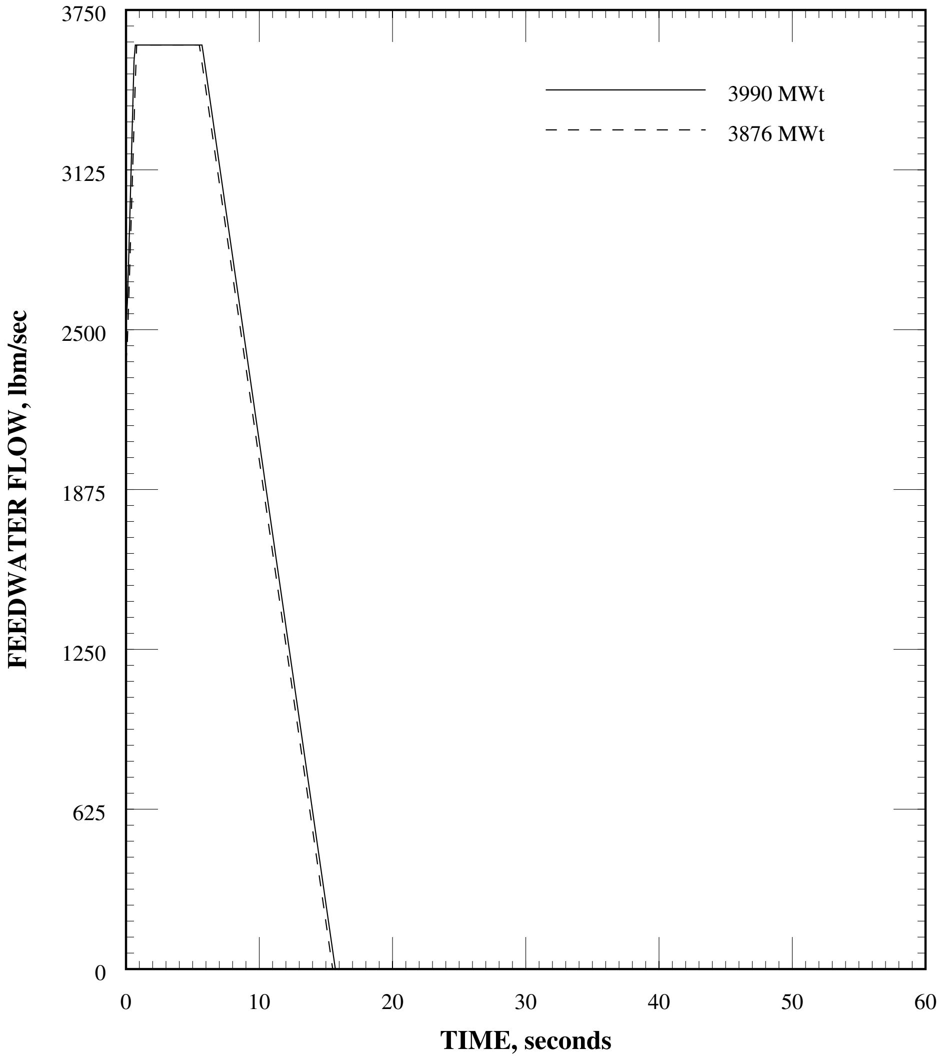


Figure 6.3-12
SBCS Malfunction Event - Main FW Flow vs. Time



Section 6.3.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Section 6.3.1.4.1 Identification of Event and Causes

As described in UFSAR Section 15.1.4, an operator error or a failure of the control system may inadvertently open an ADV or a TBV. The event will result in a maximum 11% increase over the nominal full power steam flowrate. This type of increased main steam flow event is bounded by the SBCS malfunction event detailed in Section 6.3.1.3.

The IOSGADV with LOP following turbine trip has been classified as an infrequent event. Previous analyses have attempted to simplify the simulation of this transient. Conservatively ignoring a potential CPC VOPT and LSGP trips, as well as initial thermal margin in excess of the degradation associated with an 11% excess load, these analyses assumed that the hot channel DNBR would stabilize immediately above the SAFDL, 30 seconds after the IOSGADV event. At 45 seconds, a turbine trip followed by the LOP occurred. These previous analyses conservatively force the hot channel DNBR to the SAFDL at the onset of the LOP and simply model the loss of forced flow portion of the event.

This evaluation separates the IOSGADV + LOP event from the Loss of Flow (LOF) from SAFDL methodology and presents a more realistic IOSGADV excess steam demand event with a LOP following turbine trip. The limiting infrequent event with single failure (i.e., loss of forced flow from SAFDL) is presented in Section 6.3.8.

The radiological consequences of IOSGADV + LOP are presented in Section 6.4.1.1.

Section 6.3.1.4.2 Acceptance Criteria

As defined in the SRP Section 15.1.4, the specific acceptance criteria for this event are:

a. Pressure in the RCS and main steam system should be maintained below 110% of the design.
b. Fuel cladding integrity should be maintained by ensuring that Acceptance Criterion 1 of SRP Section 4.4 is satisfied throughout the transient.
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Section 6.3.1.4.3 Description of Analysis

The NSSS response to the IOSGADV + LOP event was simulated using the CENTS code. The initial and transient DNBR values were calculated using the CETOP-D code that uses the CE-1 CHF correlation. At the time of minimum DNBR, a more accurate prediction of the DNBR was calculated using the more detailed TORC code.

Both the existing configuration and PUR were analyzed in order to compare the NSSS response to the IOSGADV + LOP event. Input parameters and initial conditions were selected to maximize the DNBR degradation.

Section 6.3.1.4.3.1 Transient Simulation

The system was initialized at a POL using the most limiting initial parameters. At time equal zero an operator action or system malfunction results in an IOSGADV. The increase in main steam demand results in a decrease in core inlet temperature and decrease in SG pressure. In the presence of a large negative MTC, this leads to a power excursion and produces a CPC VOPT. The presence of a negative MTC leads to a gradual power increase and an eventual LSGP trip.

The CPC FORTRAN code determined the timing of the VOPT. The timing of the LSGP trip was determined by CENTS.

The methodology for modeling a LOP following a turbine trip is presented in UFSAR Section 15.0.2.4. The UFSAR demonstrates that a delay of at least 3 seconds exists between the turbine trip and the LOP. This evaluation has conservatively assumed a coincident reactor/turbine trip and LOP.

The RCP coastdown further degrades DNBR, especially during the Control Element Drive Mechanism (CEDM) holding coil decay. DNBR degradation is terminated when the mitigating effects of the scram CEA insertion dominate the flow coastdown.

Section 6.3.1.4.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-7 contains the initial conditions used for the IOSGADV + LOP event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. There is no operator action for the first 30 minutes of the event. At 30 minutes, the operators are credited with closing the TBV or ADV that inadvertently opened and initiated the event.

Table 6.3-7
Parameters Used for the IOSGADV + LOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	100	100
Initial core inlet temperature (°F)	562	568
Initial pressurizer pressure (psia)	2310	2310
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-0.2E-04	-0.2E-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)	600	600
Plugged SG tubes (% of tubes/SG)	0	0
Single Failure	none	none
LOP	yes	yes

Section 6.3.1.4.5 Results

Table 6.3-8 presents a sequence of events that occur following the IOSGADV + LOP event until operator action is initiated at 30 minutes. Figure 6.3-13 through Figure 6.3-25 presents the behavior of NSSS parameters following the IOSGADV + LOP event.

The increased main steam flow caused by the IOSGADV leads to a reduction in core inlet temperature and SG pressure. Depending on core burnup, a CPC VOPT due to an MTC driven power excursion or an eventual LSGP signal provides a reactor trip. A LOP concurrent with the reactor/turbine trip yields further DNBR degradation. The minimum DNBR remains above the SAFDL limit throughout the transient. RCS and secondary system pressures decrease because of the increased main steam flow and remain below 110% of design.

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 impact of the different LSGP trip/MSIS setpoint is reflected as a shift in the timing of the NSSS response.

Radiological consequences for this event are presented in Section 6.4.1.1.

Table 6.3-8
Sequence of Events for IOSGADV + LOP Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	Inadvertent opening of SG #1 ADV	---	---
150.1	179.9	SG pressure reaches MSIS/trip setpoint (psia)	850	915
151.2	181.0	Reactor trip breakers open	---	---
151.2	181.0	Turbine trip occurs	---	---
151.2	181.0	LOP occurs	---	---
151.8	181.6	Scram CEAs begin falling	---	---
153.3	183.1	Minimum DNBR	1.41	1.37
155.7	185.5	MSIVs closed	---	---
360.8	341.9	SG #2 MSSV bank 1 begins cycling open/close (psia)	1303	1303
464.3	430.3	RCS pressure reaches SIAS setpoint (psia)	1750	1750
965	1088	SG #1 empties	---	---
1800	1800	Operator manually closes ADV	---	---
1800	1800	Operator initiate cooldown (min)	30	30

Section 6.3.1.4.6 Conclusions

For the IOSGADV + LOP event, all acceptance criteria are met. The peak primary and secondary pressures remained below 110% of design, thus ensuring the integrity of the RCS and the main steam system. Offsite doses remained below the acceptance criteria for this category of event. Specifically, a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body).

Figure 6.3-13
IOSGADV + LOP Event - Core Power vs. Time

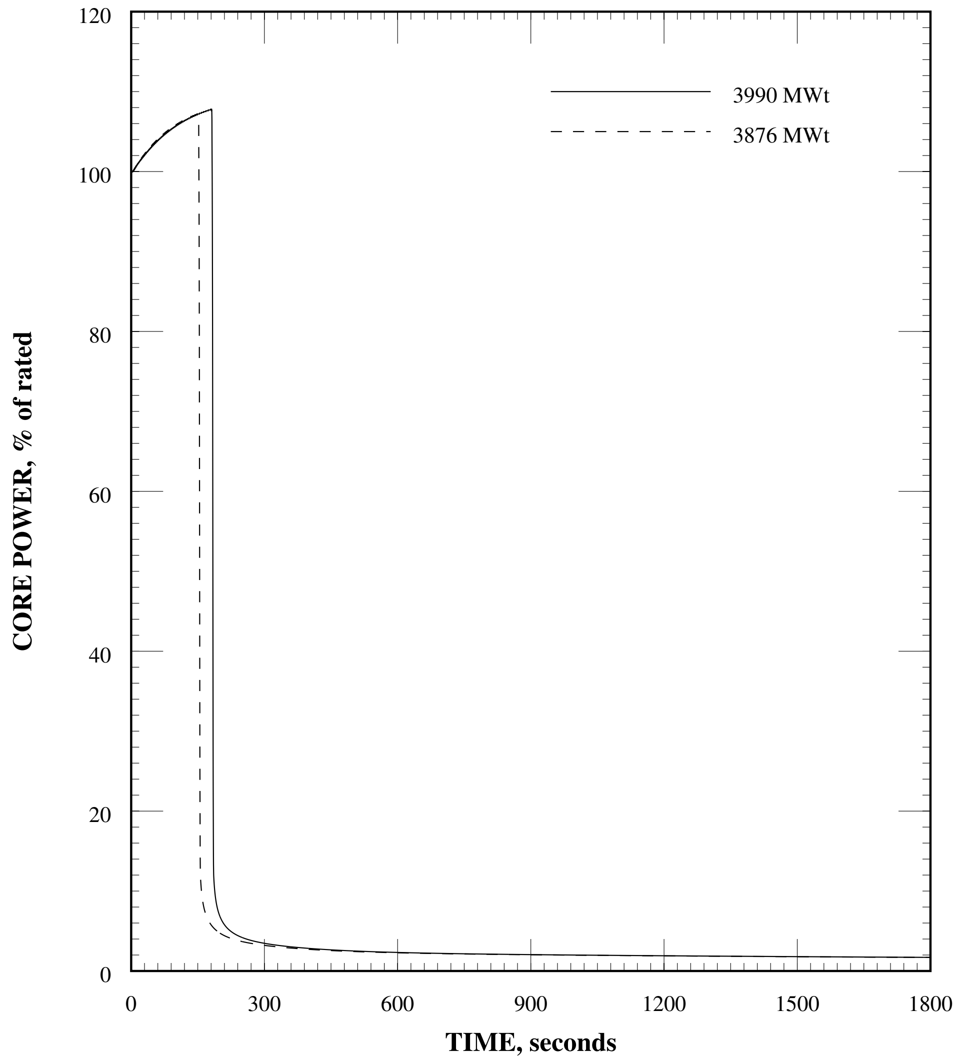


Figure 6.3-14
IOSGADV + LOP Event - Core Average Heat Flux vs. Time

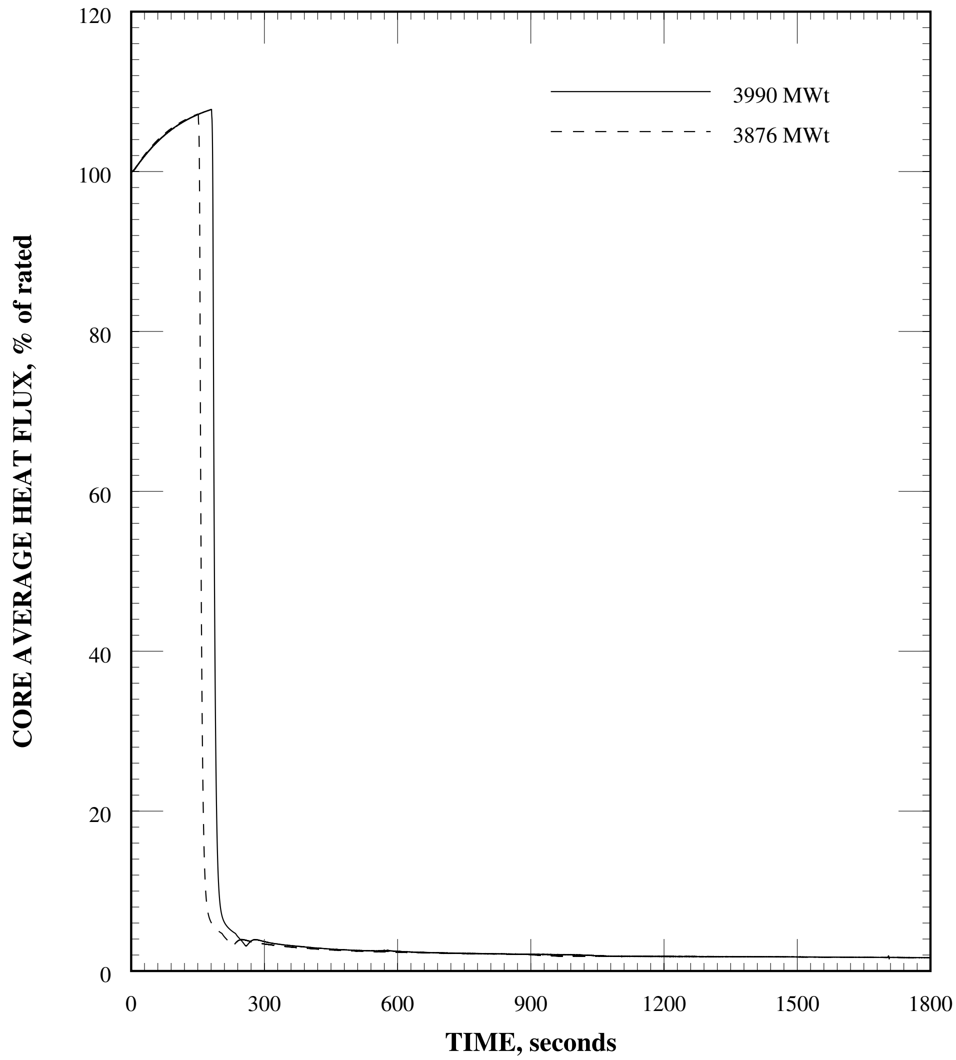


Figure 6.3-15
IOSGADV + LOP Event - Minimum DNBR vs. Time

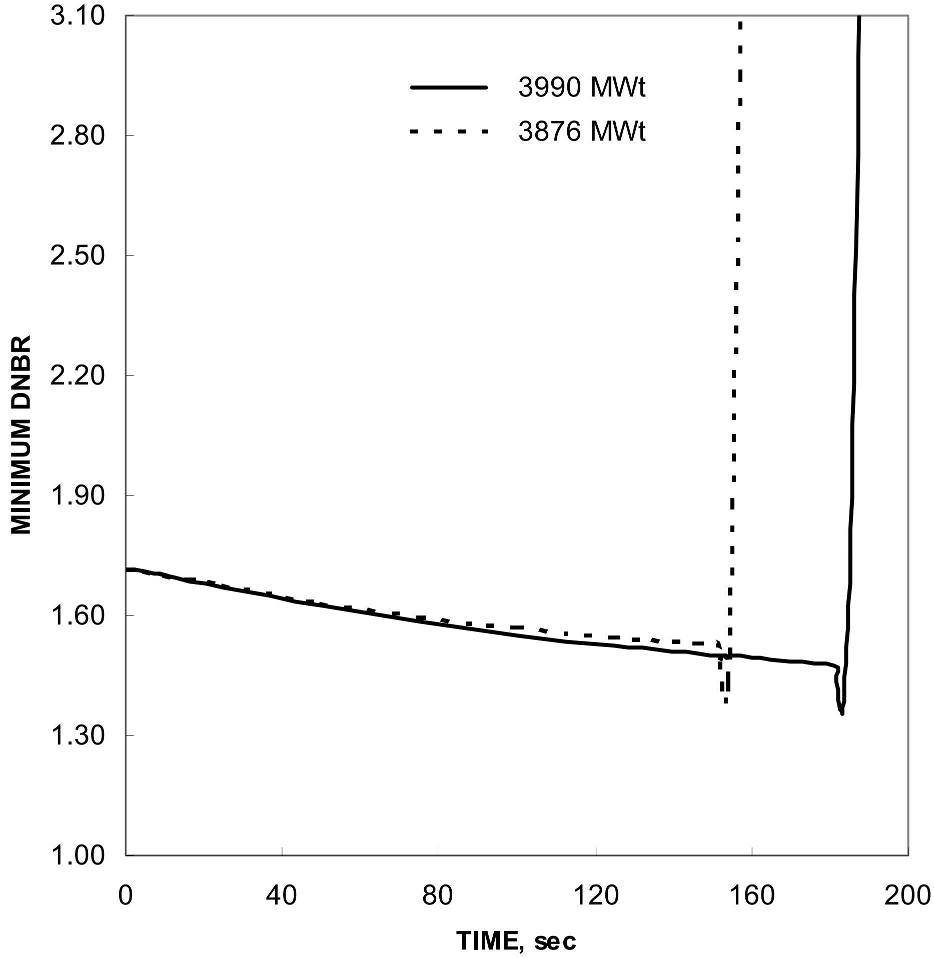


Figure 6.3-16
IOSGADV + LOP Event - Core Reactivity vs. Time

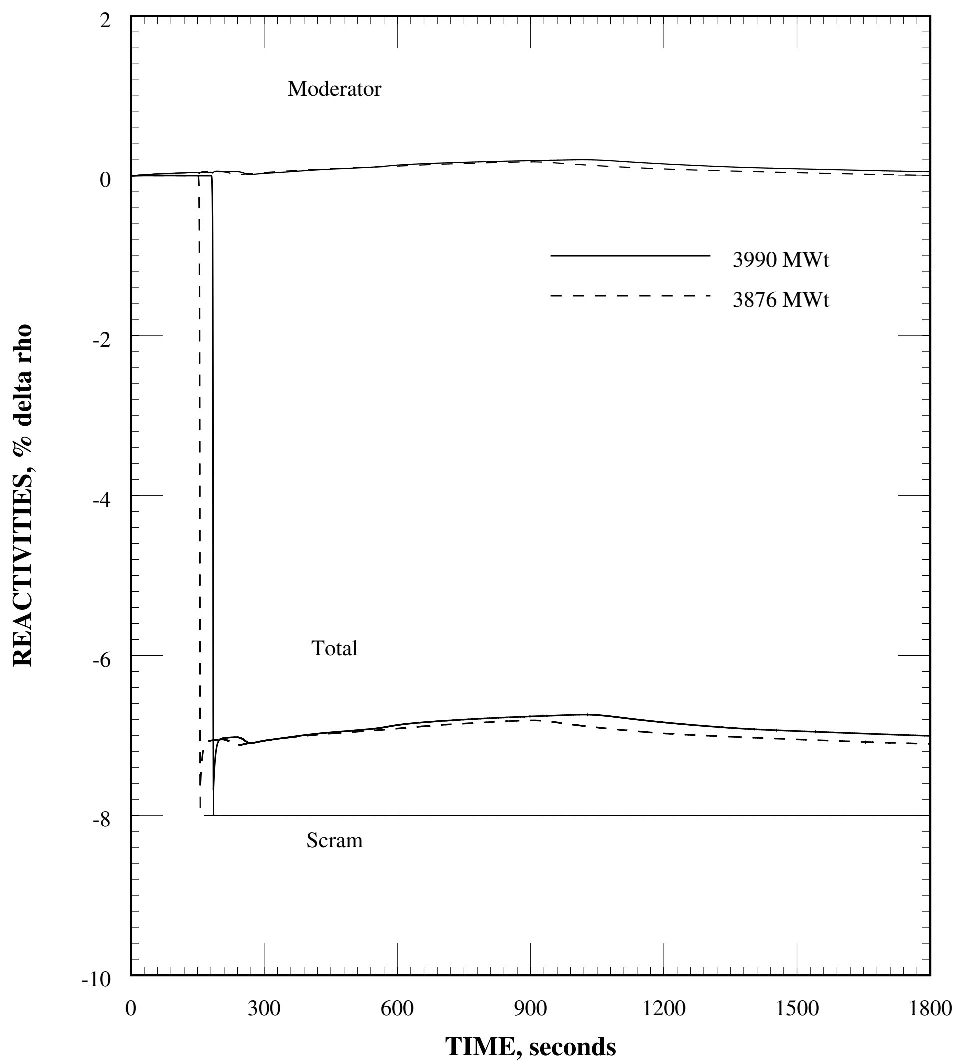


Figure 6.3-17
IOSGADV + LOP Event - RCS Temperature vs. Time
(Sheet 1 of 3 - Core Average Temperature)

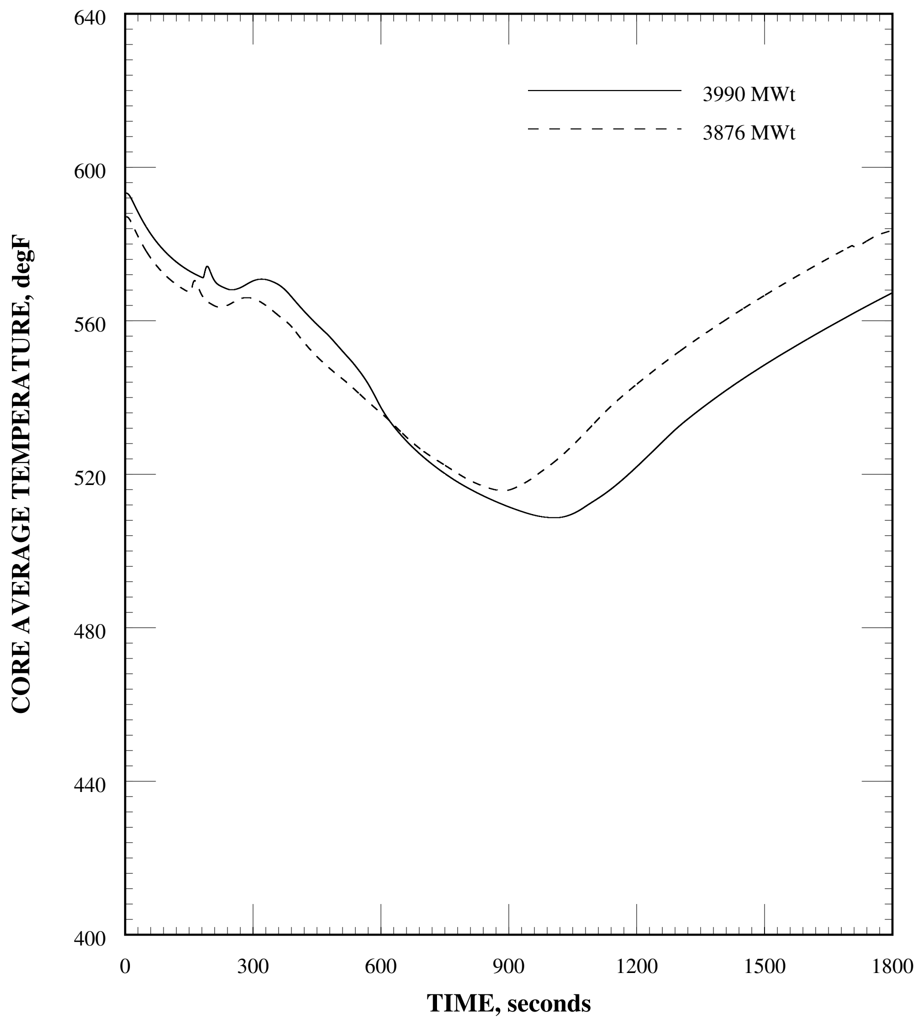


Figure 6.3-17
IOSGADV + LOP Event - RCS Temperature vs. Time

(Sheet 2 of 3 - Cold Leg Loop Temperature)

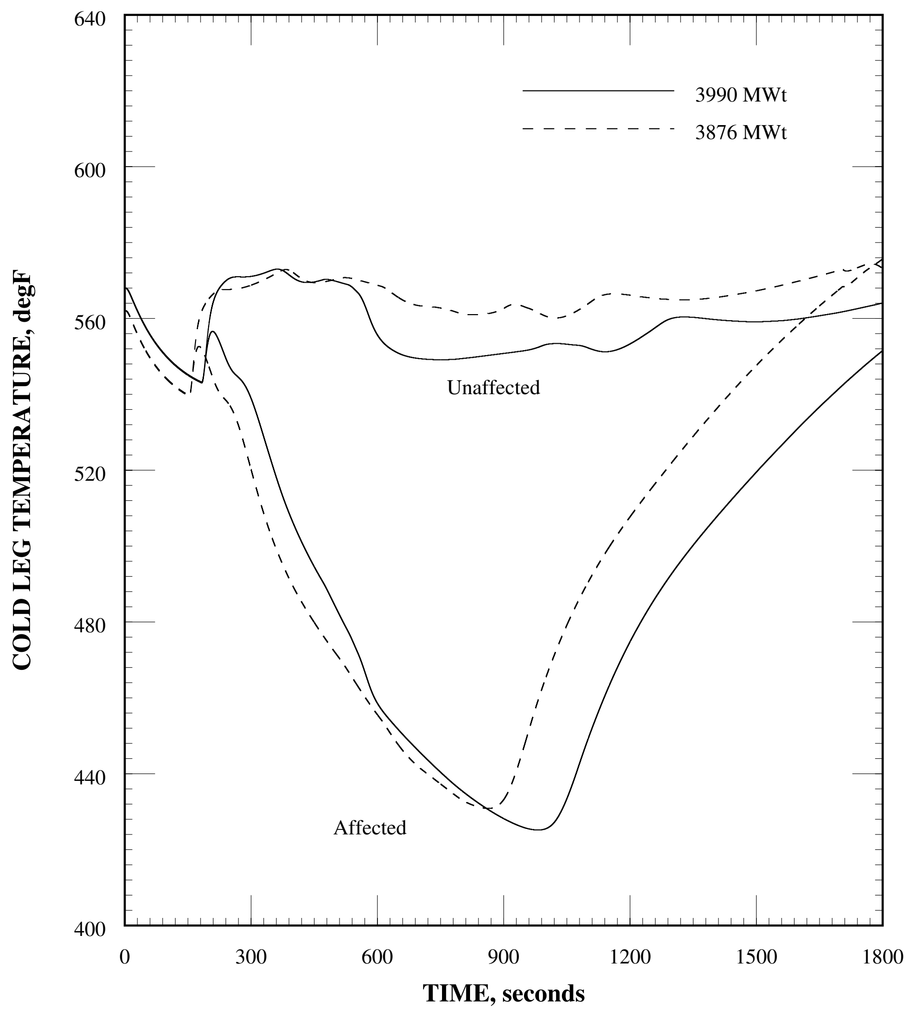


Figure 6.3-17
IOSGADV + LOP Event - RCS Temperature vs. Time

(Sheet 3 of 3 - Hot Leg Loop Temperature)

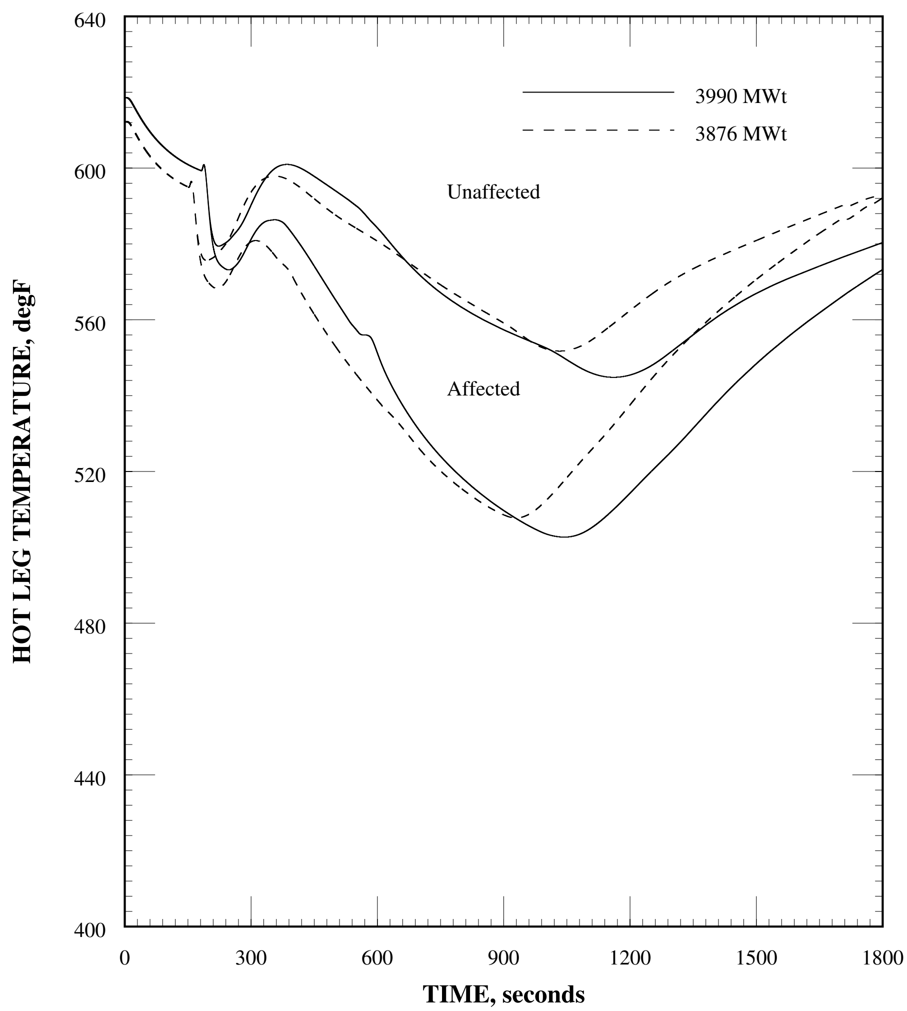


Figure 6.3-18
IOSGADV + LOP Event - RCS Pressure vs. Time

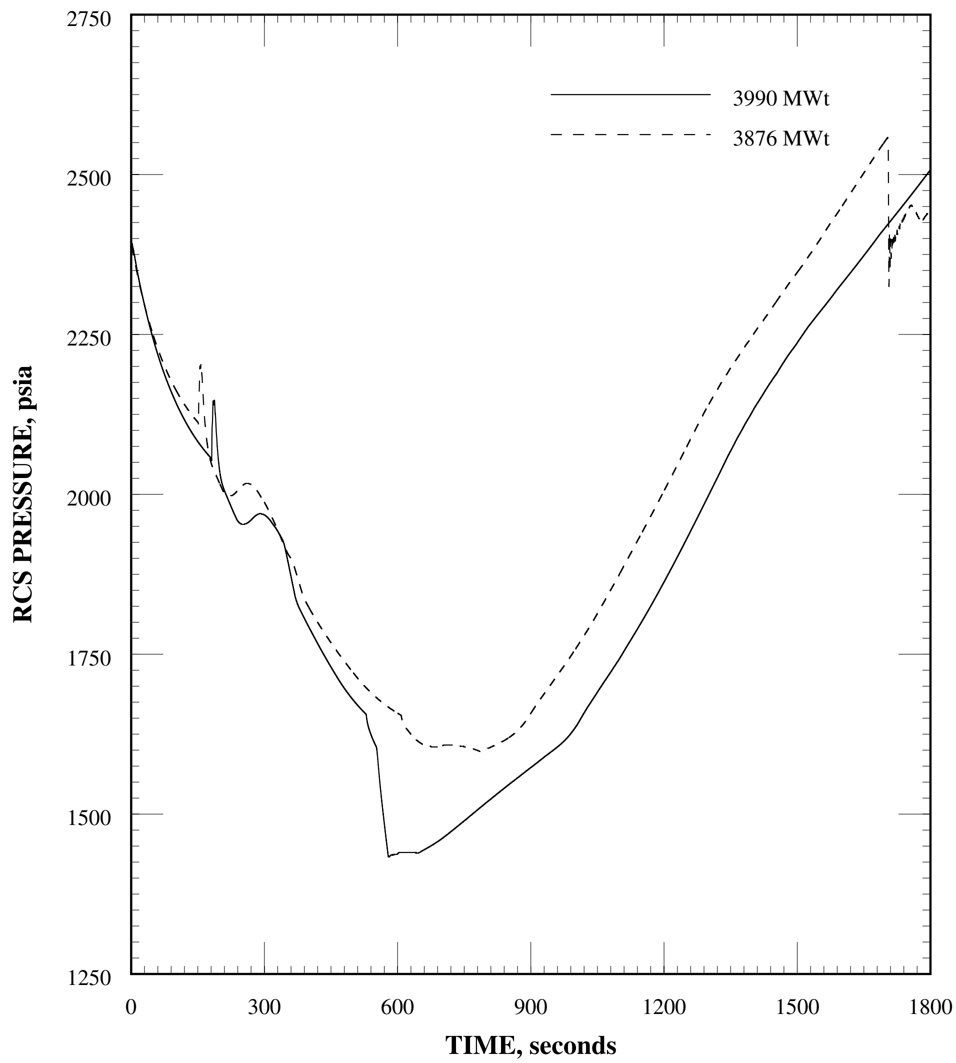


Figure 6.3-19
IOSGADV + LOP Event - Pressurizer Water Volume vs. Time

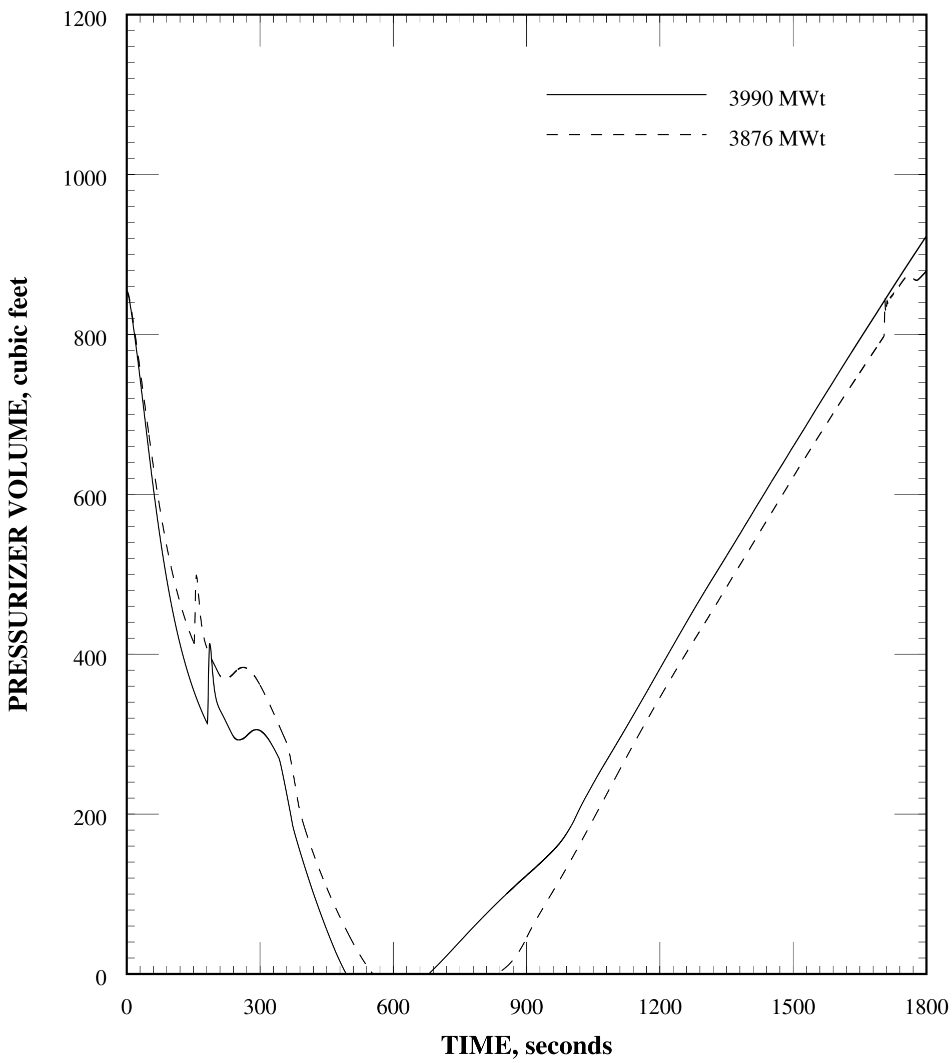


Figure 6.3-20
IOSGADV + LOP Event - Steam Flow vs. Time

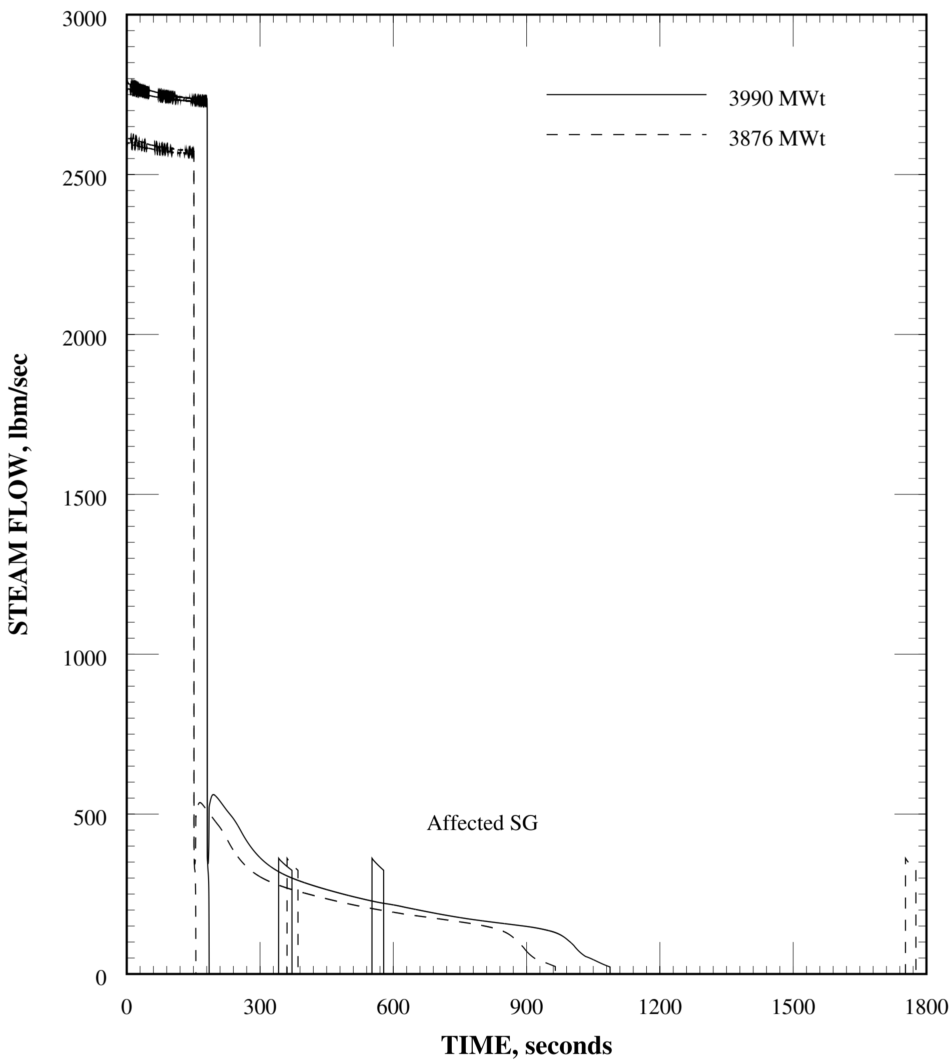


Figure 6.3-21
IOSGADV + LOP Event - SG Pressure vs. Time

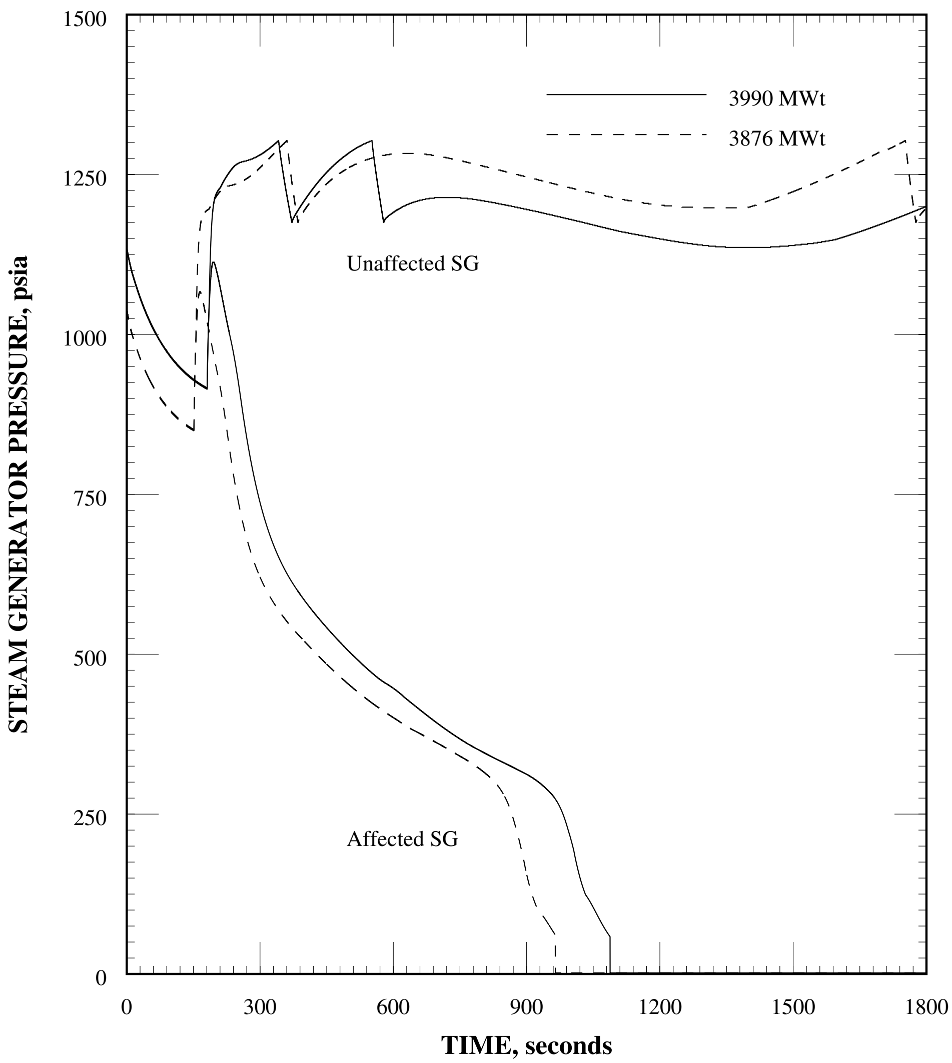


Figure 6.3-22
IOSGADV + LOP Event - SG Level vs. Time

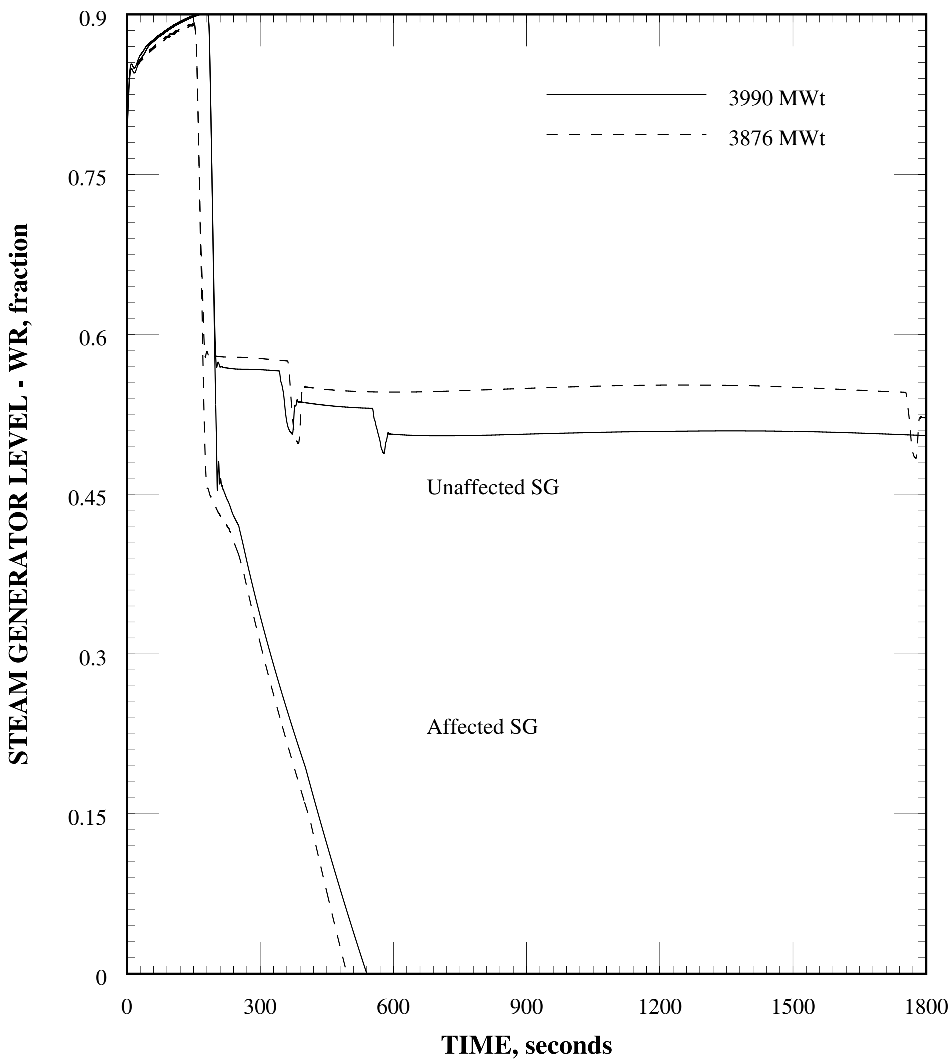


Figure 6.3-23
IOSGADV + LOP Event - SG Liquid Mass vs. Time

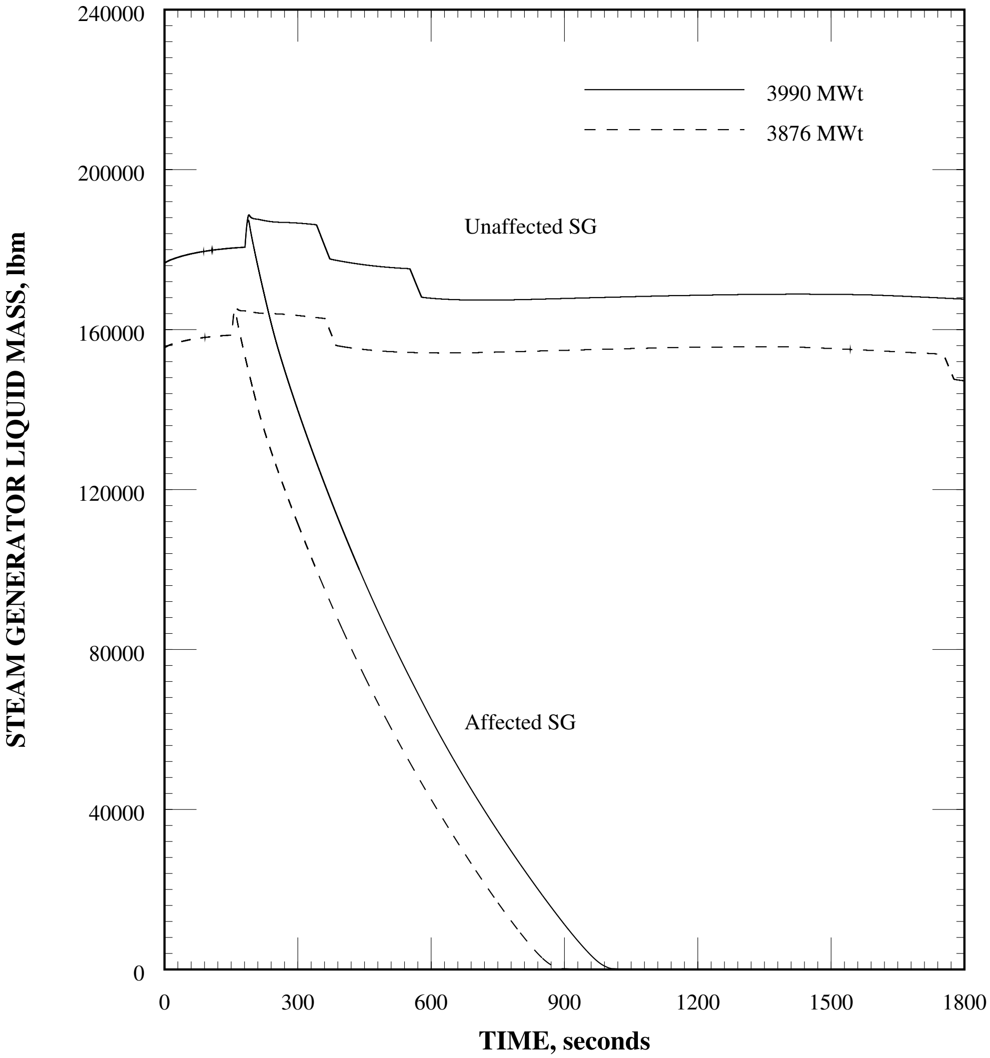


Figure 6.3-24
IOSGADV + LOP Event - Main FW Flow vs. Time

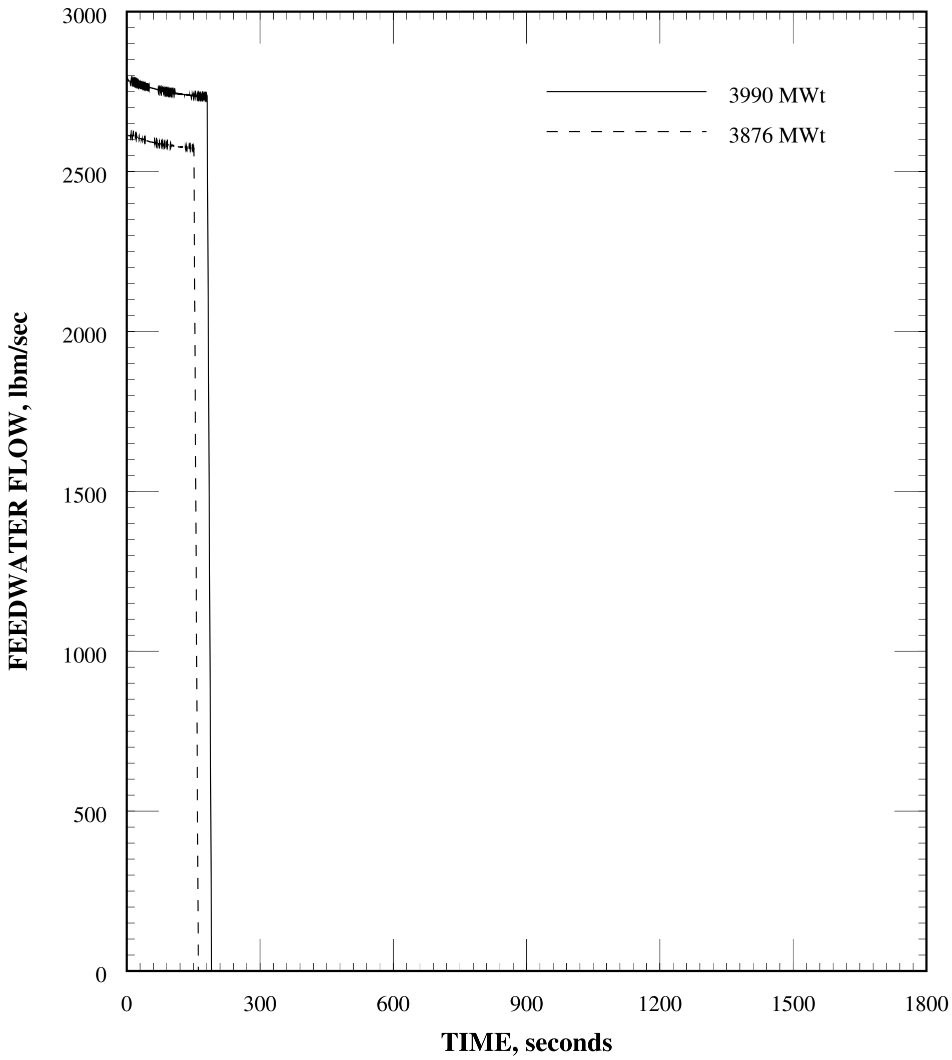
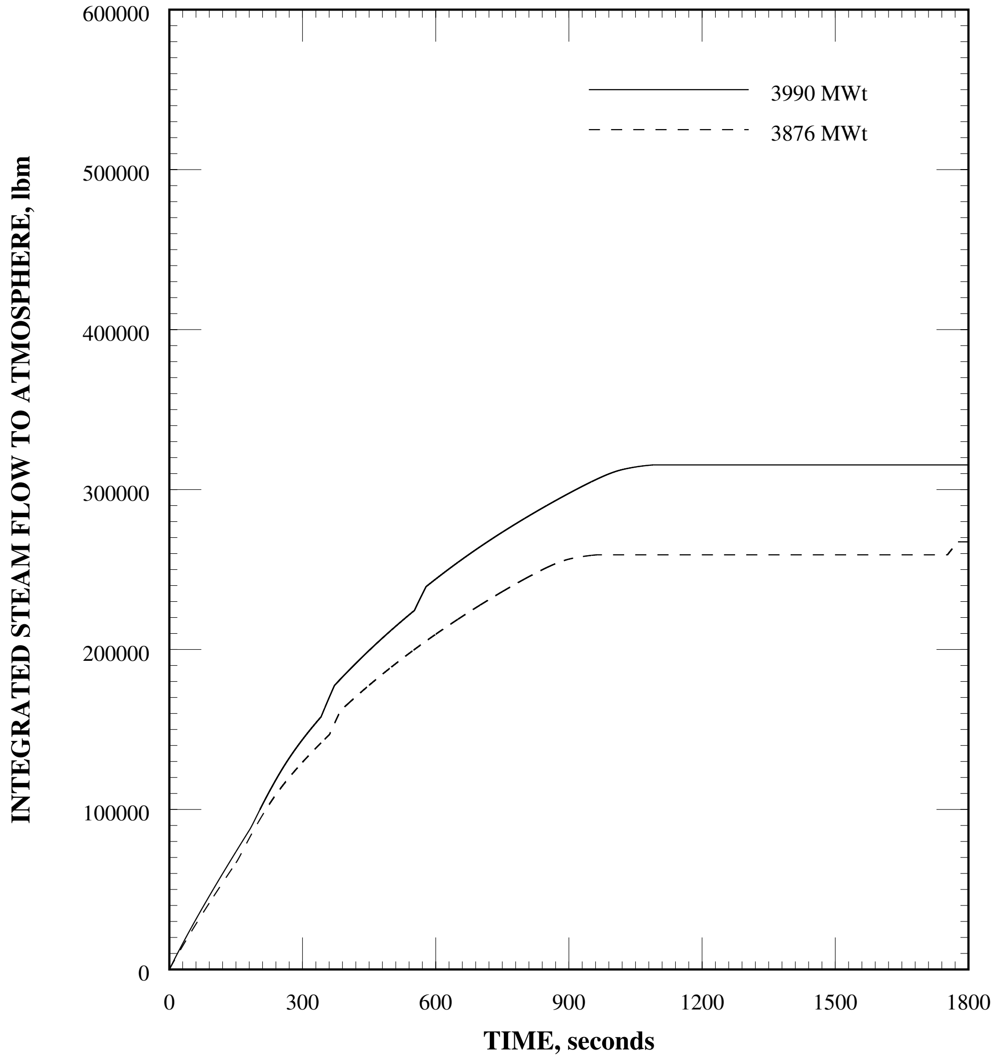


Figure 6.3-25
IOSGADV + LOP Event - Integrated Steam Flow to Atmosphere vs. Time



Section 6.3.1.5 Steam System Piping Failures Inside and Outside Containment - Mode 1 Operation

Section 6.3.1.5.1 Identification of Event and Causes

As described in UFSAR Section 15.1.5, the increase in steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected SG that causes a decrease in the overall RCS inlet temperature and pressure.

In the presence of a negative MTC, the cooldown causes positive reactivity to be added to the core. A highly negative MTC and a large break size (guillotine breaks) can combine to degrade shutdown margin and result in a potential post-trip Return to Power (R-t-P).

With a concurrent LOP, the analysis assumes that the reactor will trip on low RCP shaft speed trip, in less than one second after the pipe break. When offsite power is available, the reactor trips on high containment pressure, LSGP, low SG level, or CPC VOPT. The earliest trip is the high containment pressure that occurs in less than one second after the break. However, VOPT will be credited for the analysis.

The largest possible SLB size is the double-ended rupture of a steam line upstream of the MSIV. The integral flow restrictor in each SG outlet nozzle limits the effective steam blowdown area for each steam line.

In all guillotine break cases, the LSGP/high SG level signal initiates a MSIS that causes closure of the MSIVs and FWIVs. The steam flow from the unaffected SG is terminated by the closure of the MSIVs. Since the pipe break is assumed to occur upstream of the MSIV, the steam flow from the affected SG is not terminated until the affected SG dries out. Because of the large cooldown and shrinkage of the RCS, the RCS pressure will decrease and a SIAS will be initiated. The emptying of the affected SG and the injection of boron terminates the R-t-P and causes the core reactivity to decrease. The operators will initiate plant cooldown in accordance with the emergency procedures.

Inside containment breaks are found to be more limiting in terms of R-t-P potential. Conversely, outside containment breaks result in more adverse dose consequences and are evaluated in Section 6.4.1.2.

The following inside containment Steam Line Break (SLB) scenarios were examined for their R-t-P potential:

1. A large SLB during full power operation with a concurrent LOP in combination with a single failure and a stuck CEA.
2. A large SLB during full power operation with offsite power available in combination with a single failure and a stuck CEA.
3. A large SLB during zero power operation with concurrent LOP in combination with a single failure and a stuck CEA.

4. A large SLB during zero power operation with offsite power available in combination with a single failure and a stuck CEA.

Section 6.3.1.5.2 Acceptance Criteria

As defined in the SRP Section 15.1.5, the specific criteria for this event are:

1. The general objective of the review of steam line rupture events is to verify that short term and long-term coolability has been achieved by confirming that the primary RCS is maintained in a safe status for a break equivalent in area to the double-ended rupture of the largest steam line.
2. The specific criteria against which the consequences of this break are to be evaluated are:
    a. Pressure in the RCS and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
    b. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above specified limits based on acceptable correlations. If the DNBR falls below these values, fuel damage (rod perforation) must be assumed unless it can be shown, based on an acceptable fuel damage model that includes the potential adverse effects of hydraulic instabilities, that fuel failure has not occurred. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and maintain a coolable geometry.

The radiological criteria used in the evaluation are in the appendix to this SRP Section:

1. small fractions (less than 10%) of the 10 CFR Part 100 exposure guidelines, and
2. within 10 CFR Part 100 guidelines for the cases of a Preaccident Iodine Spike (PIS) or one control assembly held out of the core.

Section 6.3.1.5.3 Description of Analysis

The NSSS response to the SLB was simulated using the CENTS code. Moderator, Doppler, boron, and scram rod reactivity contributions are explicitly modeled. Moderator and Doppler reactivities are parameterized as functions of average moderator density on the cold edge of the core and effective fuel temperature, respectively, for use by CENTS. Reactivity coefficients corresponding to EOC operation (most negative MTC) were used for the SLBs to maximize post-trip reactivity insertion.

Section 6.3.1.5.3.1 Change in Method of Evaluation

The CENTS three-dimensional reactivity feedback option incorporates the capability of three-dimensional reactivity effects associated with local changes in moderator density. The three-dimensional reactivity contribution is based on HERMITE calculations and is parameterized in CENTS as a function of core inlet plane temperature tilt (difference between hot and cold edge temperatures), core flow, and core fission power. The limiting R-t-P scenario was evaluated with and without this reactivity feedback option to demonstrate its usage for the applications.

Section 6.3.1.5.4 Input Parameters, Initial Conditions, and Assumptions

Input Parameters

Degradation in fuel performance during the post-trip portion of SLB initiated transients can only occur if there is a R-t-P. Therefore, the primary consideration for maximizing post-trip degradation in fuel performance is to select those parameters and conditions that will maximize R-t-P.

The plant parameter that has the first-degree effect on the R-t-P is the positive reactivity added due to the MTC. A moderator cooldown curve based upon the most negative MTC is used in the analysis. Included in this cooldown curve is a component modeling reduction in CEA worth as the RCS temperature decreases. This most negative MTC value occurs at the EOC conditions. The fuel temperature reactivity coefficient (FTC) is also most negative at EOC. This most negative FTC will also adversely affect the R-t-P by adding more positive reactivity as the fuel goes from operating temperatures before the break to the lowered temperatures during the cooldown caused by the SLB. Table 6.3-9 through Table 6.3-12 contains the initial conditions used for the four MSLB scenarios.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. Saturated steam blowdown with no moisture carryover yields the maximum energy removal.
3. For the determination of DNBR for post-trip SLB conditions, the MacBeth DNBR correlation (Reference 6-22 and Reference 6-23) has been selected to represent margin to DNB during periods of possible R-t-P. As explained in the UFSAR Section 15C.2.3, MacBeth correlates critical heat flux to mass flux, inlet subcooling, pressure, heated diameter, and channel length. Application of a channel heat balance allows the correlation to be converted to a “local conditions” form. Using this form of the correlation, critical heat flux as a function of height in the hot channel (which is located near the stuck CEA location) is calculated, where the effect of non-uniform axial heating is incorporated using the method applied by Lee in Reference 6-24.
4. The failure of one HPSI pump is assumed to occur.
5. The SLB occurs upstream of the MSIV resulting in an unisolatable SG blowdown.
6. Cooldown is maximized by assuming that the full SG heat transfer area is maintained (rather than decreasing as SG water mass decreases) until the affected SG is empty.

There is no operator action for the first 30 minutes of the event.

Table 6.3-9
Parameters Used for HFP MSLB with LOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	562	568
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC	most negative	most negative
FTC	most negative	most negative
Kinetics	maximum β	maximum β
Inverse boron worth (ppm/%Δρ)	130	120
CEA worth at trip - WRSO (%Δρ)	-8.75	-8.75
Fuel rod gap conductance (Btu/hr-ft^2-°F)	594	594
Plugged SG tubes (% of tubes/SG)	16	10
Single failure	1 HPSI pump	1 HPSI pump
LOP	yes	yes

Table 6.3-10
Parameters Used for HFP MSLB without LOP Event

PARAMETER	Value	
	$3876\ MW_t$	$3990\ MW_t$
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	562	568
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC	most negative	most negative
FTC	most negative	most negative
Kinetics	maximum β	maximum β
Inverse boron worth (ppm/%$\Delta\rho$)	130	120
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.75	-8.75
Fuel rod gap conductance (Btu/hr-ft^2-°F)	594	594
Plugged SG tubes (% of tubes/SG)	16	10
Single failure	1 HPSI pump	1 HPSI pump
LOP	no	no

Table 6.3-11
Parameters Used for HZP MSLB with LOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	0.025	0.025
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC	most negative	most negative
FTC	most negative	most negative
Kinetics	maximum β	maximum β
Inverse boron worth (ppm/%$\Delta\rho$)	130	120
CEA worth at trip - WRSO (%$\Delta\rho$)	-6.5	-6.5
Fuel rod gap conductance (Btu/hr-ft^2-°F)	594	594
Plugged SG tubes (% of tubes/SG)	16	10
Single failure	1 HPSI pump	1 HPSI pump
LOP	yes	yes

Table 6.3-12
Parameters Used for HZP MSLB without LOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	0.025	0.025
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC	most negative	most negative
FTC	most negative	most negative
Kinetics	maximum β	maximum β
Inverse boron worth (ppm/%$\Delta\rho$)	130	120
CEA worth at trip - WRSO (%$\Delta\rho$)	-6.5	-6.5
Fuel rod gap conductance (Btu/hr-ft^2-°F)	594	594
Plugged SG tubes (% of tubes/SG)	16	10
Single failure	1 HPSI pump	1 HPSI pump
LOP	no	no

Section 6.3.1.5.5 Results

The most limiting MSLB scenario is the full power event with a concurrent LOP. The behavior of NSSS parameters for this scenario is presented in Figure 6.3-26 through Figure 6.3-40, and Table 6.3-13 summarizes the sequence of events. For the other non-limiting cases, the sequences of events are presented in Table 6.3-14 through Table 6.3-16.

For a SLB with a concurrent LOP, turbine stop valve closure, termination of main FW to both SGs, and coastdown of the RCPs are assumed to occur simultaneously. At the same time, an actuation signal for the EDGs is initiated. Due to decreasing RCP speed following LOP, a CPC trip occurs. In addition, after a SLB with the LOP, a trip can be initiated by LSGP, SG ΔP, low RCS flow, or high containment pressure. Following this, reactor trip breakers open. Increasing SG level generates a MSIS when the setpoint is reached. The MSIS initiates closure of the MSIVs and FWIVs. Closure of the MSIVs terminates steam flow from the unaffected SG. Subsequently, the pressure difference between the SGs reaches the analysis setpoint for lockout of AFW to the affected SG. The affected SG eventually empties.

Decrease in the pressurizer pressure results in a SIAS. Within 20 seconds of SIAS, the operable HPSI pump is loaded on the EDG (single failure of a HPSI pump is assumed to occur). The HPSI valves open and the HPSI pump delivers full flow. SI boron begins to reach the core once the sweep out volume is displaced. Thirty minutes after the event initiation, the operator resumes the plant cooldown by manual control of the ADVs. Shutdown Cooling System (SCS) is initiated when the RCS reaches SCS entry conditions.

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). In general, the larger initial RCS inventory with PUR results in further RCS cooldown and a more severe R-t-P.

Radiological consequences for this event are presented in Section 6.4.1.2.

Section 6.3.1.5.5.1 Change in Methodology Reactivity Credit - Moderator Density Feedback

As discussed in Section 6.3.1.5.3.1, the CENTS reactivity feedback could be obtained based on three-dimensional feedback using the HERMITE methodology. This results in a considerable decrease in required scram worth for a similar amount of R-t-P and minimum DNBR.

Using the three-dimensional reactivity credit, the limiting case (HFP MSLB with LOP) required a scram worth of only 5.75%$\Delta\rho$ to achieve a similar R-t-P relative to the original case, which credited 8.75%$\Delta\rho$ scram worth. Figure 6.3-41 presents the core reactivity versus time plot for the event that credited the moderator density feedback.

Table 6.3-13
Sequence of Events for HFP MSLB with LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.0	0.0	SLB occurs	---	---
0.0	0.0	LOP occurs	---	---
0.59	0.59	Low RCP shaft speed trip condition	0.95	0.95
0.89	0.89	Reactor trip breakers open	---	---
0.89	0.89	Turbine trip occurs	---	---
1.49	1.49	Scram CEAs begin falling	---	---
1.67	2.30	MSIS on high SG level (%NR)	99.9	99.9
7.28	7.91	Complete closure of the MSIV	---	---
15.06	17.16	SG delta pressure isolation reached (psid)	270	270
94.26	49.61	Pressurizer empties (<0.5 ft)	---	---
95.84	48.99	SIAS setpoint is reached (psia)	1670	1670
115.84	68.99	SI pumps reach full speed and begin injecting	---	---
148.6	81.3	Voids begin to form in the RV upper head	---	---
206.3	159.9	Boron reaches RCS	---	---
---	360.0	Minimum DNBR occurs	>5.0	1.49
---	361.0	Maximum post-trip core power (% of rated)	No R-t-P	4.45
219.0	365.0	Maximum post-trip fission power (%)	No R-t-P	+1.872
333.7	351.6	Affected SG empties	---	---
336.2	336.0	Maximum post-trip reactivity (%$\Delta\rho$)	-0.309	+0.026
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Table 6.3-14
Sequence of Events for HFP MSLB without LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.0	0.0	SLB occurs	---	---
1.67	2.32	MSIS on high SG level (%NR)	99.9	99.9
4.82	5.72	CPC VOPT (% power)	110	110
5.57	6.47	Reactor trip breakers open	---	---
5.57	6.47	Turbine trip occurs	---	---
6.12	7.07	Scram CEAs begin falling	---	---
7.28	7.93	Complete closure of the MSIV	---	---
13.76	16.56	SG delta pressure isolation reached (psid)	270	270
57.96	46.91	Pressurizer empties (<0.5 ft)	---	---
58.61	47.21	SIAS setpoint is reached (psia)	1670	1670
74.71	60.85	Voids begin to form in the RV upper head	---	---
78.61	67.21	SI pumps reach full speed and begin injecting	---	---
144.8	137.8	Boron reaches RCS	---	---
145.8	160.6	Affected SG empties	---	---
149.5	162.0	Maximum post-trip reactivity (%$\Delta\rho$)	-1.204	-0.678
----	---	Maximum post-trip core power (% of rated)	No R-t-P	No R-t-P
----	---	Maximum post-trip fission power (%)	No R-t-P	No R-t-P
---	---	Minimum DNBR occurs	>5.0	>5.0
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Table 6.3-15
Sequence of Events for HZP MSLB with LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.0	0.0	SLB occurs	---	---
0.0	0.0	LOP occurs	---	---
0.59	0.59	Low RCP shaft speed trip condition	0.95	0.95
0.89	0.89	Reactor trip breakers open	---	---
0.89	0.89	Turbine trip occurs	---	---
1.49	1.49	Scram CEAs begin falling	---	---
4.71	3.96	MSIS on high SG level (%NR)	99.9	99.9
10.3	9.57	Complete closure of the MSIV	---	---
21.9	26.0	SG delta pressure isolation reached (psid)	270	270
64.2	59.1	Pressurizer empties (<0.5 ft)	---	---
69.4	59.5	SIAS setpoint is reached (psia)	1670	1670
89.4	79.5	SI pumps reach full speed and begin injecting	---	---
156.1	136.2	Voids begin to form in the RV upper head	---	---
161.5	147.9	Boron reaches RCS	---	---
295.0	354.0	Maximum post-trip reactivity (%$\Delta\rho$)	-0.998	-0.783
---	---	Maximum post-trip core power (% of rated)	No R-t-P	No R-t-P
---	---	Maximum post-trip fission power (%)	No R-t-P	No R-t-P
---	---	Minimum DNBR occurs	>5.0	>5.0
---	----	SG does not empty within 1800 sec	---	---
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Table 6.3-16
Sequence of Events for HZP MSLB without LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.0	0.0	SLB occurs	---	---
4.75	3.96	MSIS on high SG level (%NR)	99.9	99.9
8.22	8.52	CPC VOPT (% normal power)	30	30
8.97	9.27	Reactor trip breakers open	---	---
8.97	9.27	Turbine trip occurs	---	---
9.57	9.87	Scram CEAs begin falling	---	---
10.4	9.58	Complete closure of the MSIV	---	---
26.7	17.9	SG delta pressure isolation reached (psid)	270	270
51.6	50.1	Pressurizer empties (<0.5 ft)	---	---
54.1	51.4	SIAS setpoint is reached (psia)	1670	1670
74.1	71.1	SI pumps reach full speed and begin injecting	---	---
80.6	78.3	Voids begin to form in the RV upper head	---	---
124.4	122.9	Boron reaches RCS	---	---
283.0	344.0	Maximum post-trip reactivity (%$\Delta\rho$)	-1.323	-0.990
---	---	Maximum post-trip core power (% of rated)	No R-t-P	No R-t-P
---	---	Maximum post-trip fission power (%)	No R-t-P	No R-t-P
---	---	Minimum DNBR occurs	>5.0	>5.0
330.9	357.6	Affected SG empties	---	---
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.1.5.6 Conclusions

For the SLB events with and without a LOP, all acceptance criteria are met. The peak primary and secondary pressures remain below 110% of design at all times thus ensuring the integrity of the RCS and the main steam system. The minimum DNBR remains above the safety analysis limit thus ensuring fuel cladding integrity.

Figure 6.3-26
MSLB HFP with LOP - Core Power vs. Time

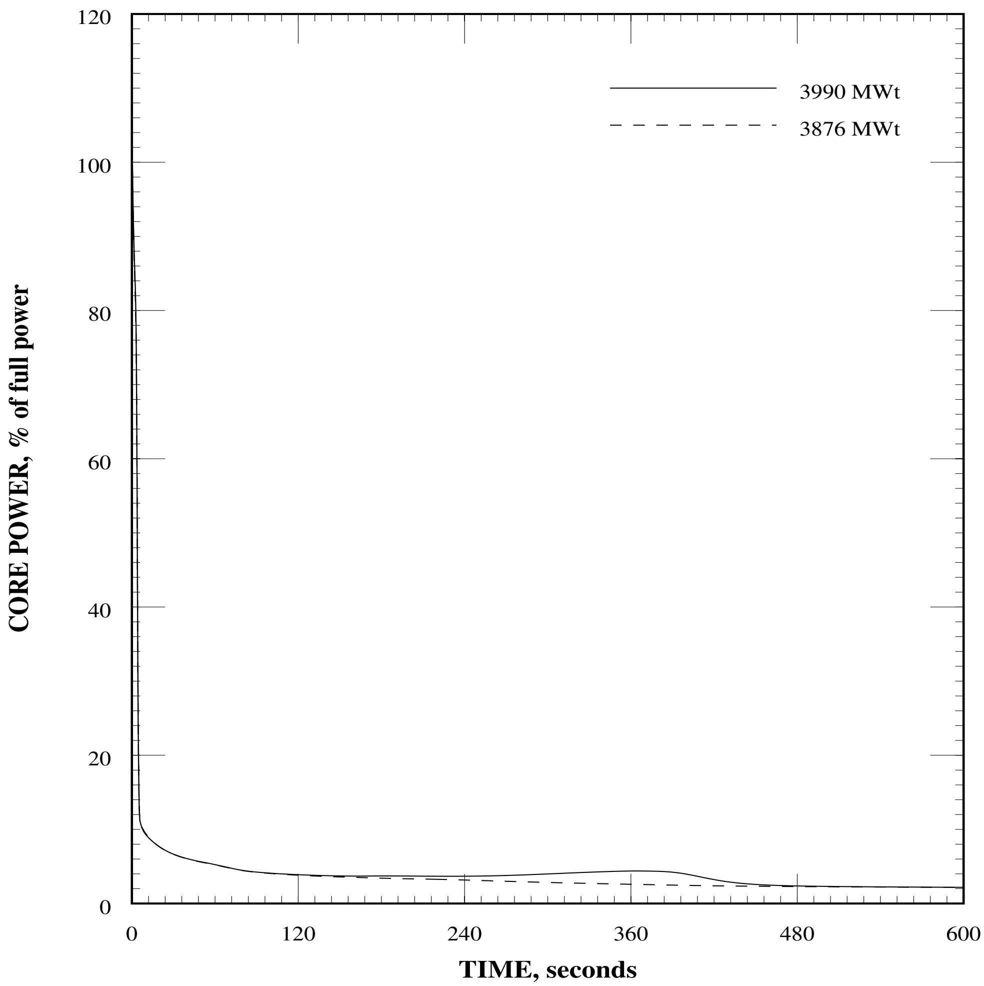


Figure 6.3-27
MSLB HFP with LOP - Core Average Heat Flux vs. Time

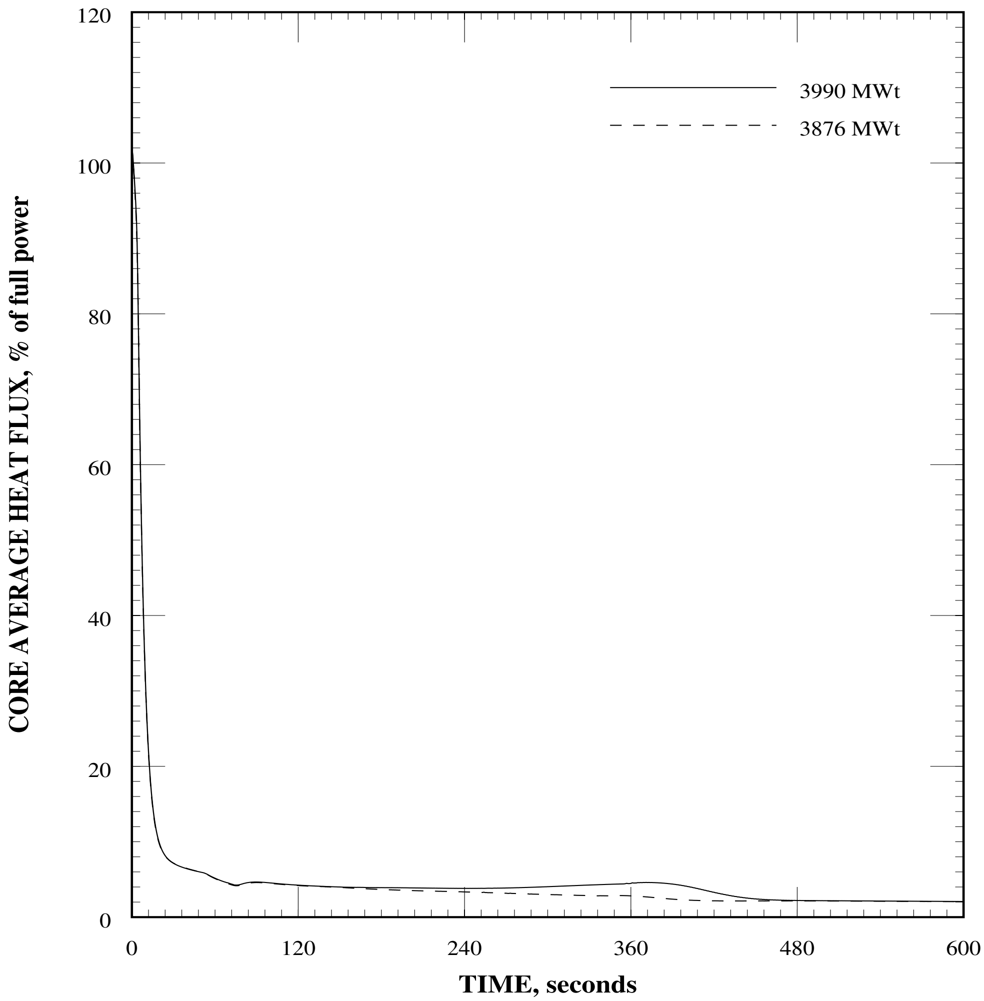


Figure 6.3-28
MSLB HFP with LOP - RCS Pressure vs. Time

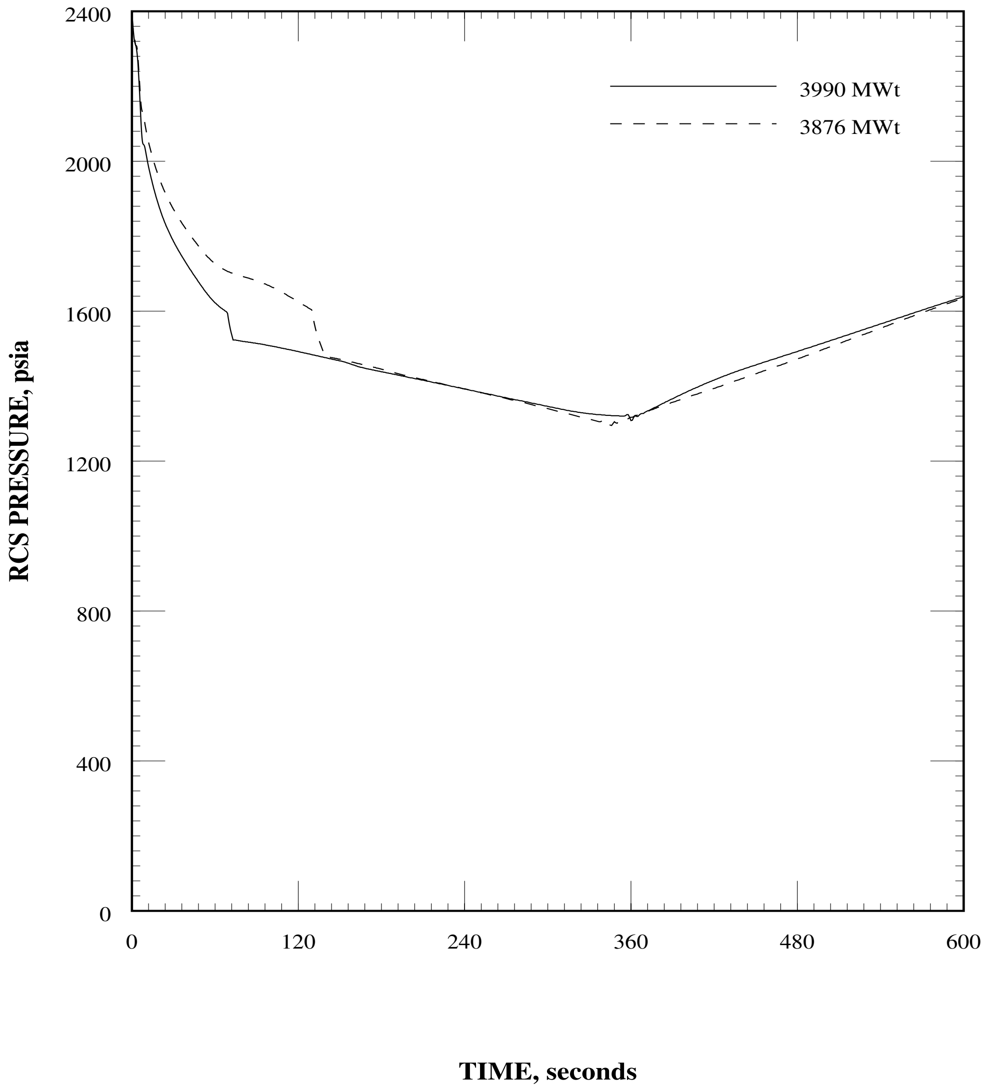


Figure 6.3-29
MSLB HFP with LOP - RCS Flow vs. Time

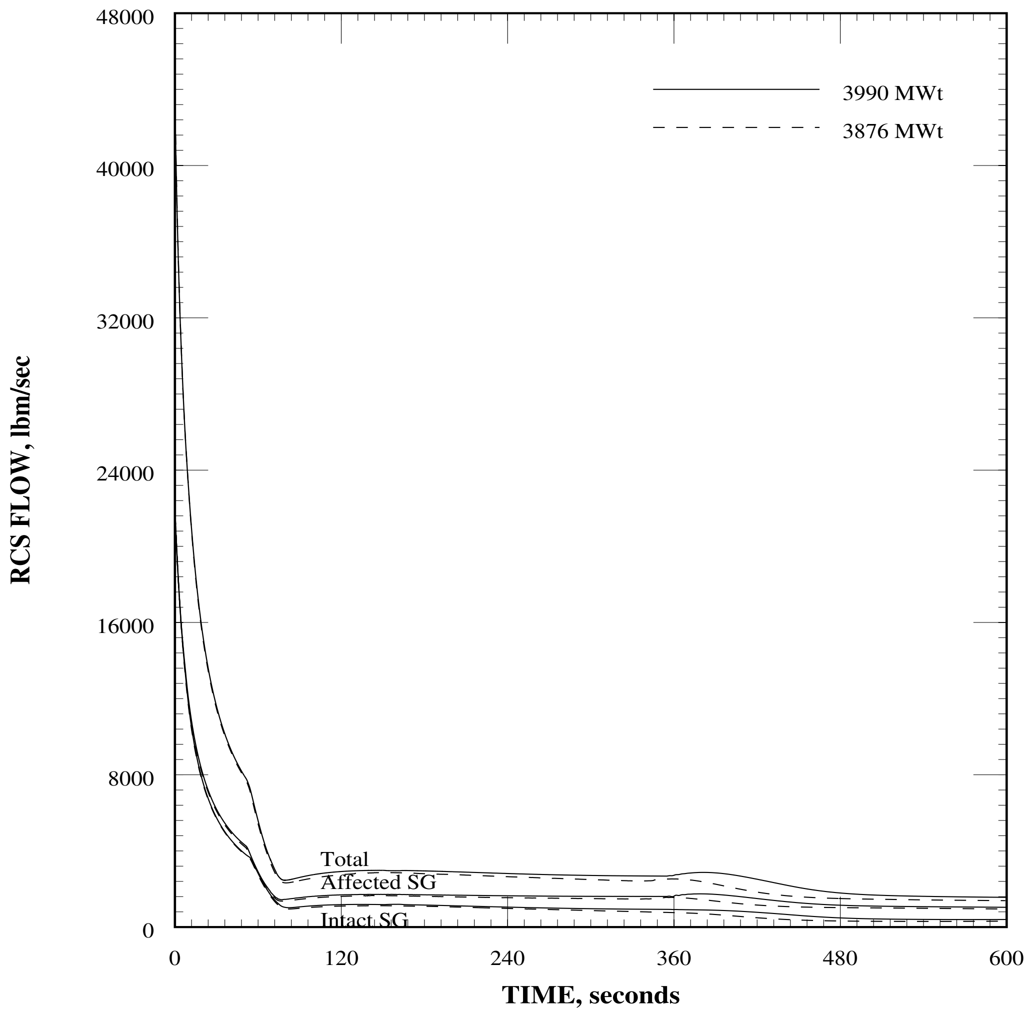


Figure 6.3-30
MSLB HFP with LOP - RCS Temperature vs. Time
(Sheet 1 of 3)

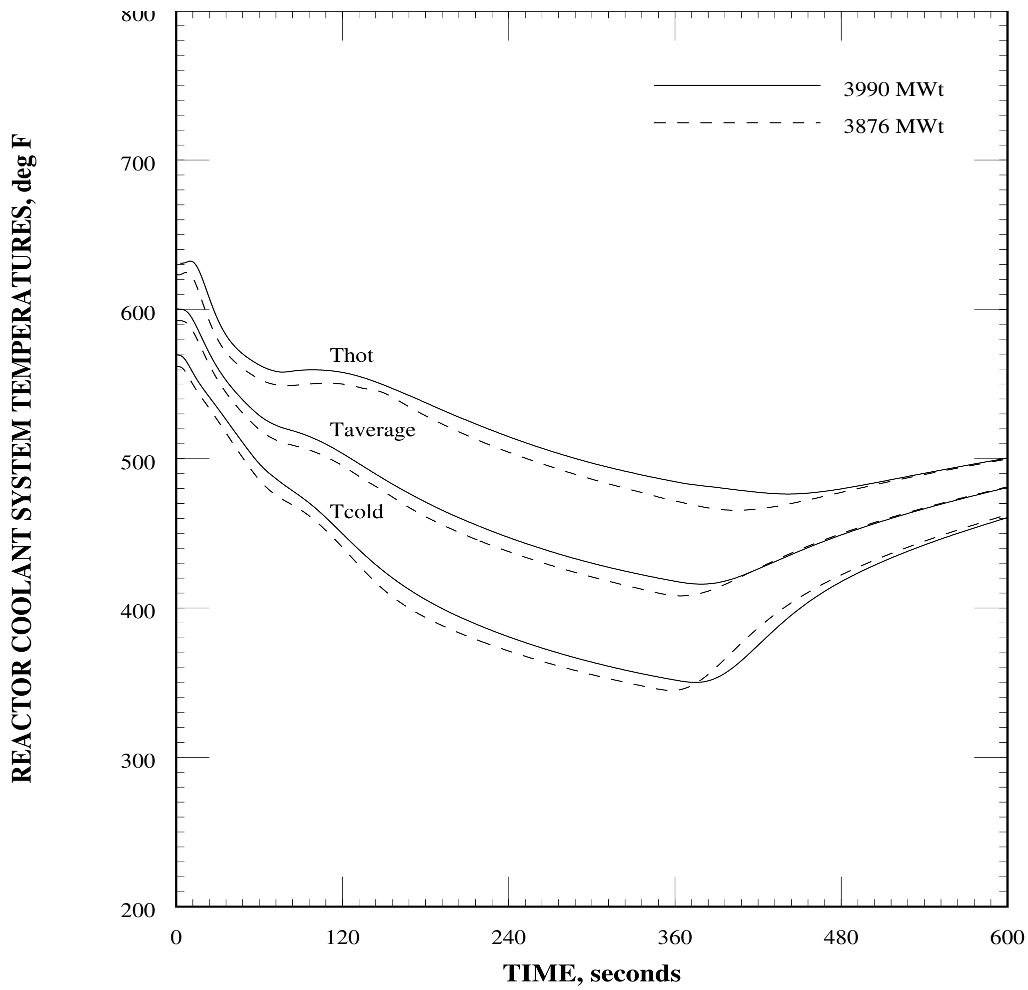


Figure 6.3-30
MSLB HFP with LOP - RCS Temperature vs. Time

(Sheet 2 of 3)

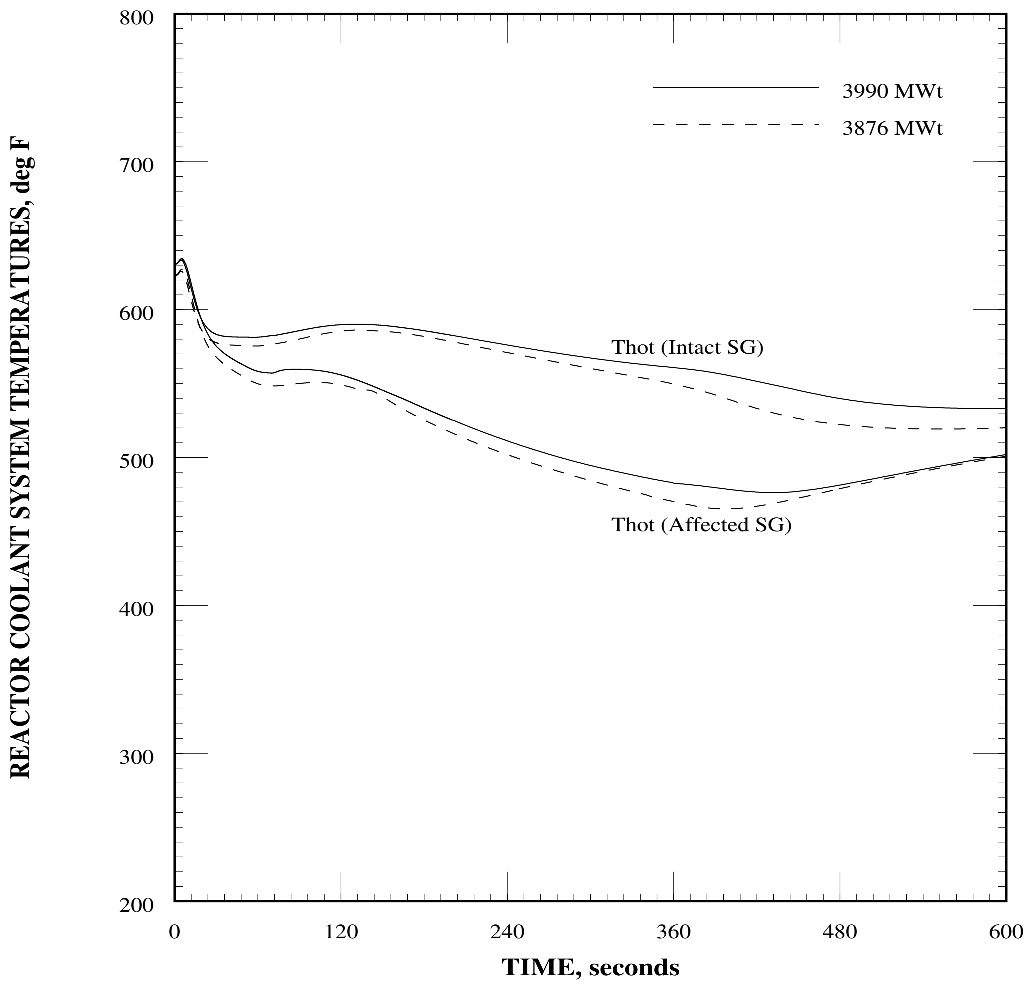


Figure 6.3-30
MSLB HFP with LOP - RCS Temperature vs. Time

(Sheet 3 of 3)

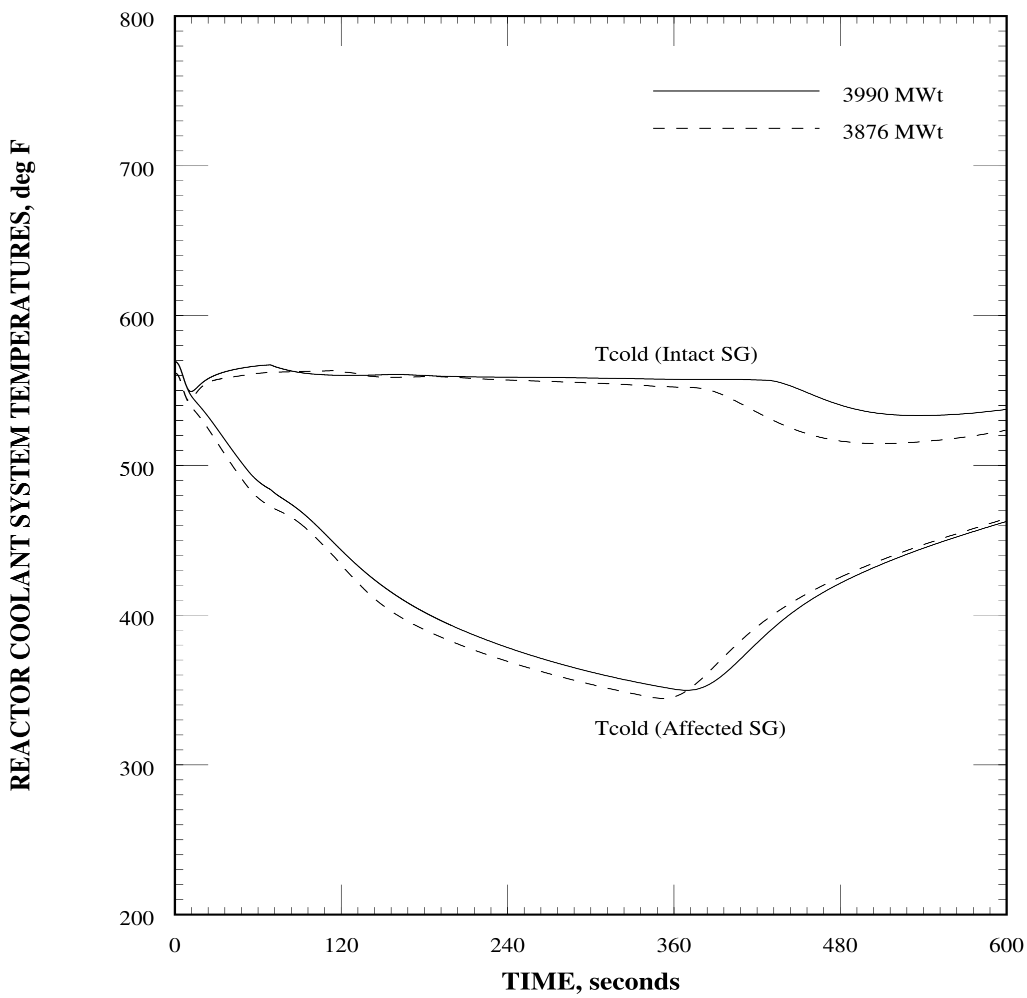


Figure 6.3-31
MSLB HFP with LOP - Core Reactivity vs. Time

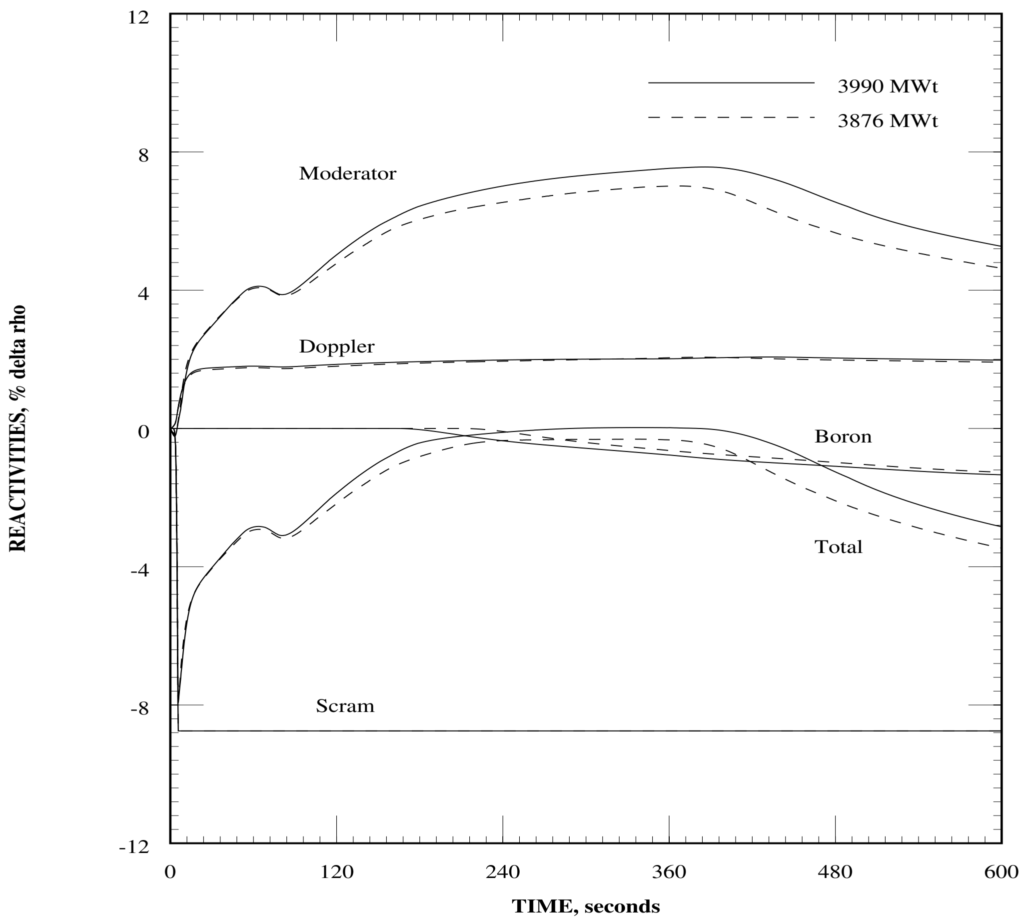


Figure 6.3-32
MSLB HFP with LOP - Pressurizer Volume vs. Time

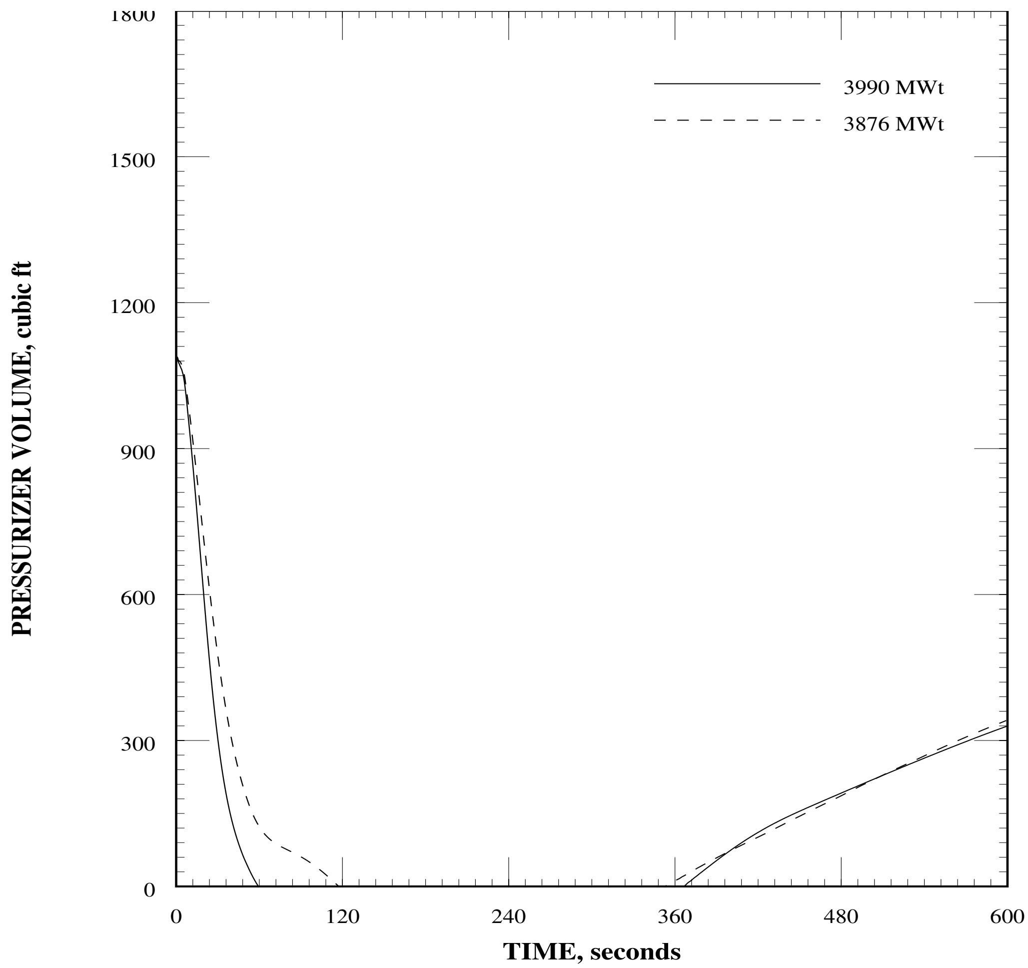


Figure 6.3-33
MSLB HFP with LOP - SG Pressure vs. Time

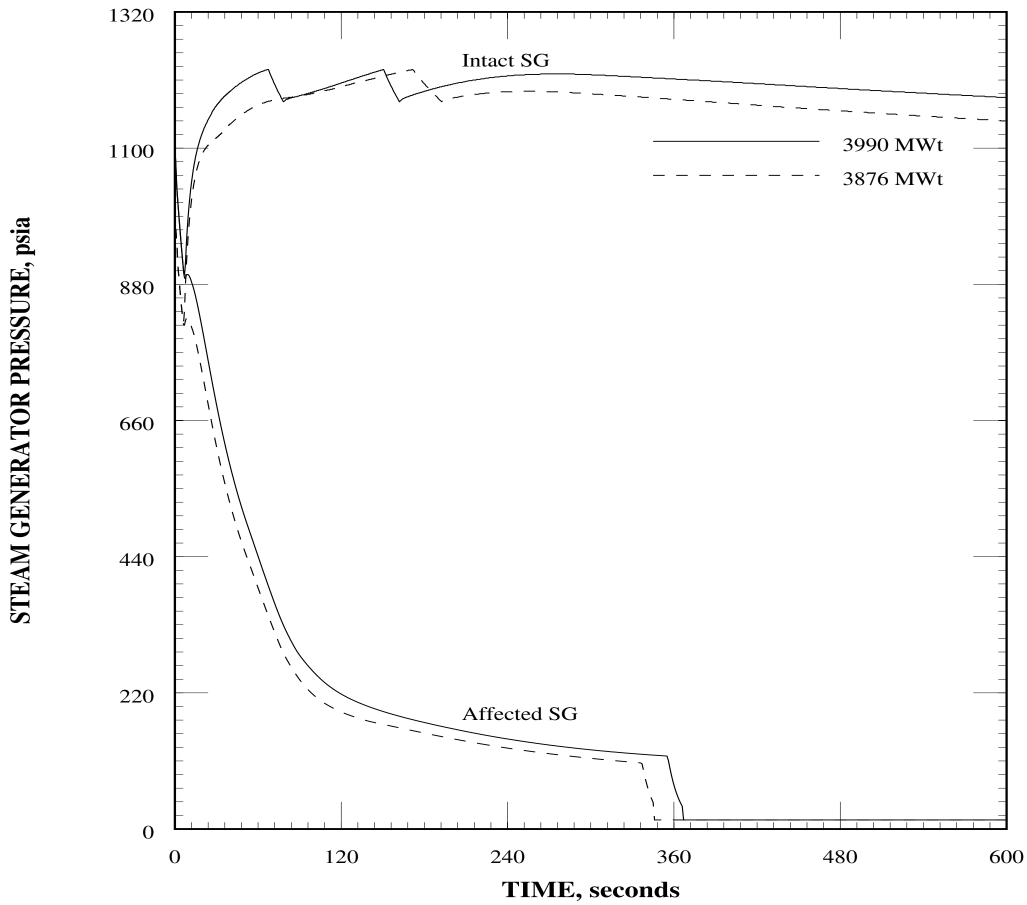


Figure 6.3-34
MSLB HFP with LOP - SG Steam Flow (per Nozzle) vs. Time

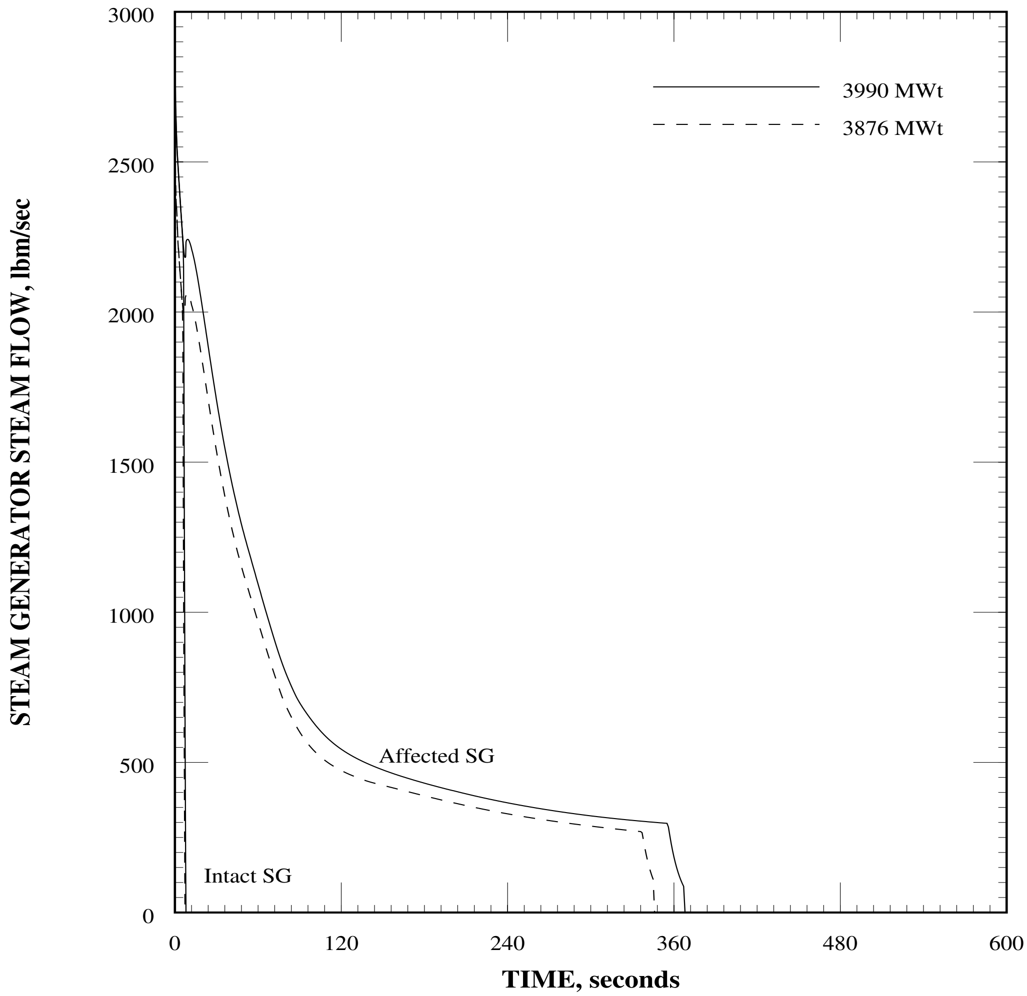


Figure 6.3-35
MSLB HFP with LOP - MFW and AFW Flow vs. Time

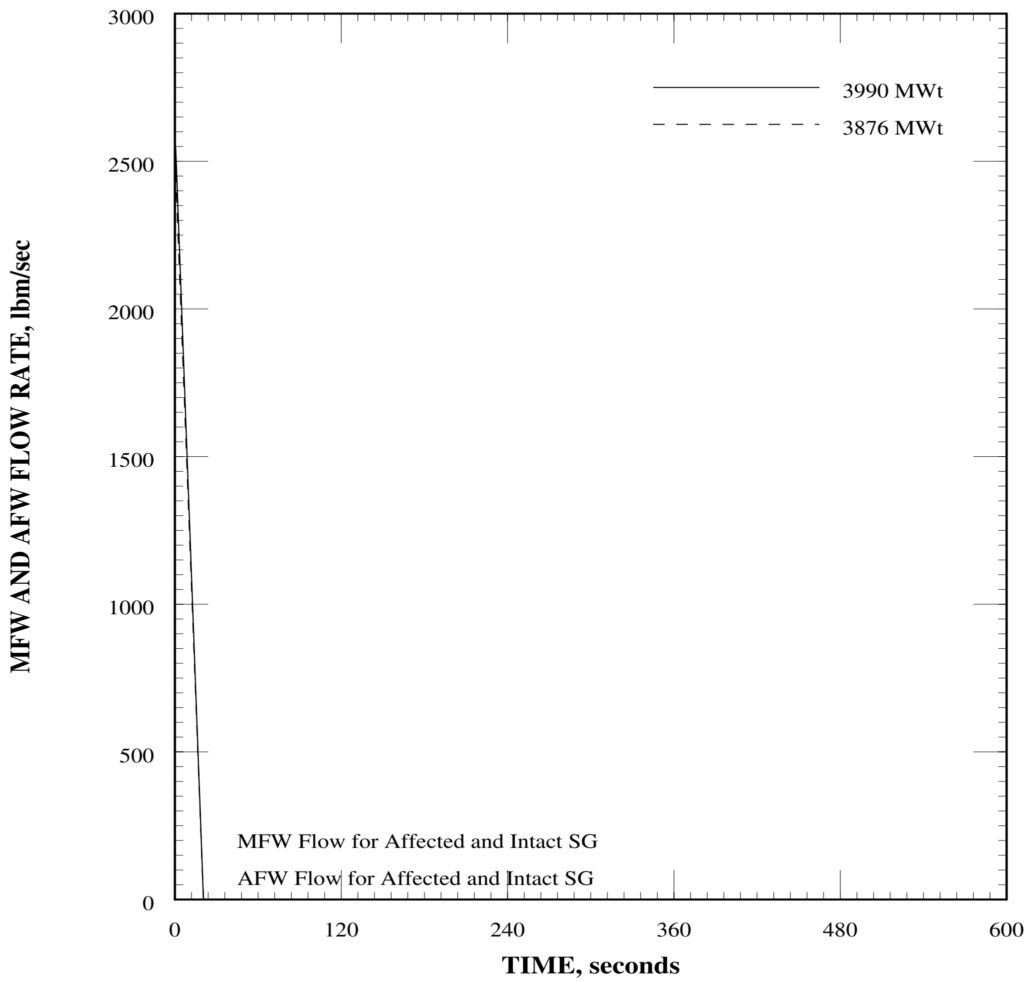


Figure 6.3-36
MSLB HFP with LOP - SG Inventory vs. Time

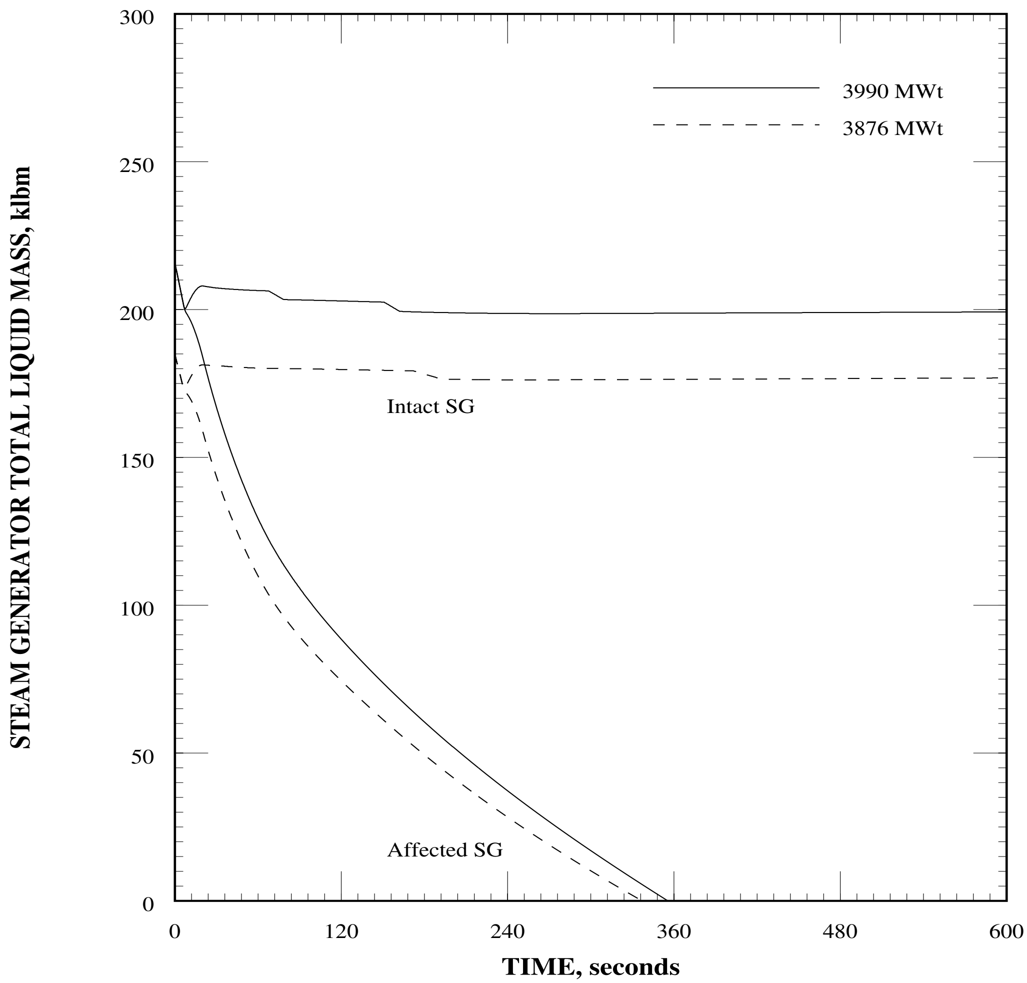


Figure 6.3-37
MSLB HFP with LOP - Integrated Steam Release vs. Time

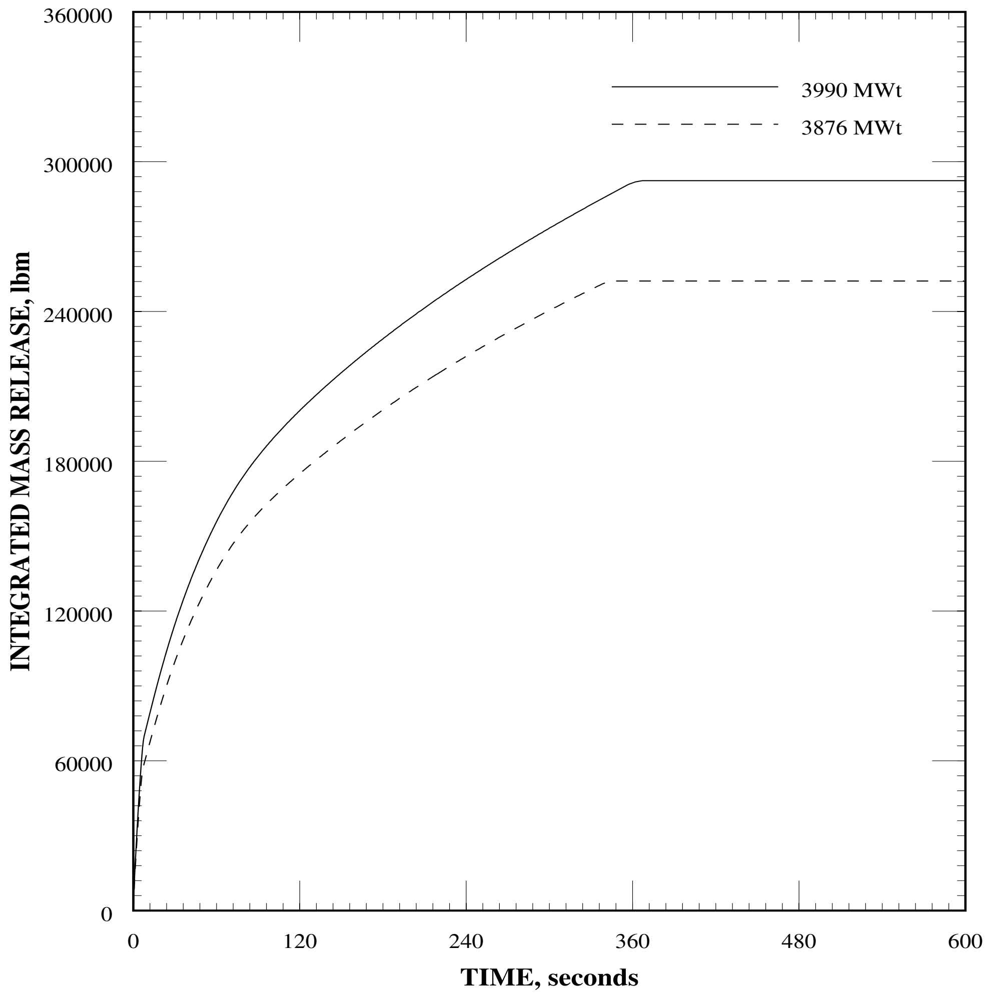


Figure 6.3-38
MSLB HFP with LOP - SI Flow vs. Time

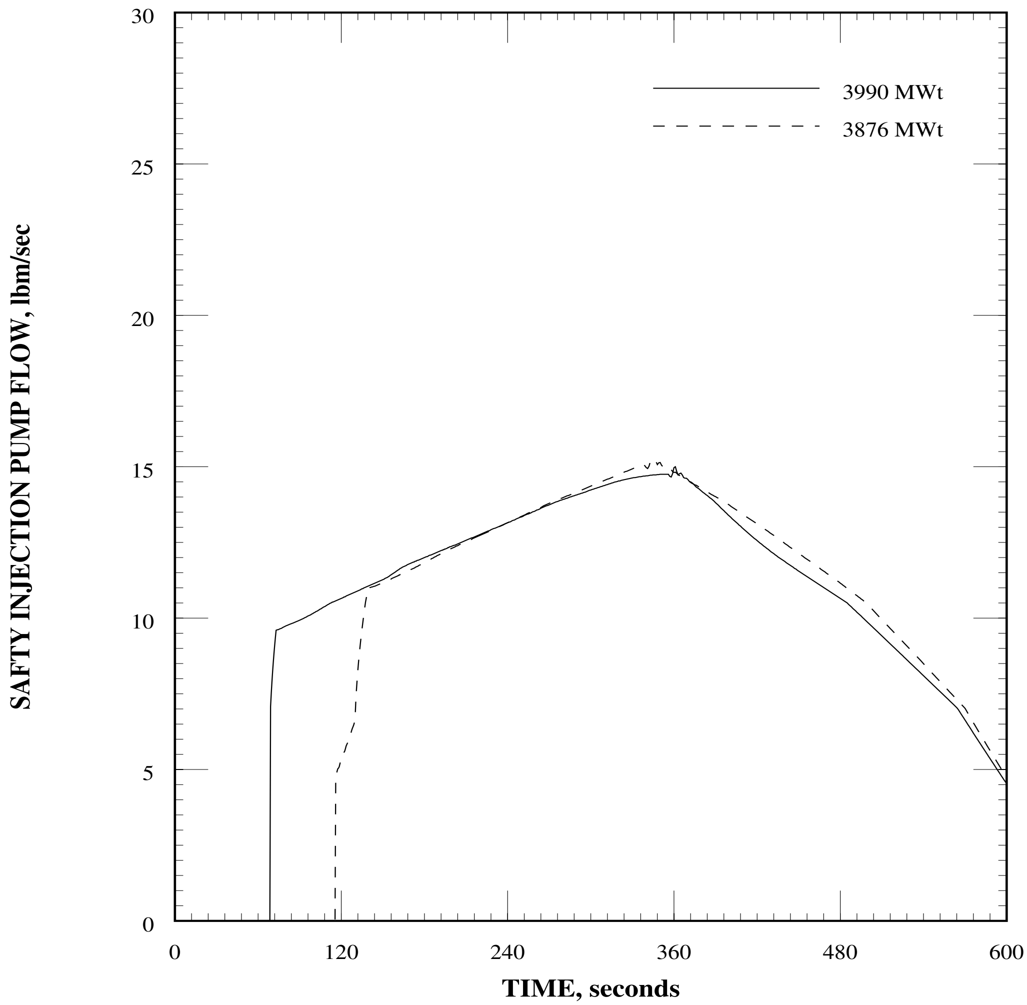


Figure 6.3-39
MSLB HFP with LOP - Reactor Vessel (Upper Head) Liquid Level vs. Time

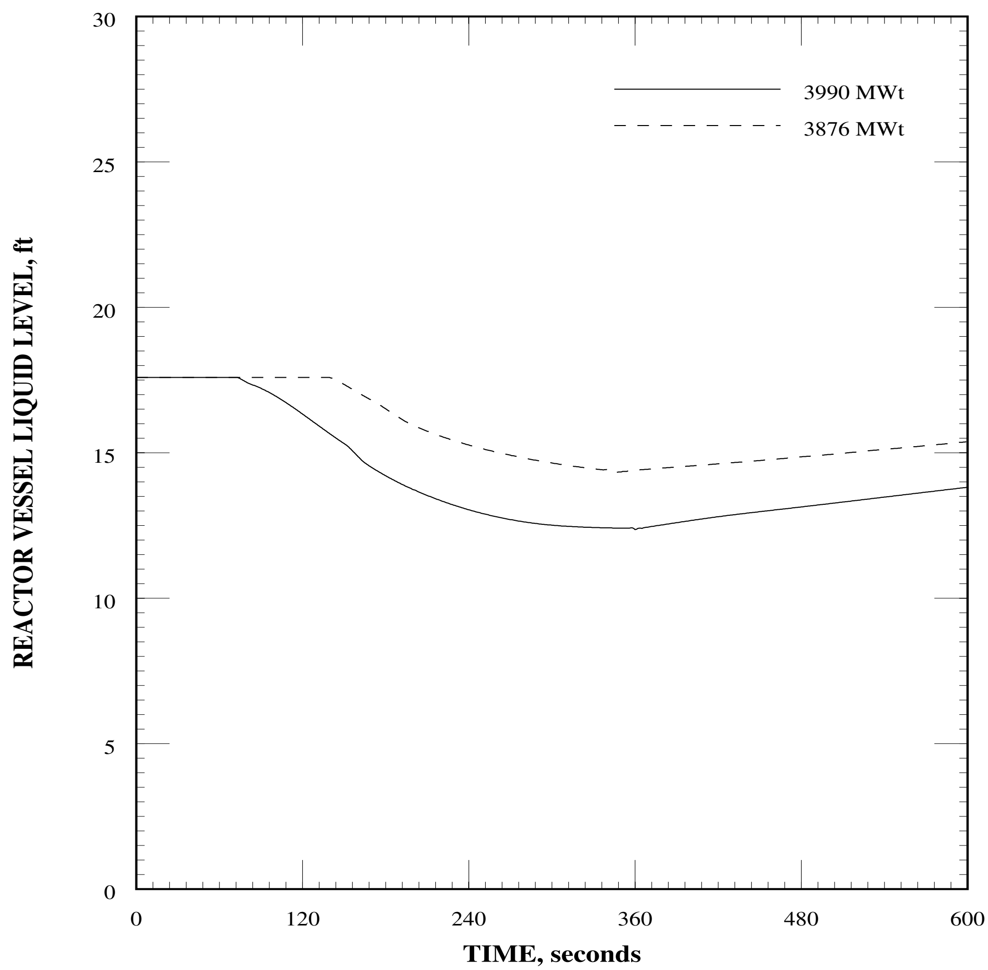


Figure 6.3-40
MSLB HFP with LOP - MacBeth DNBR vs. Time

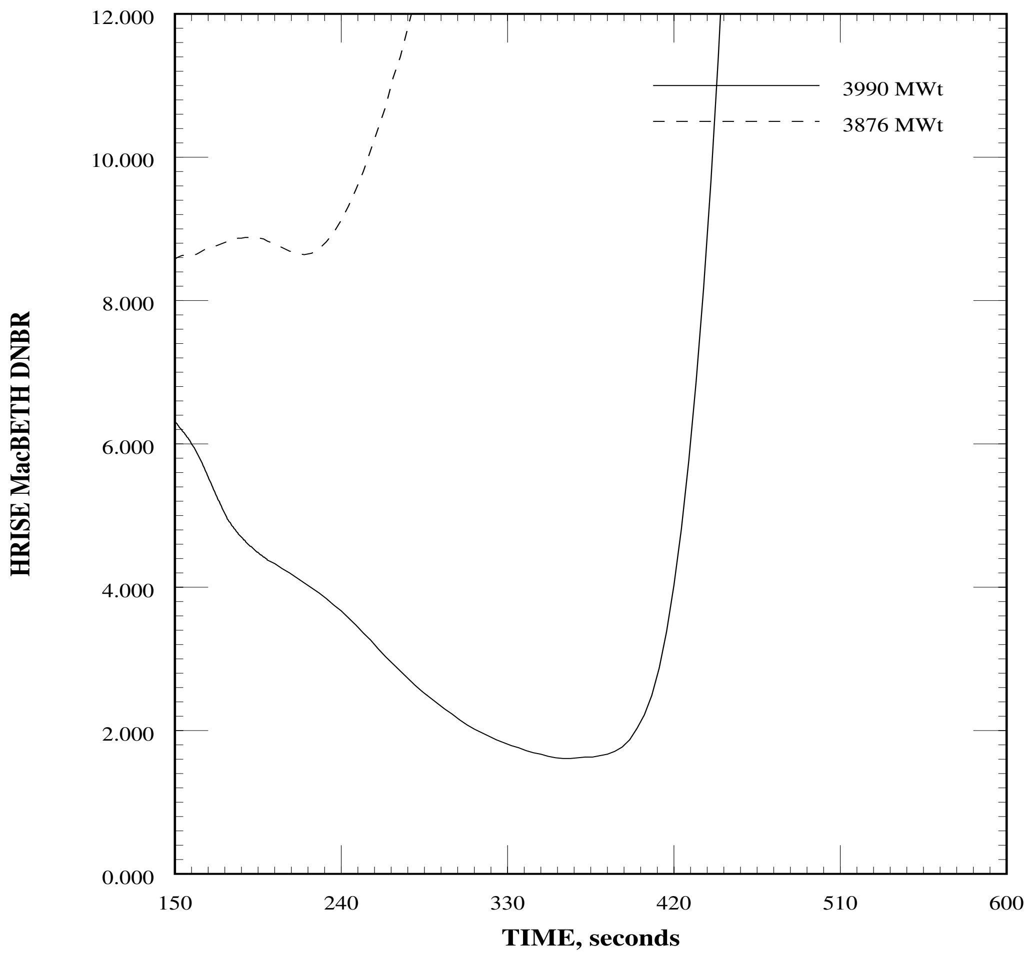
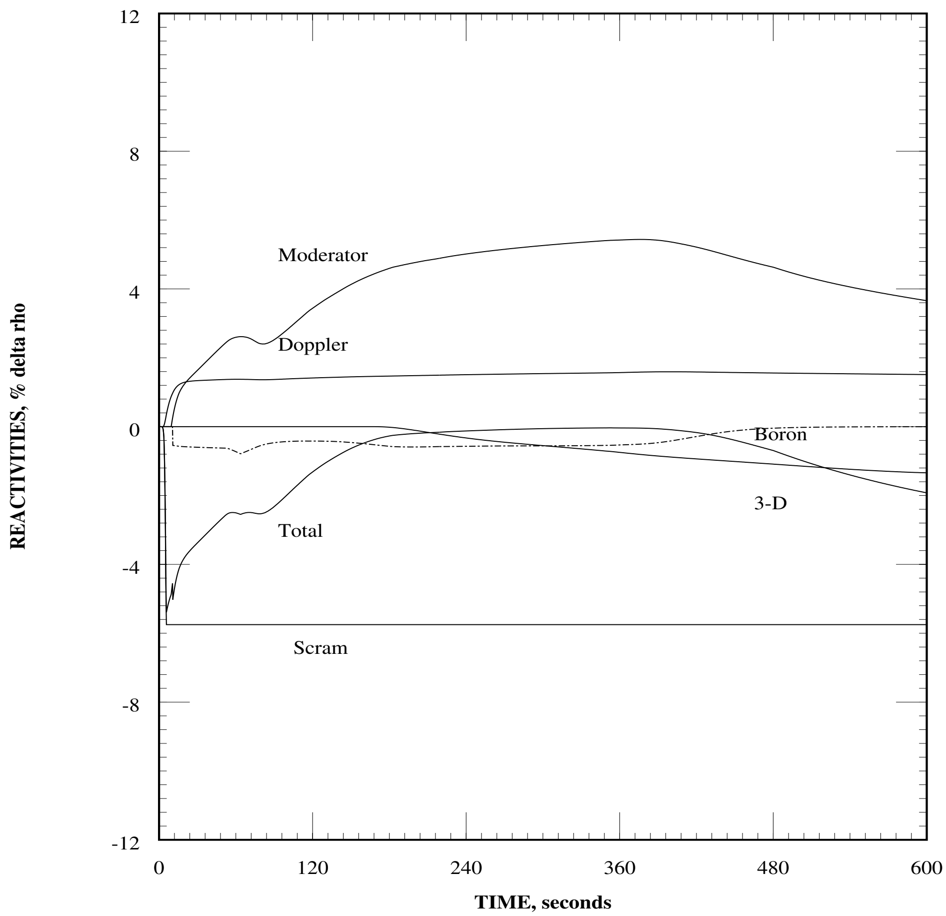


Figure 6.3-41
MSLB HFP with LOP - Core Reactivity with HERMITE 3D Credit vs. Time



Section 6.3.1.6 Steam System Piping Failures Inside and Outside Containment - Mode 3 Operation

Section 6.3.1.6.1 Identification of Event and Causes

As described in UFSAR Section 15.1.6, MSLB events during Mode 3 operation are analyzed to demonstrate the adequacy of the shutdown margin, as specified by Technical Specifications, to prevent degradation in fuel performance because of a post-trip R-t-P. The following inside containment SLBs evaluated were:

1. A large SLB during Mode 3 operation with a concurrent LOP in combination with a single failure and shutdown margin.
2. A large SLB during Mode 3 operation with offsite power available in combination with a single failure and shutdown margin.

See Section 6.3.1.5.1 for further information of the MSLB events.

Section 6.3.1.6.2 Acceptance Criteria

Section 6.3.1.5.2 lists the SRP criteria for the MSLB events.

Section 6.3.1.6.3 Description of Analysis

The NSSS response to Mode 3 MSLB events was simulated using the CENTS code. The minimum DNBR was calculated using the HRISE code. A hand calculation was performed to determine the Linear Heat Generation Rate (LHGR) at the time of the maximum post-trip R-t-P.

Both the existing configuration and PUR were evaluated in order to compare the NSSS response to the MSLB event. Input parameters and initial conditions were selected to maximize the R-t-P and demonstrate the adequacy of shutdown margin.

SLBs are characterized as cooldown events due to the increased steam flowrate that causes excessive energy removal from the SGs and the RCS. This results in a decrease in RCS temperatures and in RCS and SG pressures. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Mode 3 SLBs are initiated from a subcritical reactivity condition. Detection of the cooldown is by the pressurizer and SG low-pressure alarms, by the high reactor power alarm and the low SG water level alarm. Reactor trip is provided by one of two available reactor trip signals. These are the LSGP and the HLPT.

The SLB that occurs with a concurrent LOP includes coastdown of the RCPs. The depressurization of the affected SG results in actuation of a MSIS. This closes the MSIVs, isolating steam flow from the unaffected SG. After closure of the MSIVs, the SGs reach the SG differential pressure setpoint. At this setpoint, the AFW system will not allow automatic delivery to the lower pressure (affected) SG. This differential pressure setpoint is exceeded before the level in the affected SG drops below the AFAS setpoint. Therefore, there is no significant delivery of AFW to the affected SG.

The pressurizer pressure eventually decreases and a SIAS is initiated. The isolation of the unaffected SG and subsequent emptying of the affected SG terminates the cooldown. The introduction of SI boron upon SIAS causes core reactivity to decrease.

A parametric study of single failures that would have an adverse impact on the SLB event determined that the failure of one HPSI pump to start has the most adverse effect for those cases that result in generation of SIAS.

Section 6.3.1.6.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-17 and Table 6.3-18 contain the initial conditions used for the Mode 3 MSLB event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The initial reactivity state of the core is consistent with the core being subcritical by the minimum creditable negative reactivity between the “Reactor Trip Breaker Open” Shutdown Margin and the “Reactor Trip Breaker Closed” Shutdown Margin.
3. There is no operator action for the first 30 minutes of the event.

Table 6.3-17
Parameters Used for Mode 3 MSLB with LOP Event

PARAMETER	Value	
	$3876\ MW_t$	$3990\ MW_t$
Initial core power (% of rated)	1.23E-10	1.23E-10
Initial core inlet temperature (°F)	575	575
Initial pressurizer pressure (psia)	1700	1700
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
FTC	most negative	most negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-6.5	-6.5
Inverse boron worth (ppm/%$\Delta\rho$)	130	130
Fuel rod gap conductance (Btu/hr-ft^2-°F)	5755	5755
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	1 HPSI pump	1 HPSI pump
LOP	yes	yes

Table 6.3-18
Parameters Used for Mode 3 MSLB without LOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	1.23E-10	1.23E-10
Initial core inlet temperature (°F)	575	575
Initial pressurizer pressure (psia)	1700	1700
Initial RCS Flow (% of design)	95	95
Initial pressurizer level (ft)	25.6	25.6
Initial SG level (ft)	41.8	45.3
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
FTC	most negative	most negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-6.5	-6.5
Inverse boron worth (ppm/%$\Delta\rho$)	130	130
Fuel rod gap conductance (Btu/hr-ft^2-°F)	5755	5755
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	1 HPSI pump	1 HPSI pump
LOP	no	no

Section 6.3.1.6.5 Results

Table 6.3-19 and Table 6.3-20 present a sequence of events occurring until operator action is initiated at 30 minutes for Mode 3 MSLB with and without LOP, respectively. Figure 6.3-42 through Figure 6.3-51 present the behavior of NSSS parameters following the Mode 3 MSLB with a LOP event. Figure 6.3-52 through Figure 6.3-61 illustrates the NSSS response without a concurrent LOP.

The Mode 3 MSLB with a LOP is the limiting case. Upon a LOP, the CEA holding coils de-energize, allowing the scram rods to drop into the core. The LOP is simulated in this analysis by tripping the RCPs at the beginning of the transient.

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). In general, the larger initial inventory associated with the larger SGs results in further RCS cooldown and thus requires more shutdown margin.

Since no R-t-P results from the Mode 3 MSLB events, the Technical Specification shutdown margin requirements remain bounding following PUR.

Table 6.3-19
Sequence of Events for Mode 3 MSLB with LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	SLB occurs	---	---
0.1	0.1	LOP occurs	---	---
8.6	7.9	SG pressure reaches MSIS setpoint (psia)	810	875
8.6	7.9	MSIVs begin to close	---	---
10.1	9.4	Reactor trip breakers open	---	---
10.1	9.4	Turbine trip occurs	---	---
14.1	13.5	MSIVs are closed	---	---
14.2	13.6	Unaffected SG flow stops	---	---
29.1	27.1	SIAS trip setpoint reached (psia)	1300	1300
59.1	57.1	SI pumps reach full speed	---	---
115.5	113.3	Boron reaches RCS	---	---
308.0	387.0	Maximum total reactivity (%$\Delta\rho$)	0.285	0.224
437.0	478.0	Maximum heat flux fraction	0.025	0.012
439.0	480.0	Maximum core power fraction	0.024	0.011
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Table 6.3-20
Sequence of Events for Mode 3 MSLB without LOP

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	SLB occurs	---	---
9.0	8.3	SG pressure reaches MSIS setpoint (psia)	810	875
9.0	8.3	MSIVs begin to close	---	---
10.5	9.8	Reactor trip breakers open	---	---
10.5	9.8	Turbine trip occurs	---	---
14.6	13.9	MSIVs are closed	---	---
14.7	13.9	Unaffected SG flow stops	---	---
21.6	20.2	SIAS Setpoint reached (psia)	1300	1300
51.6	50.2	SI pumps reach full speed	---	---
53.5	100.0	Maximum heat flux fraction	0.003	0.003
104.6	103.9	Boron reaches RCS	---	---
346.0	389.0	Maximum total reactivity (%$\Delta\rho$)	0.285	0.086
353.0	394.0	Maximum core power fraction	<0.001	<0.001
380.0	425.0	SG dryout	---	---
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.1.6.6 Conclusions

For both the Mode 3 SLB events with and without a LOP, all acceptance criteria are met. The peak primary and secondary pressures remain below 110% of design at all times thus ensuring the integrity of the RCS or the main steam system. The minimum DNBR remains above the safety analysis limit thus ensuring fuel cladding integrity.

Figure 6.3-42
Mode 3 MSLB with LOP - T_{cold} vs. Time

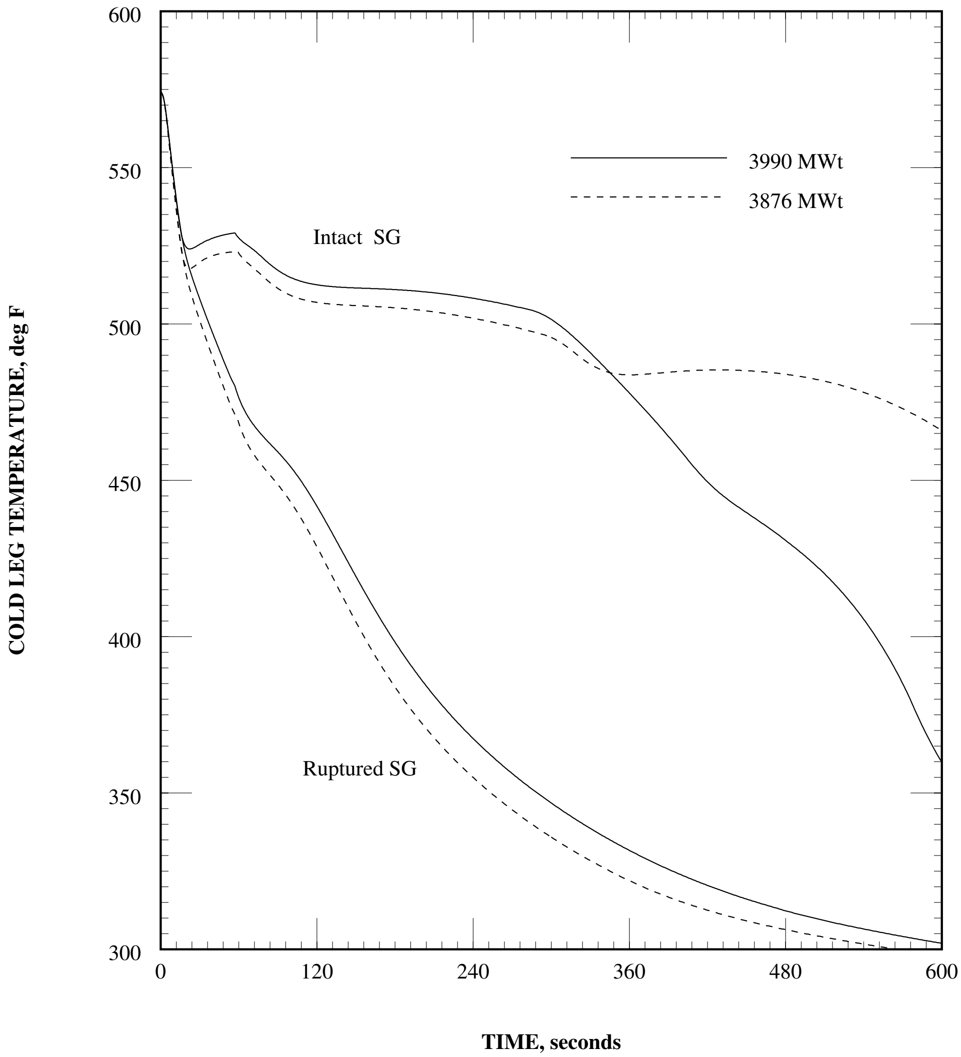


Figure 6.3-43
Mode 3 MSLB with LOP - Core Average Heat Flux vs. Time

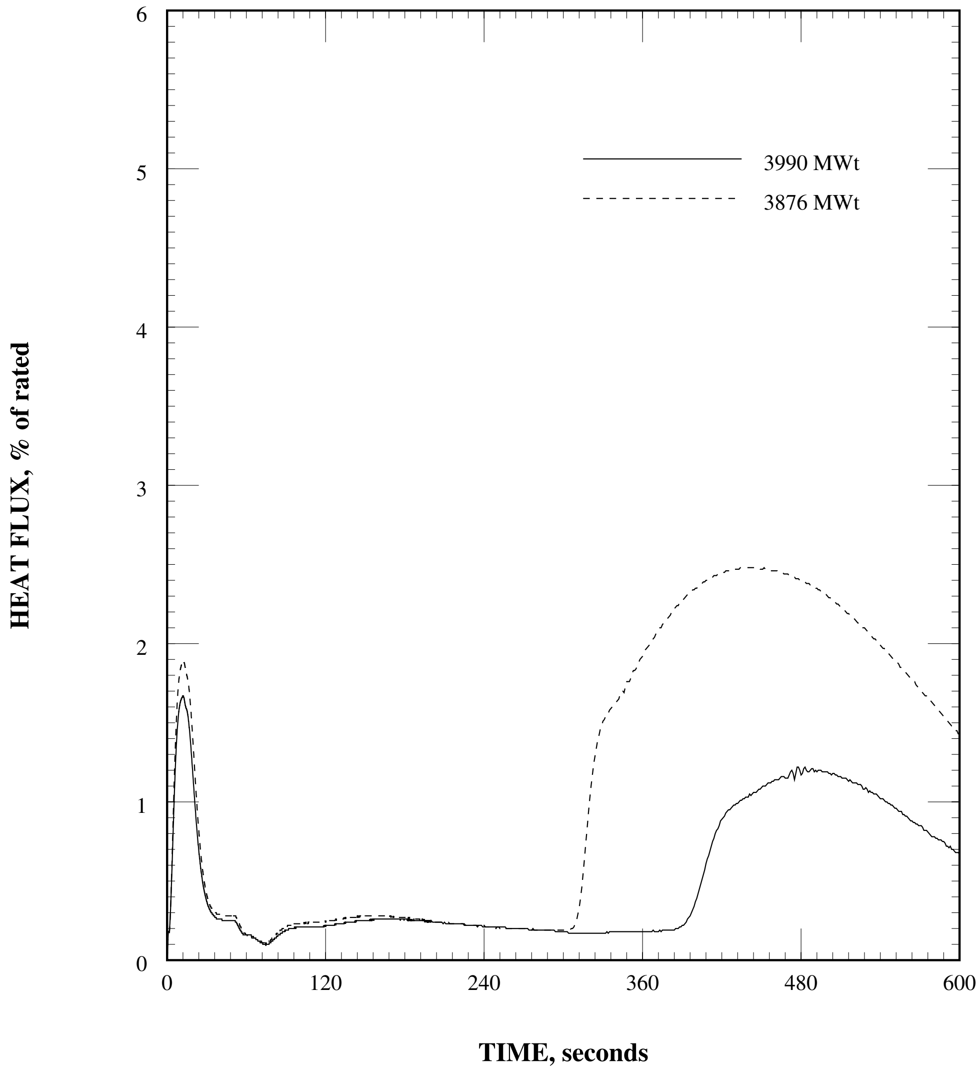


Figure 6.3-44
Mode 3 MSLB with LOP - Core Power vs. Time

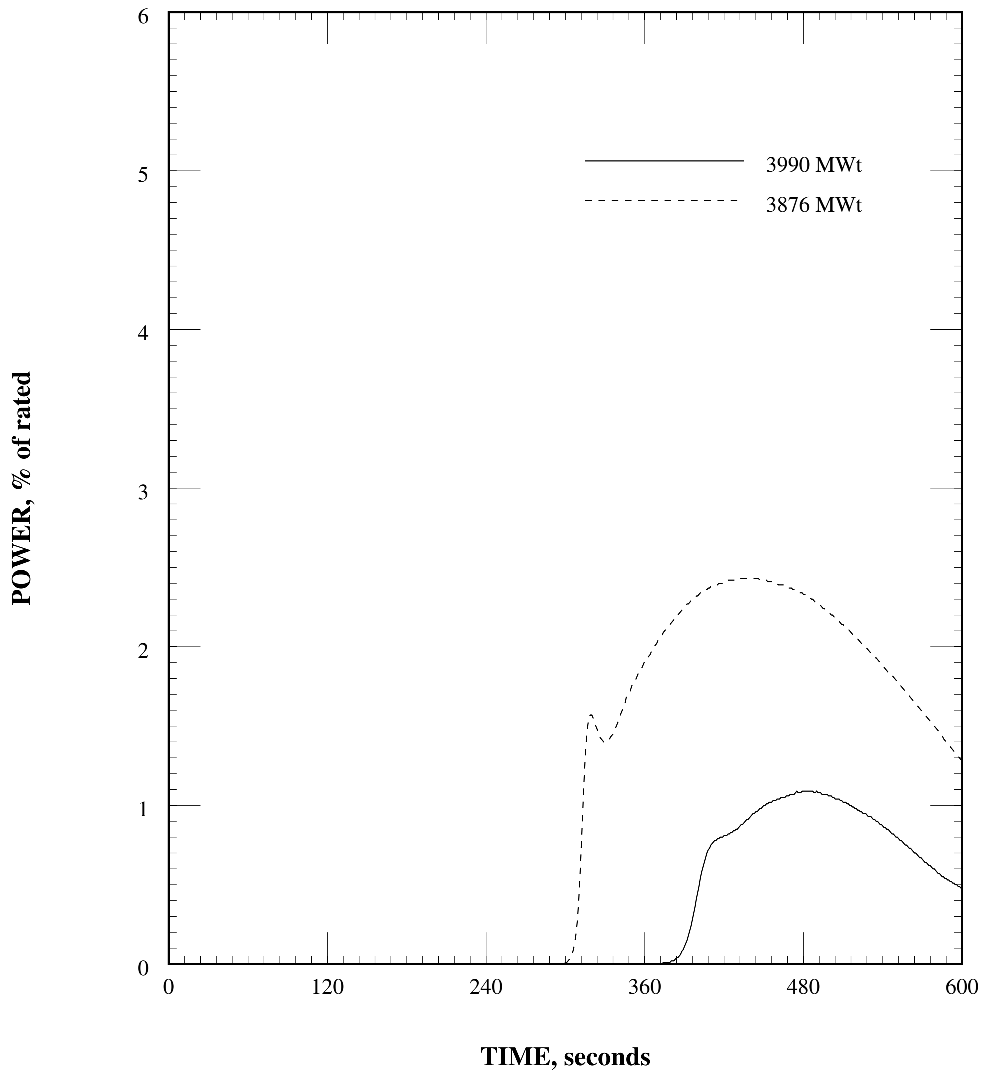


Figure 6.3-45
Mode 3 MSLB with LOP - Pressurizer Pressure vs. Time

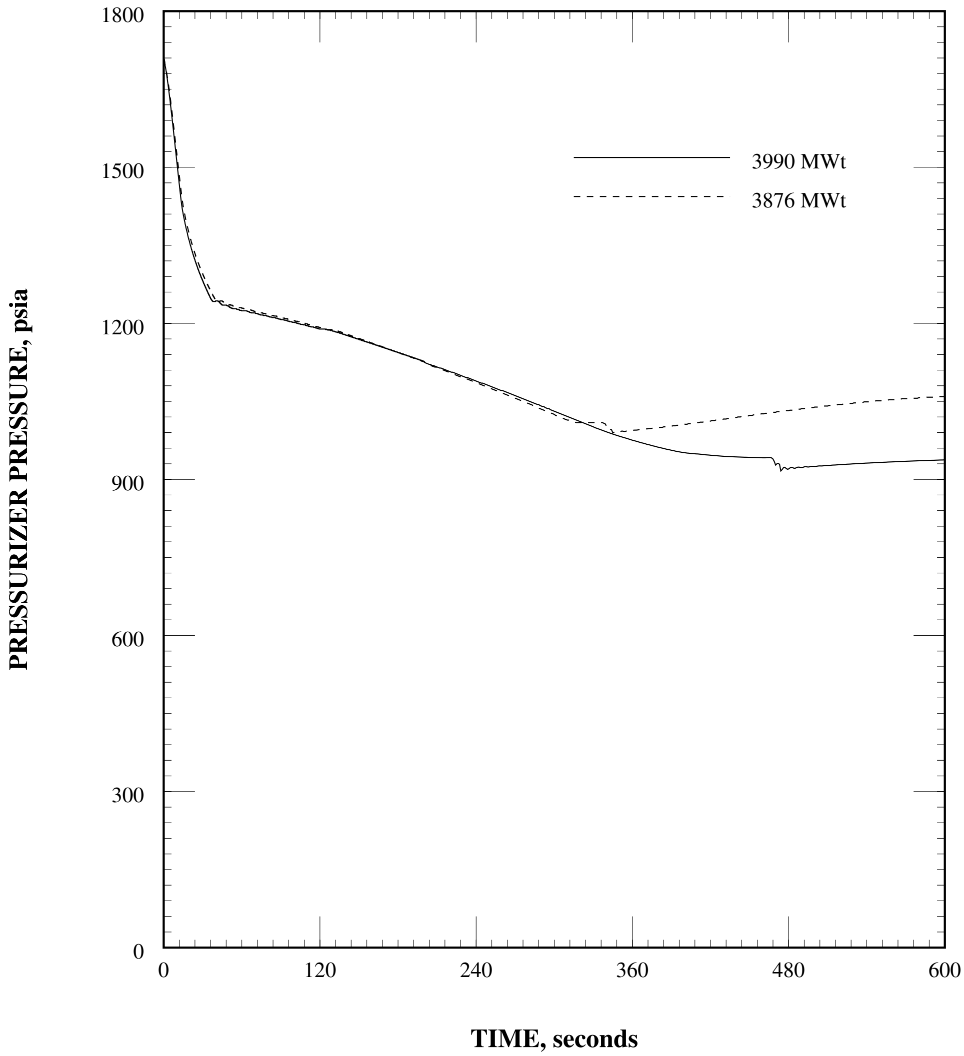


Figure 6.3-46
Mode 3 MSLB with LOP - RCS Pressure vs. Time

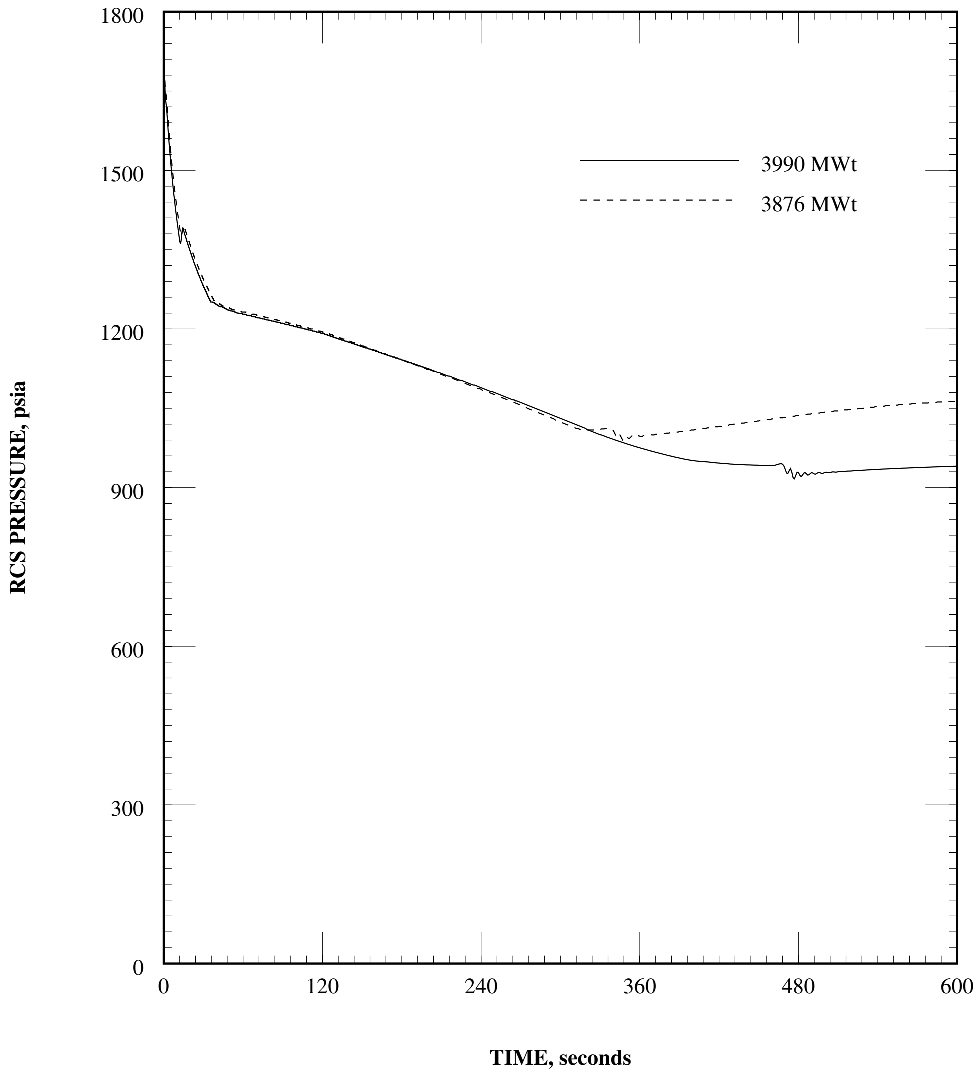


Figure 6.3-47
Mode 3 MSLB with LOP - Core Reactivity vs. Time

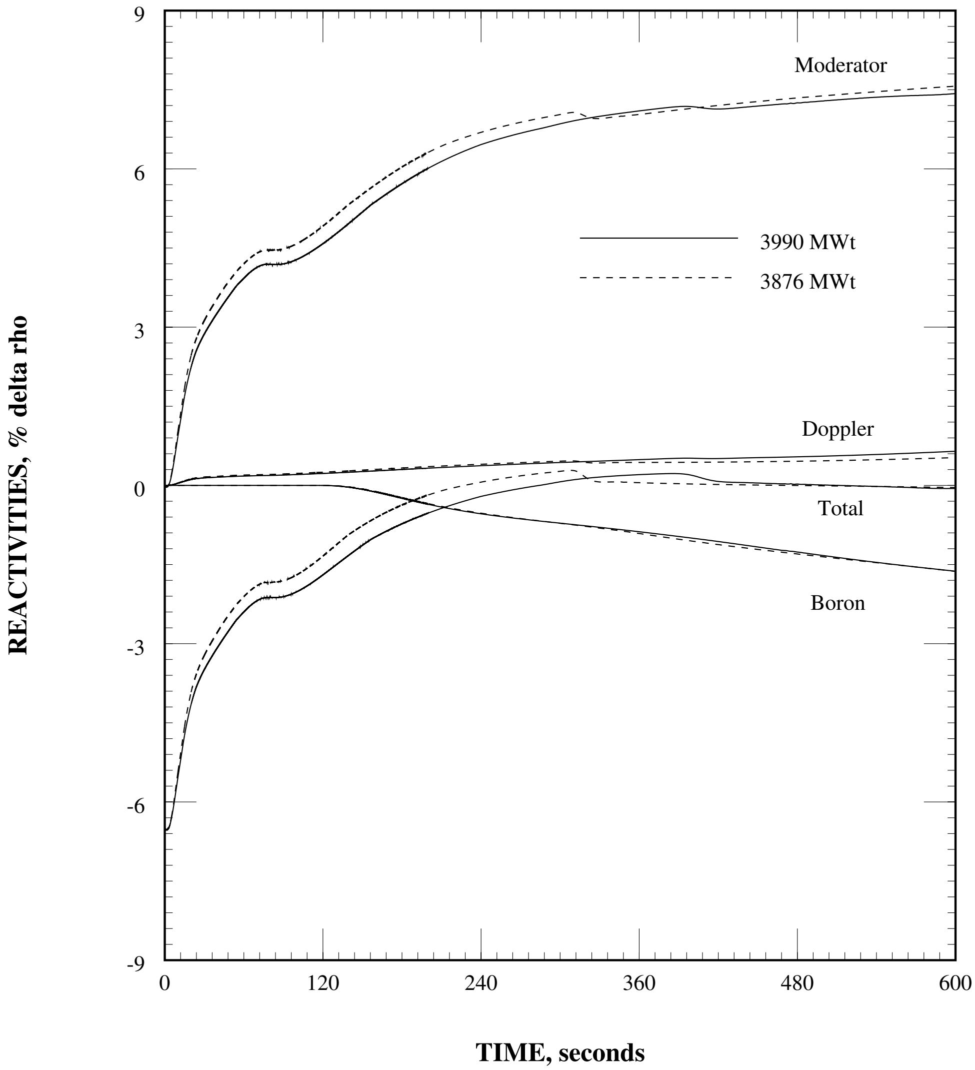


Figure 6.3-48
Mode 3 MSLB with LOP - Total SG Liquid Mass vs. Time

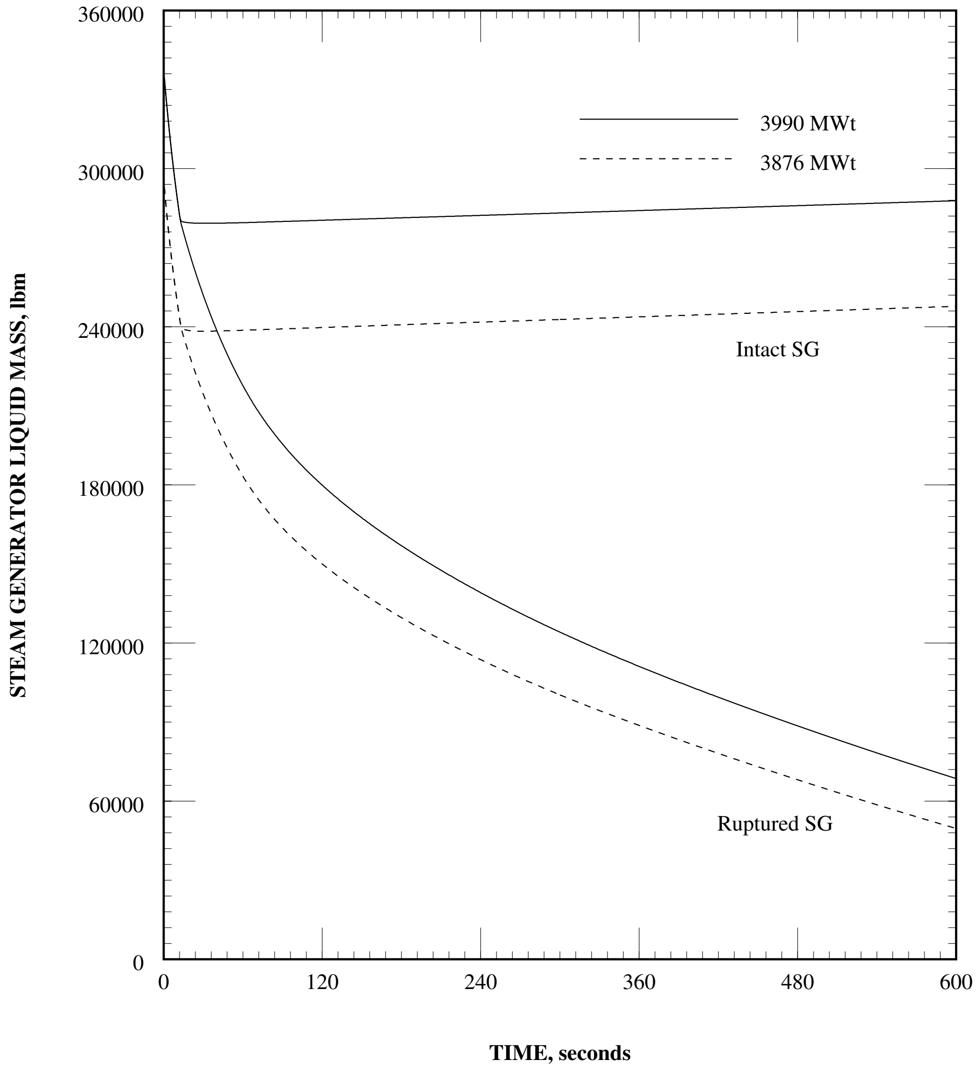


Figure 6.3-49
Mode 3 MSLB with LOP - SG Pressure vs. Time

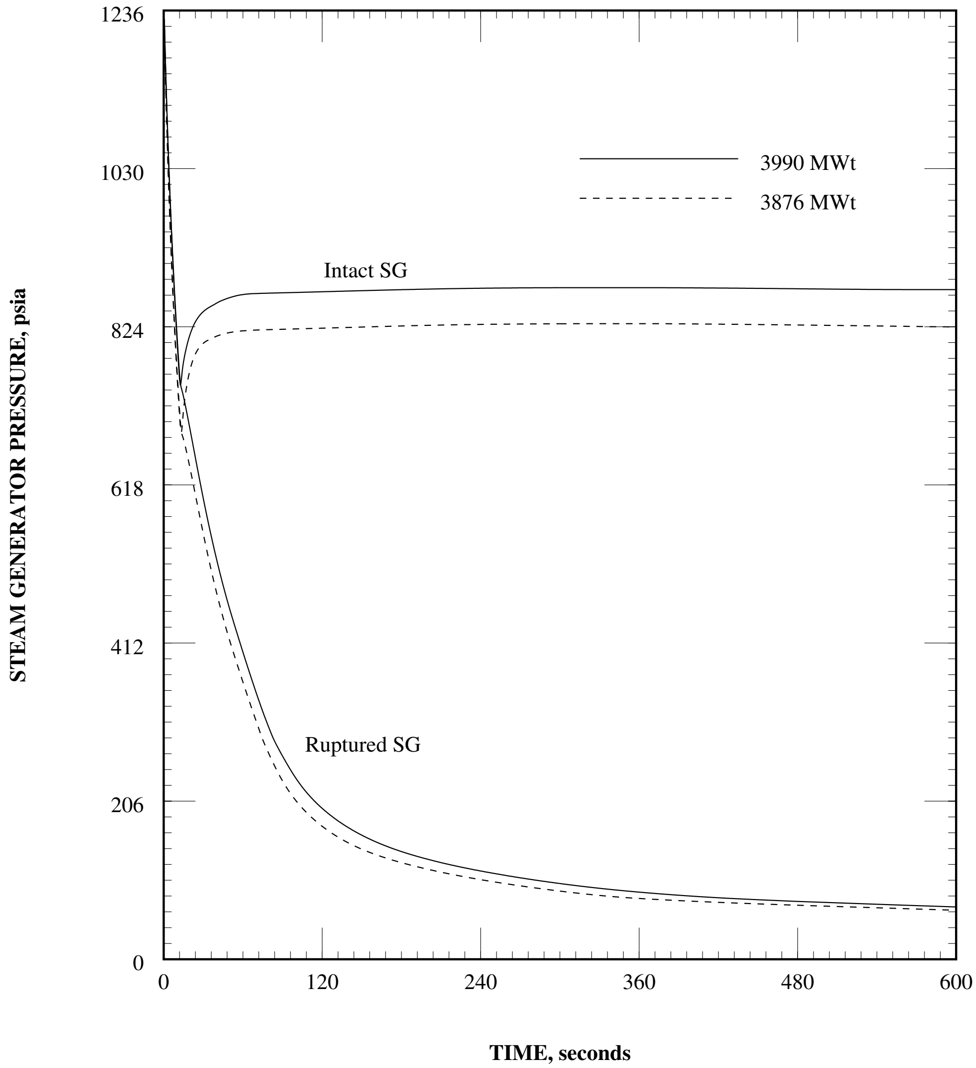


Figure 6.3-50
Mode 3 MSLB with LOP - SG Steam Flow vs. Time

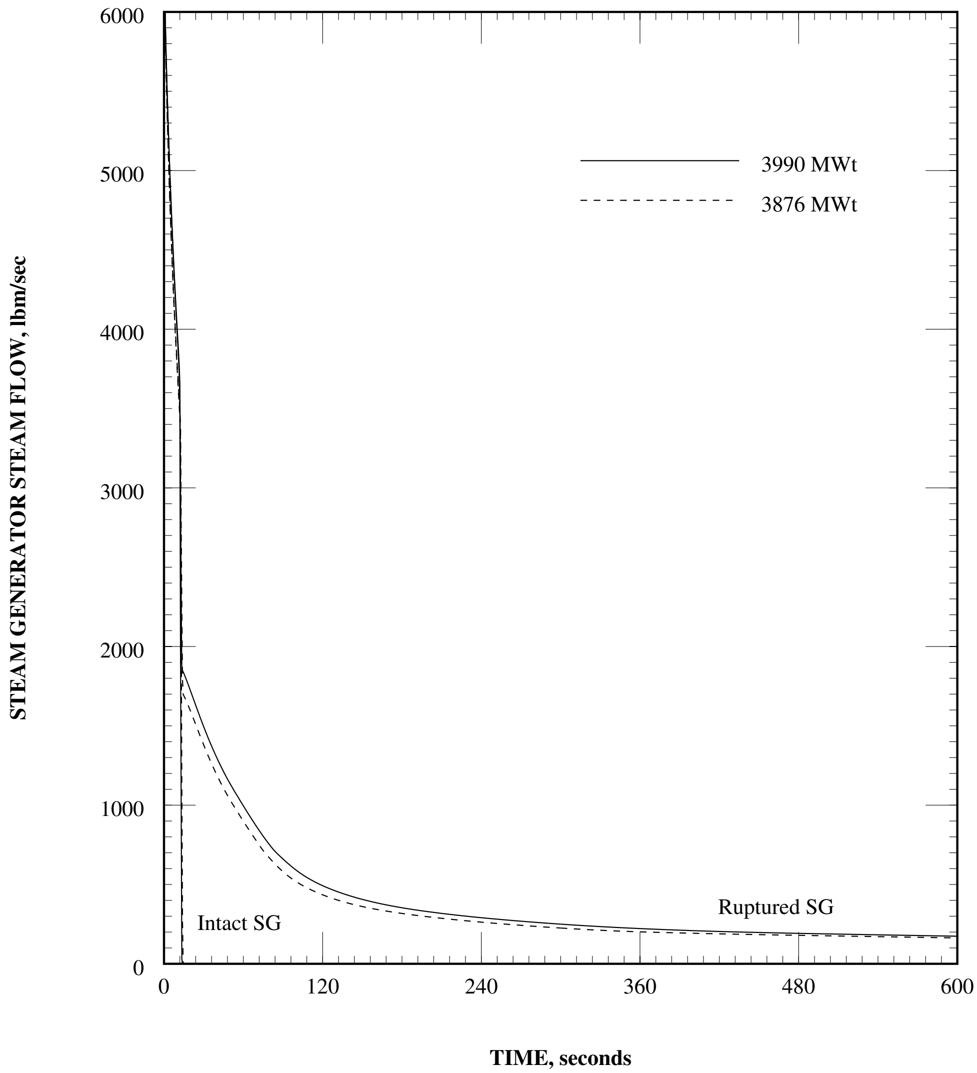


Figure 6.3-51
Mode 3 MSLB with LOP - RCS Temperatures vs. Time

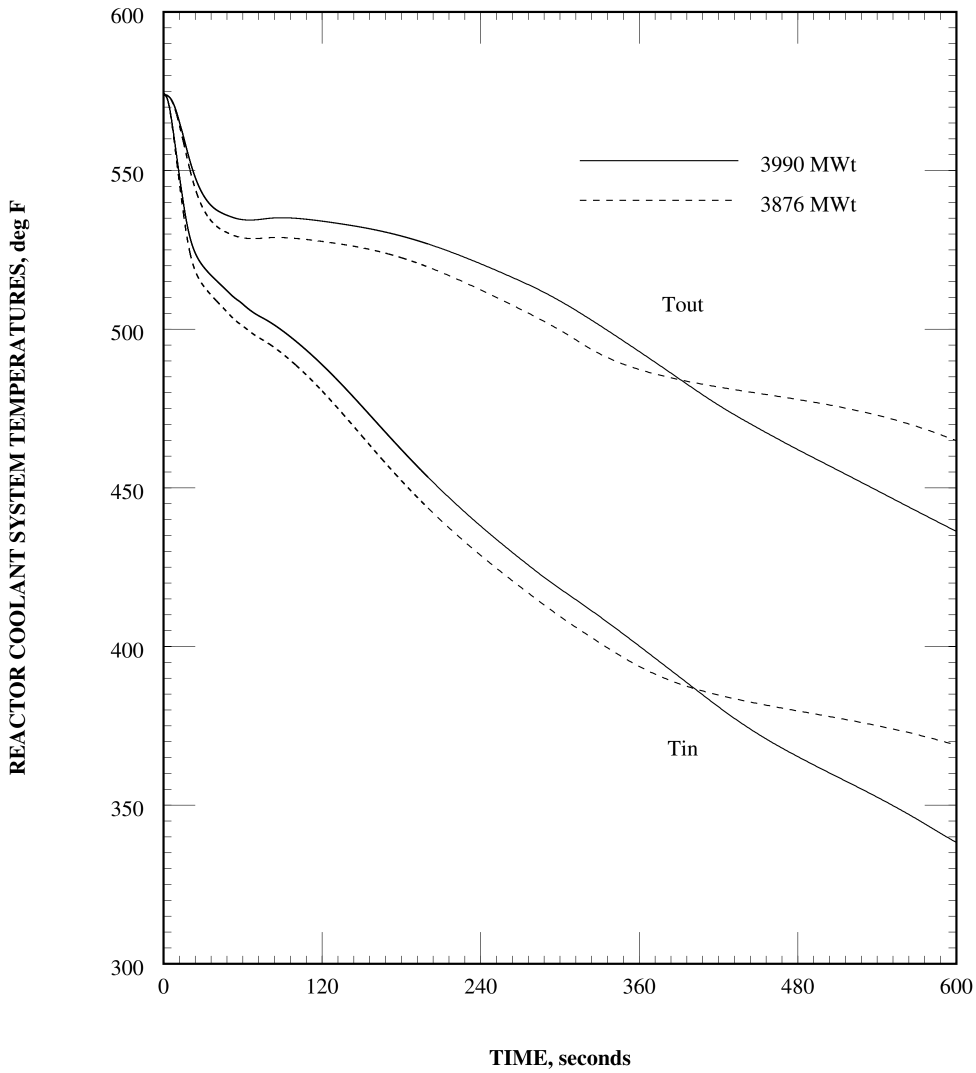


Figure 6.3-52
Mode 3 MSLB with LOP - T_{cold} vs. Time

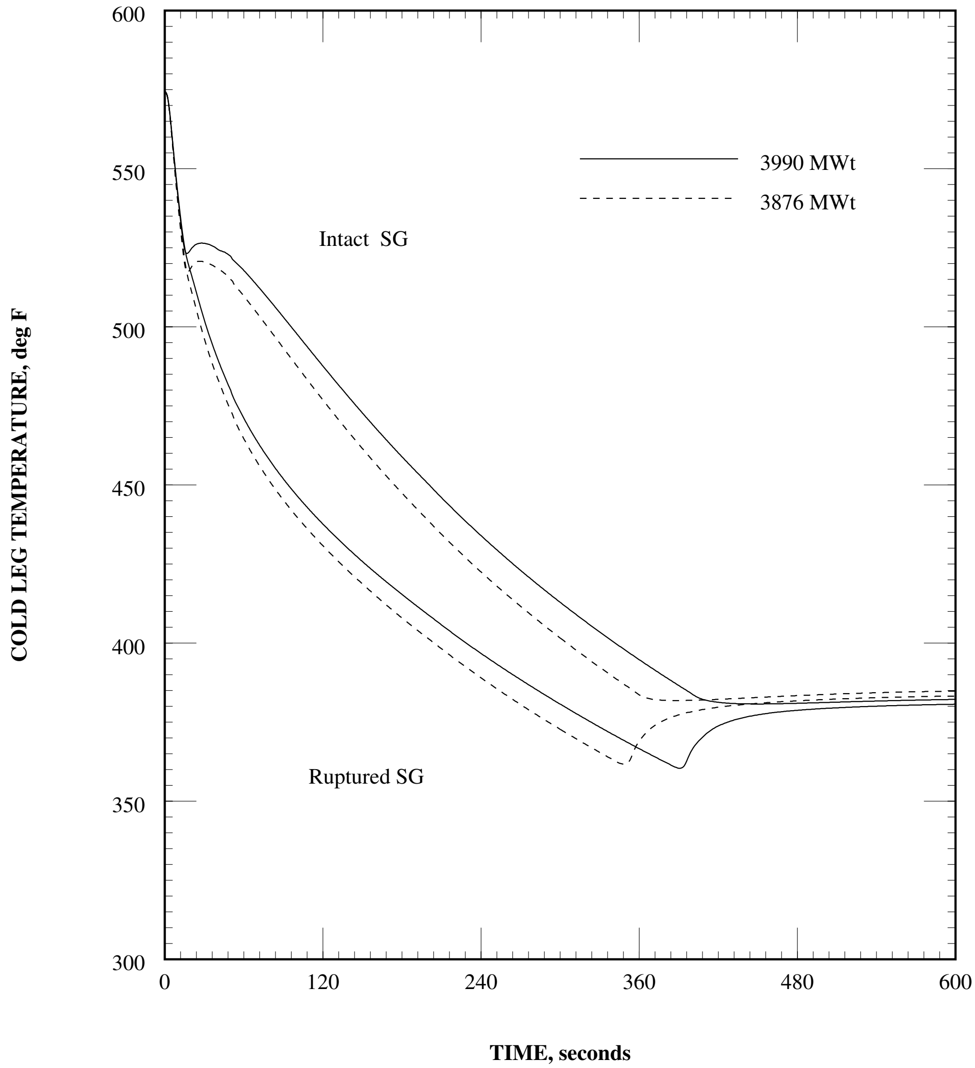


Figure 6.3-53
Mode 3 MSLB without LOP - Core Average Heat Flux vs. Time

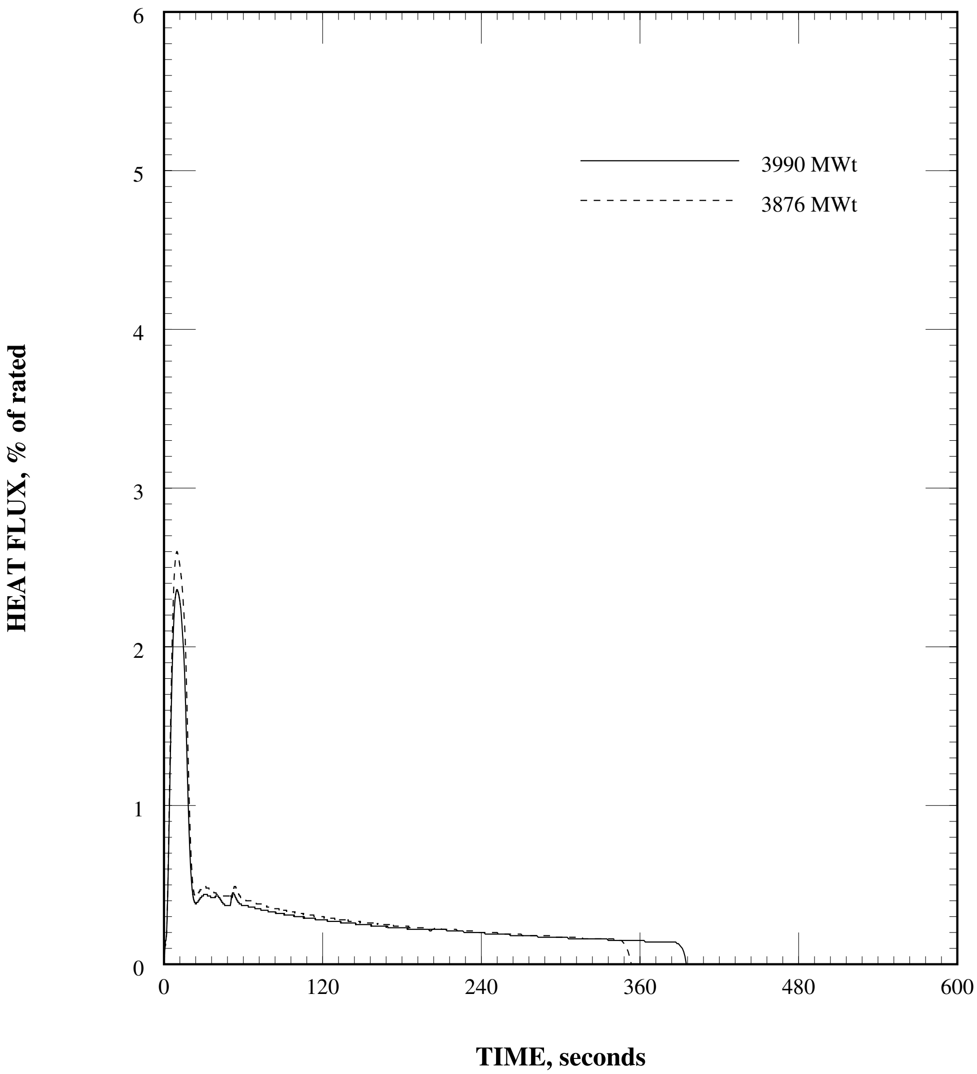


Figure 6.3-54
Mode 3 MSLB without LOP - Core Power vs. Time

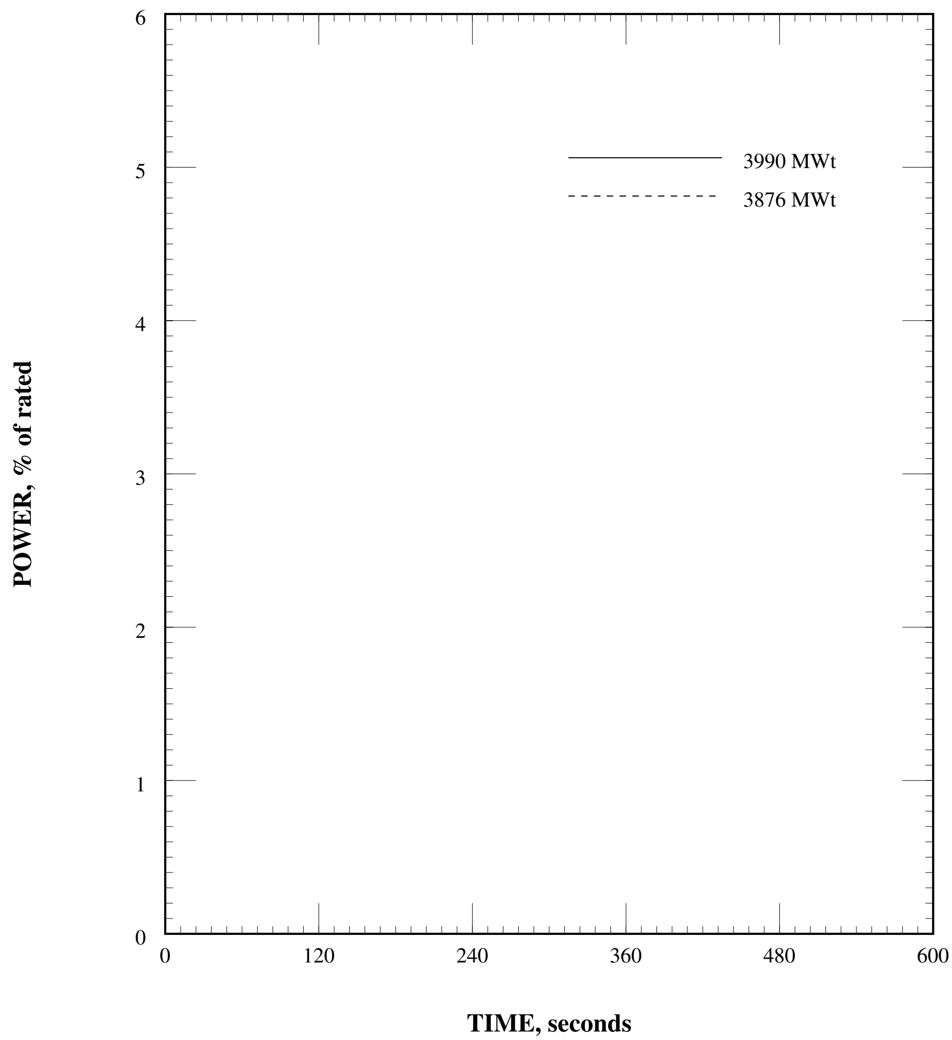


Figure 6.3-55
Mode 3 MSLB without LOP - Pressurizer Pressure vs. Time

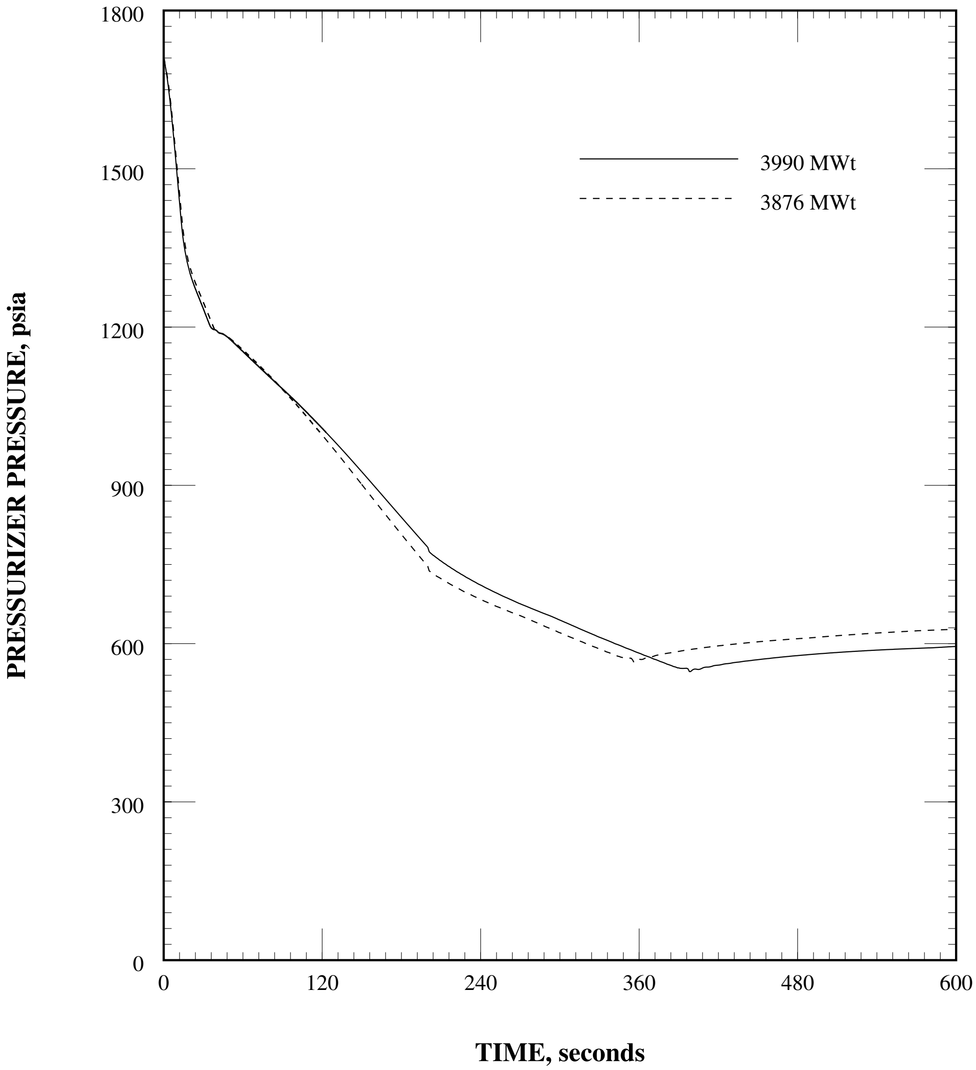


Figure 6.3-56
Mode 3 MSLB without LOP - RCS Pressure vs. Time

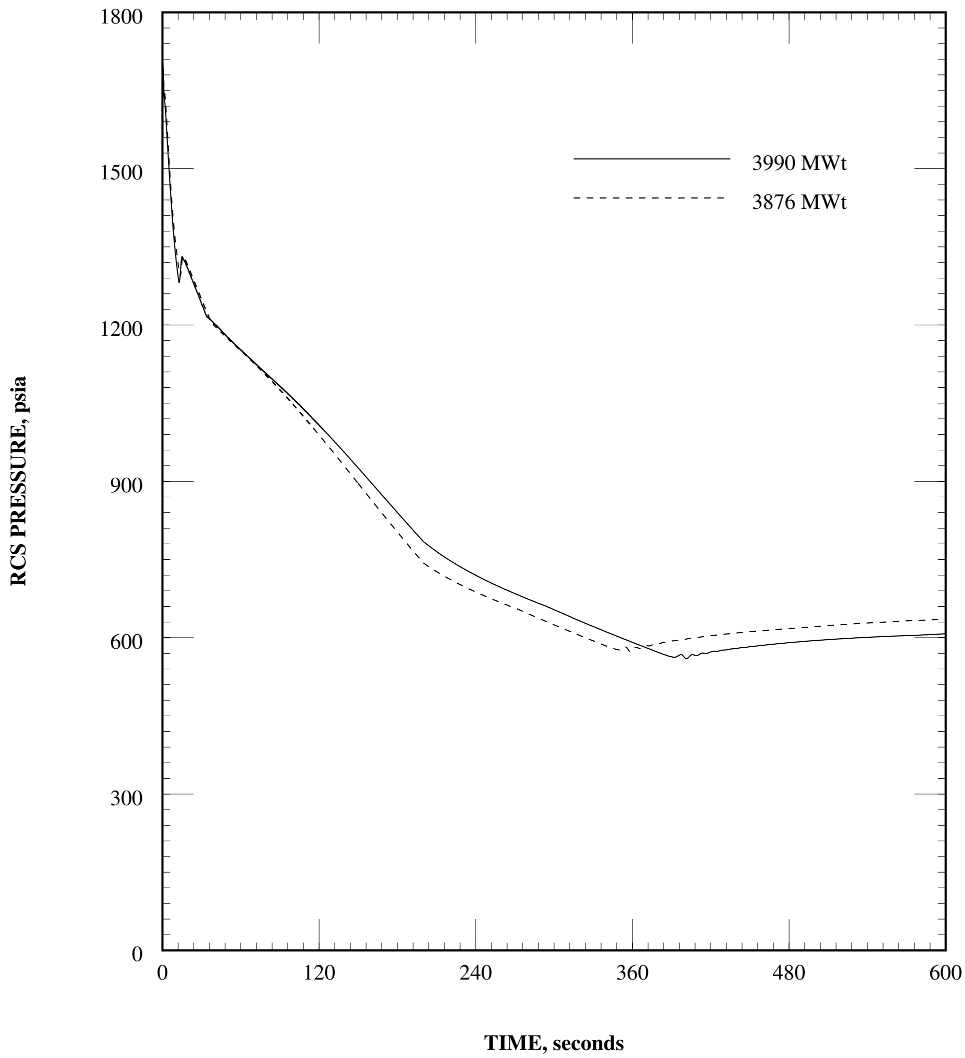


Figure 6.3-57
Mode 3 MSLB without LOP - Core Reactivity vs. Time

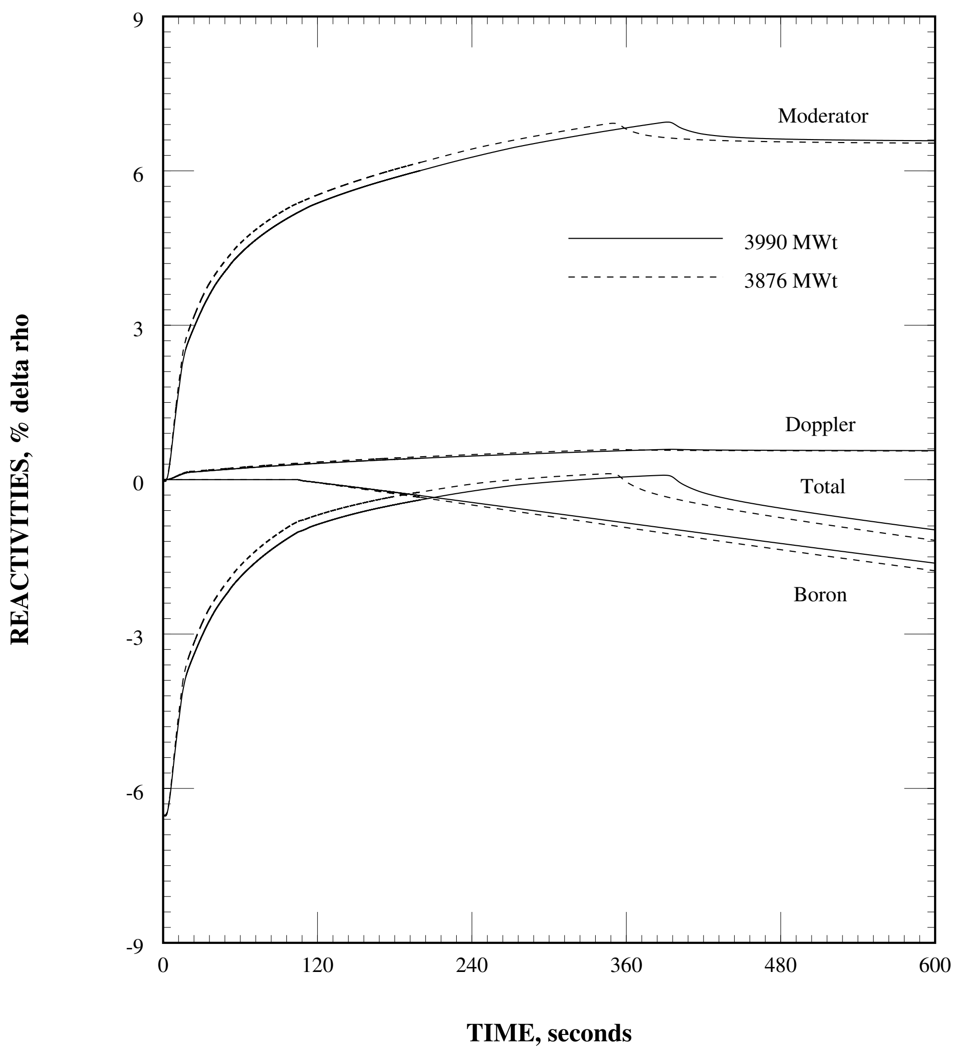


Figure 6.3-58
Mode 3 MSLB without LOP - Total SG Liquid Mass vs. Time

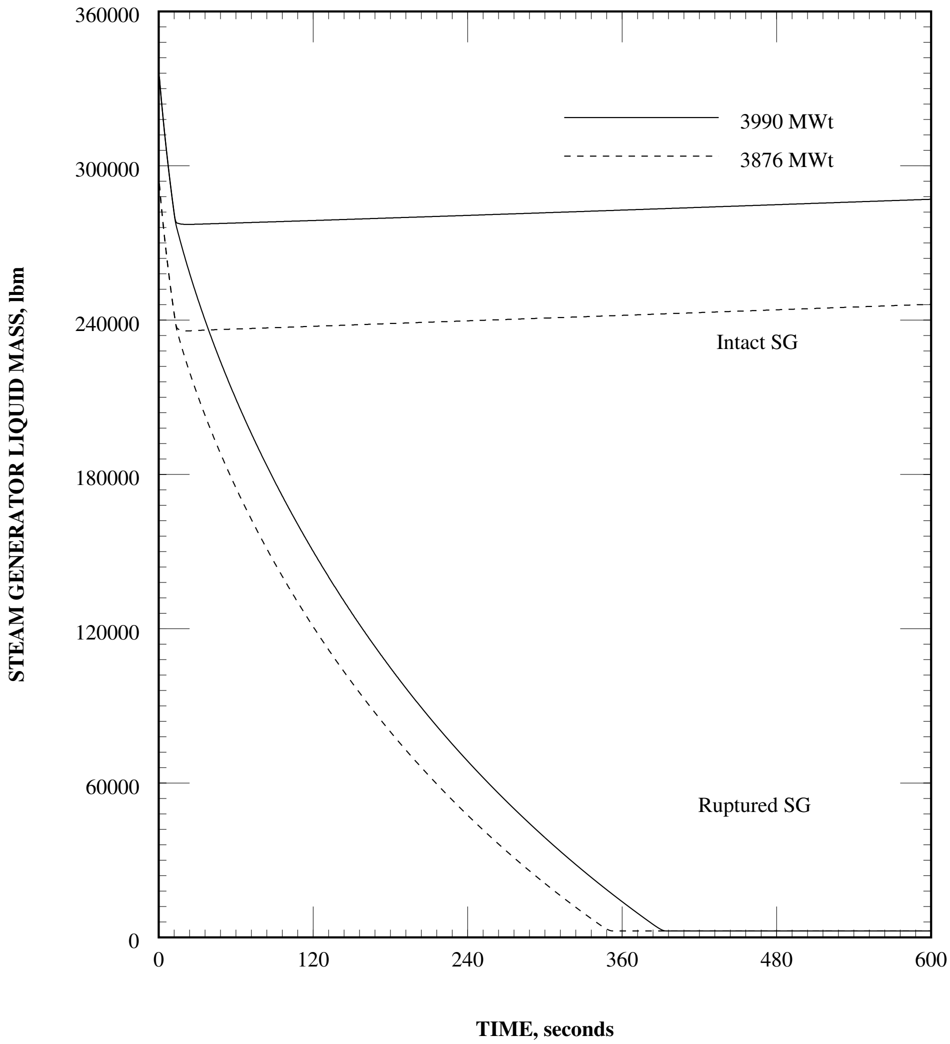


Figure 6.3-59
Mode 3 MSLB without LOP - SG Pressure vs. Time

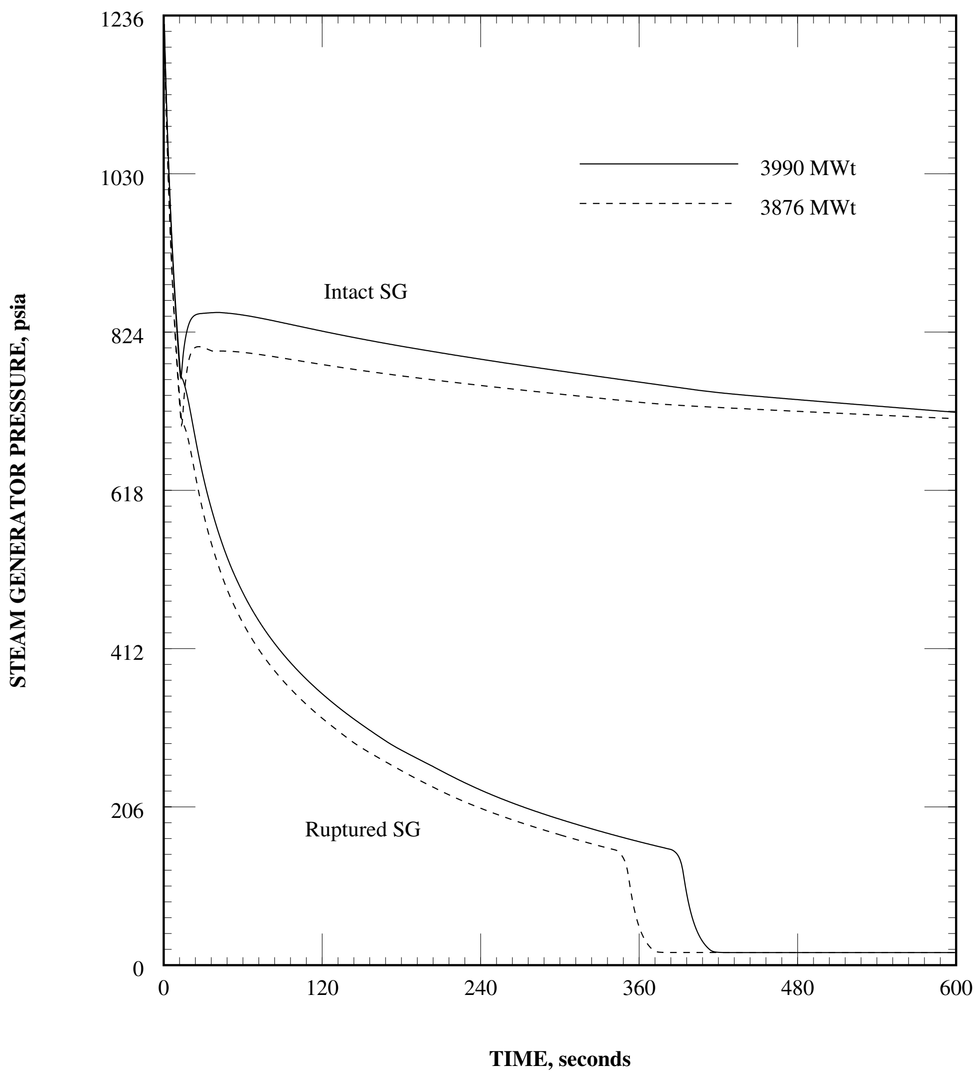


Figure 6.3-60
Mode 3 MSLB without LOP - SG Steam Flow vs. Time

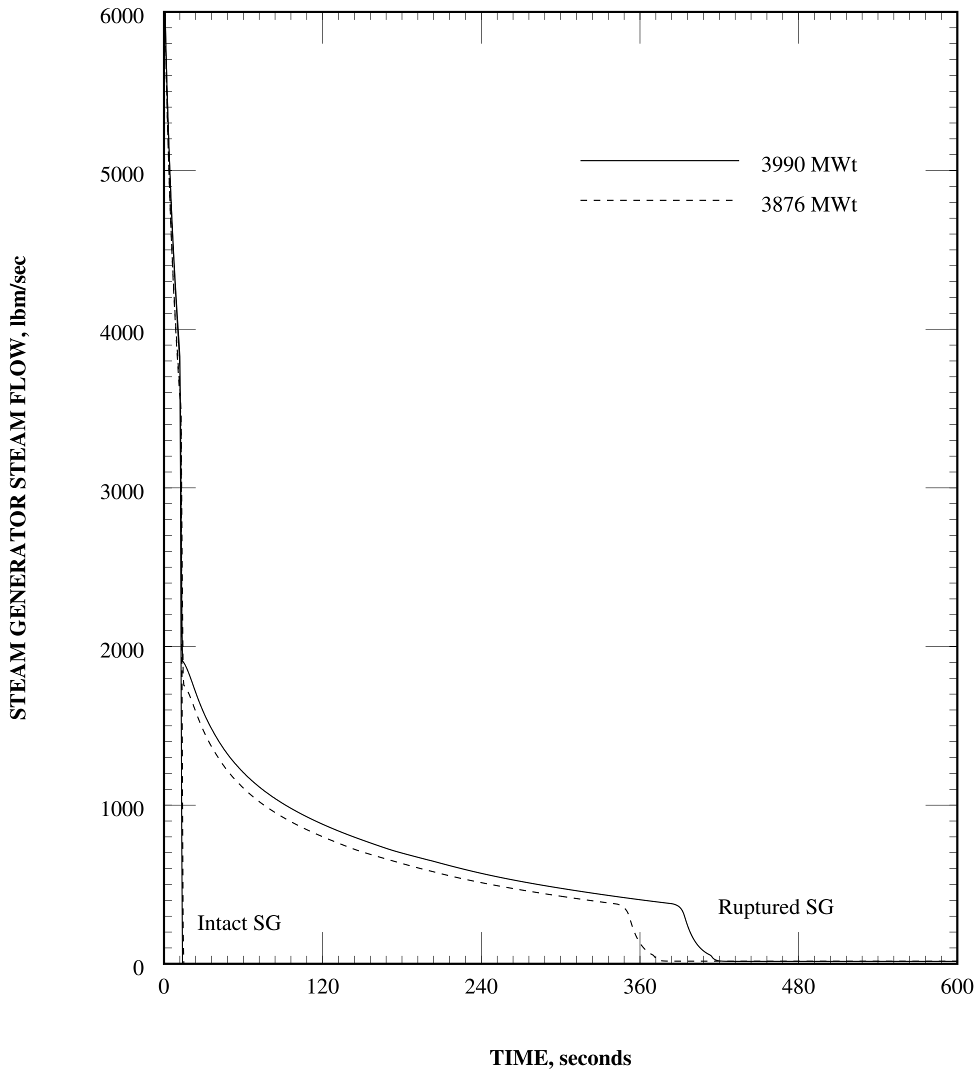
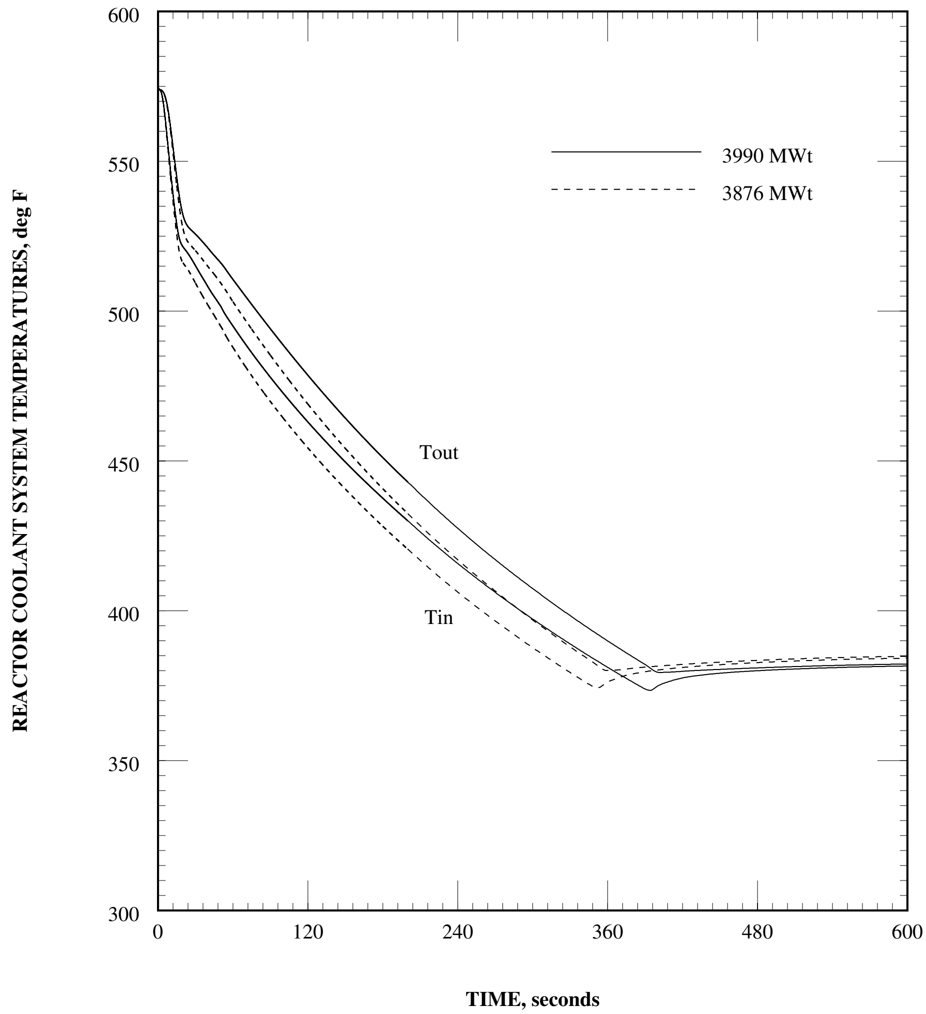


Figure 6.3-61
Mode 3 MSLB without LOP - RCS Temperatures vs. Time



Section 6.3.1.7 Pre-Trip Main Steam Line Break Power Excursion

Section 6.3.1.7.1 Identification of Event and Causes

Section 6.3.1.5.1 details the event of a pipe break in the main steam system. The previous sections detail SLBs that targeted R-t-P. This section presents a SLB event designed to maximize DNBR degradation and challenge fuel integrity before the reactor trip.

Section 6.3.1.7.2 Acceptance Criteria

Section 6.3.1.5.2 lists the SRP criteria for the MSLB events.

Section 6.3.1.7.3 Description of Analysis

The NSSS responses to several MSLB events were simulated using the CENTS code. The initial thermal margin and transient DNBR was calculated using the CETOP-D code which uses the CE-1 CHF correlation. At the time of minimum DNBR, a more accurate prediction of the DNBR was calculated using the more detailed TORC code.

Both the existing configuration and PUR were evaluated in order to compare the NSSS response to the MSLB event.

Initial plant conditions and event initiators (break size, time of life, etc.) were varied to obtain the most adverse power excursion and fuel performance degradation event.

The addition of the CPC VOPT has provided an early trip for smaller SLB sizes. However, for larger breaks and more negative MTC values, a sufficient power overshoot may occur prior to event turnaround to still result in fuel pins falling below the DNBR limit. The most adverse case is the combination of break size and MTC that produces the highest power level before reactor trip.

Other plant and core characteristics are chosen conservatively with either the time in life (as indicated by the most adverse MTC) or with initial conditions required to be at a POL as determined by the COLSS.

The following RCP scenarios are used for the pre-trip SLB:

1. offsite power maintained throughout the transient and
2. LOP concurrent with break initiation.

The limiting scenario is the full power event with offsite power available.

Section 6.3.1.7.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-21 contains the initial conditions used for the limiting pre-trip MSLB outside containment without LOP event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. An initial power of 95% of rated was selected. At this power, plant LCOs provide the same initial thermal margin to the DNB SAFDL as full power, but the CPC VOPT setpoint has room to increase in response to increasing core power caused by the SLB. These two effects are combined to make the selection of 95% power more adverse.
3. There is no operator action for the first 30 minutes of the event.

Table 6.3-21
Parameters Used for Pre-Trip MSLB Outside Containment Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	95	95
Initial core inlet temperature (°F)	562	568
Initial pressurizer pressure (psia)	2250	2250
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.0	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6984	6984
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

<u>Section 6.3.1.7.5</u>   <u>Results</u>

Table 6.3-22 presents a sequence of events that occur following the limiting pre-trip MSLB outside containment event. The limiting scenario is the full power event with

offsite power available. Figure 6.3-62 through Figure 6.3-66 presents the behavior of NSSS parameters following the MSLB event.

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).

Radiological consequences for this event are presented in Section 6.4.1.2.

Table 6.3-22
Sequence of Events for Pre-Trip MSLB Outside Containment

Time (sec)			Value	
3876 MW_t	3990 MW_t	Event	3876 MW_t	3990 MW_t
0.00	0.00	Double-ended guillotine SLB	---	---
3.62	3.72	CPC VOPT reaches trip setpoint (% full power)	103.1	103.1
4.3	4.4	Reactor trip breakers open	---	---
4.3	4.4	Turbine trip occurs	---	---
4.8	4.9	Scram CEAs begin falling	---	---
5.7	5.8	Peak power (% full power)	118.3	118.6
6.5	6.8	Minimum DNBR occurs	1.34	1.35
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the IOSGADV + LOP event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.1.7.6 Conclusions

For the pre-trip MSLB power excursion event, all acceptance criteria are met. The peak primary and secondary pressures remained below 110% of design at all times ensuring the integrity of the RCS or the main steam system. No DNBR propagation is predicted to occur. Offsite doses remained below the acceptance criteria for this category of event:

- small fractions (less than 10%) of the 10 CFR Part 100 exposure guidelines and
- within 10 CFR Part 100 guidelines for the cases of a PIS or one control assembly held out of the core.

Figure 6.3-62
Limiting Pre-Trip MSLB - Core Power vs. Time

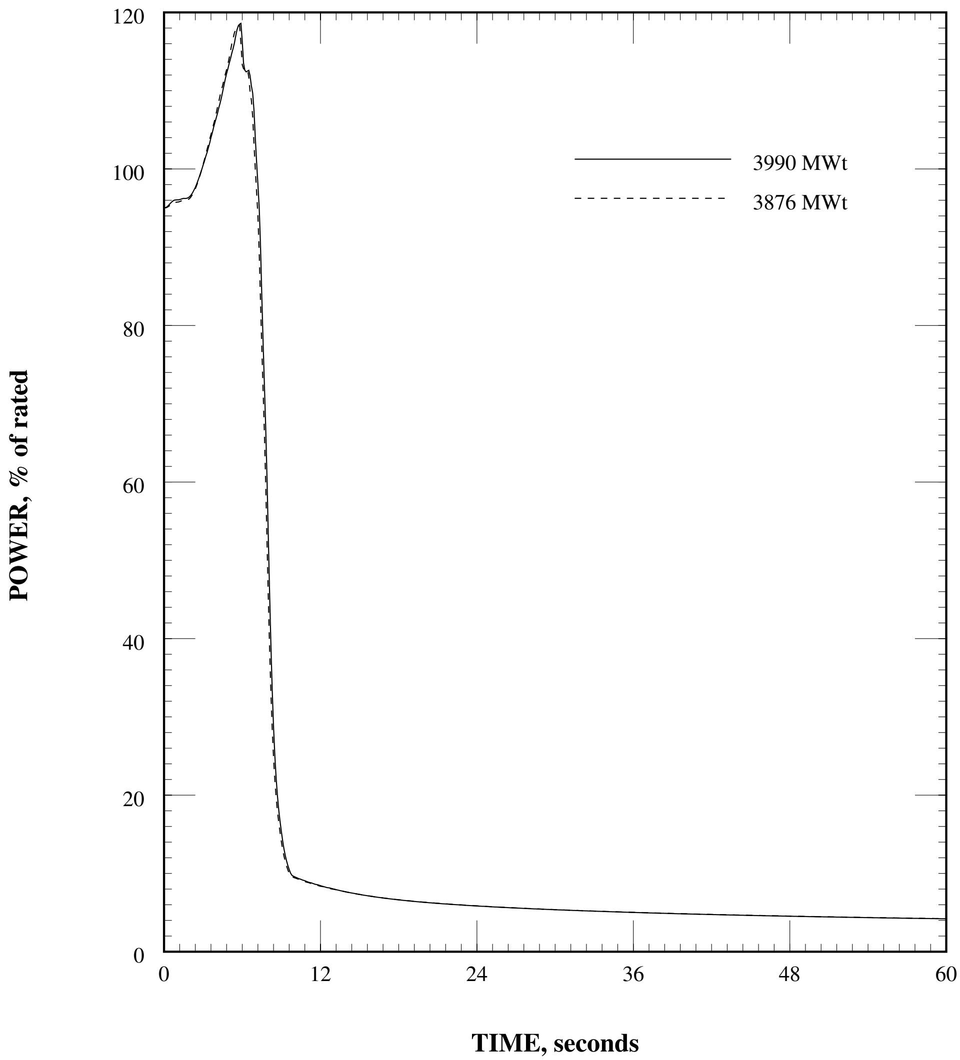


Figure 6.3-63
Limiting Pre-Trip MSLB - RCS Pressure vs. Time

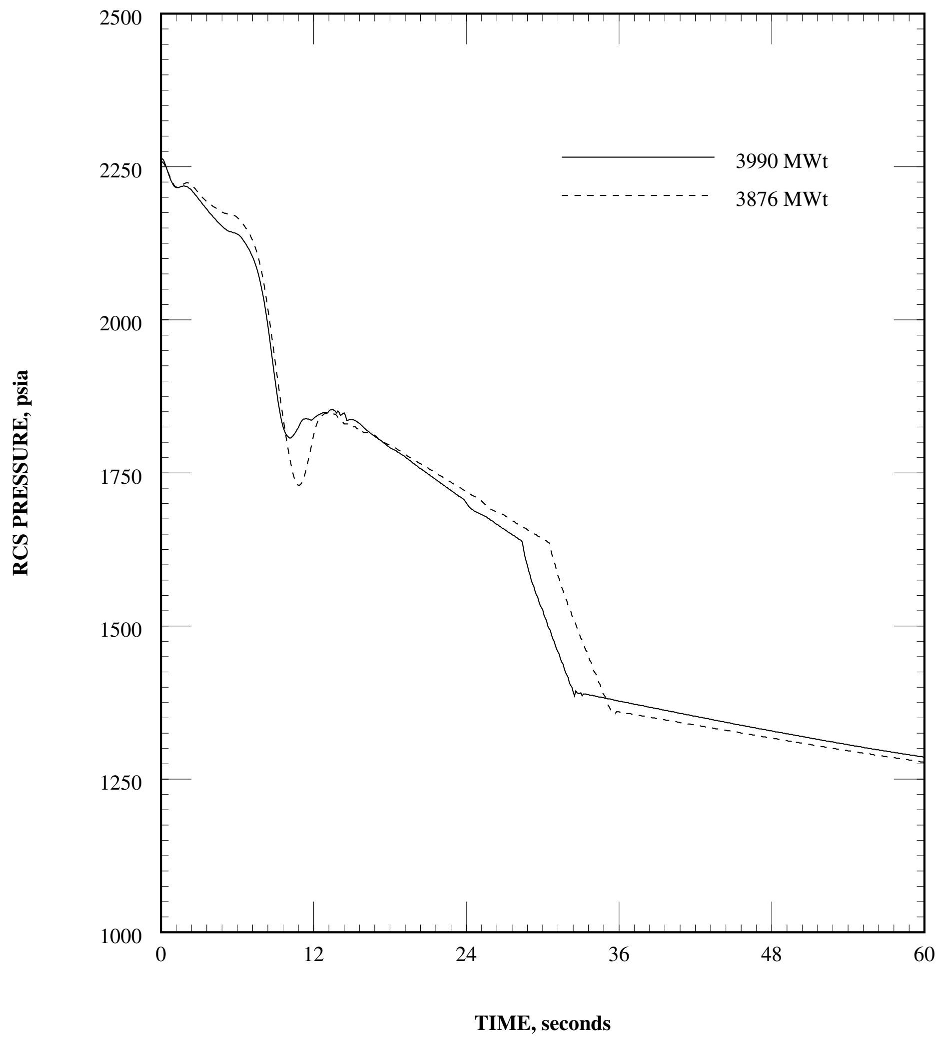


Figure 6.3-64
Limiting Pre-Trip MSLB - SG Pressure vs. Time

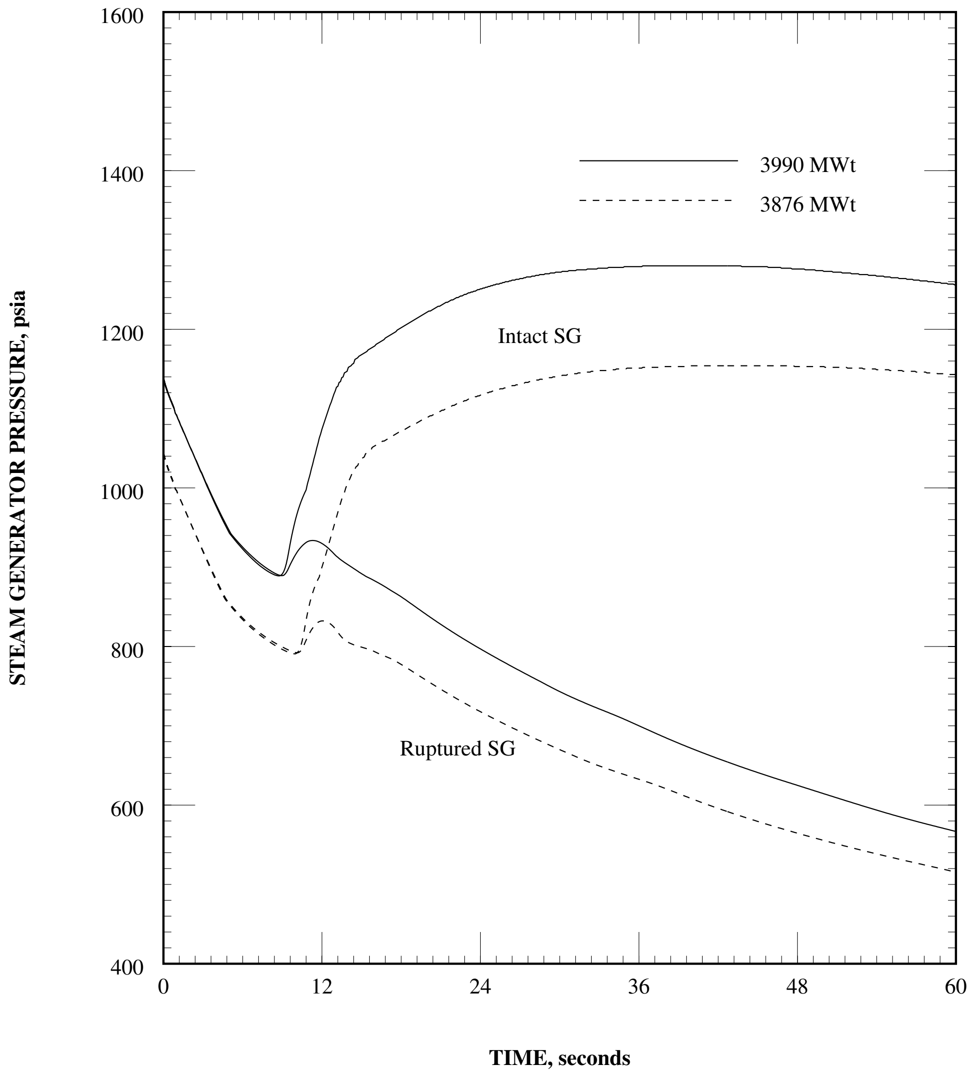


Figure 6.3-65
Limiting Pre-Trip MSLB - SG Steam Flow vs. Time

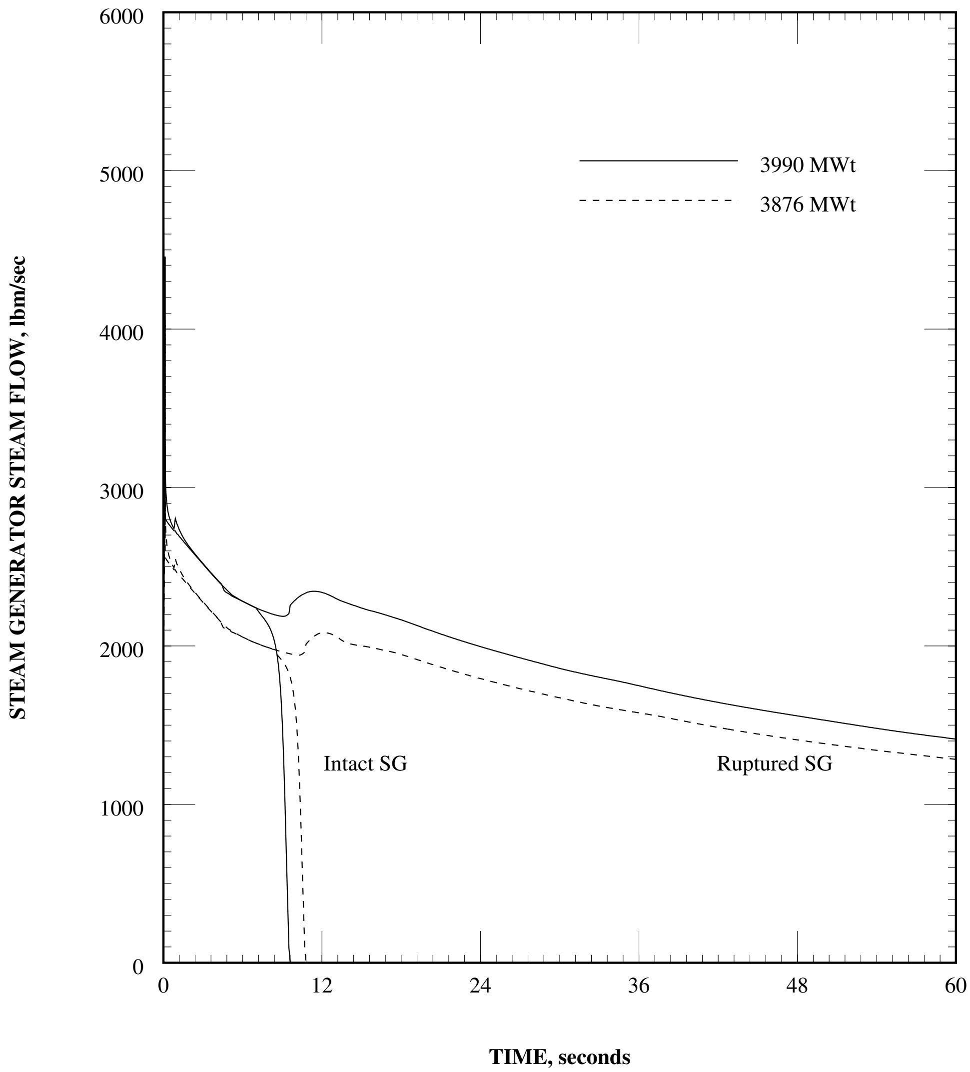
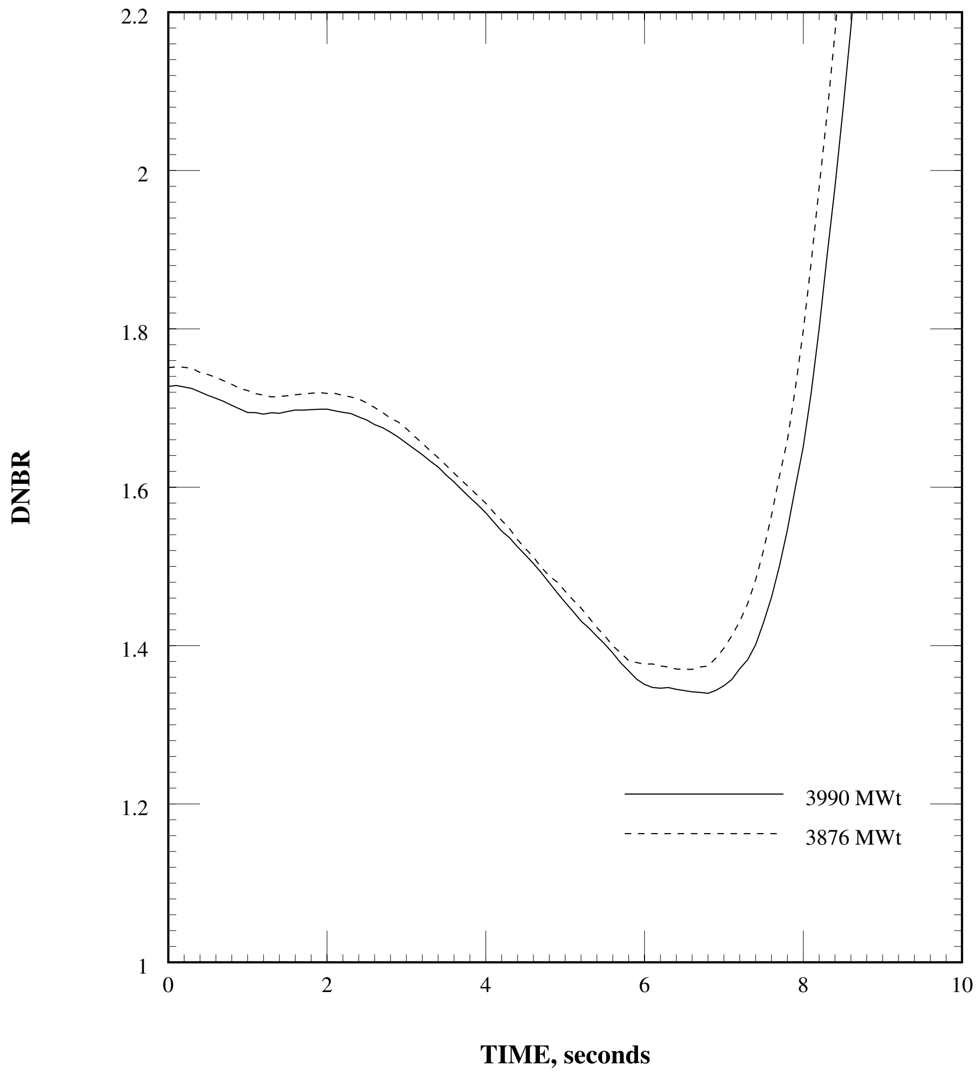


Figure 6.3-66
Limiting Pre-Trip MSLB - Minimum DNBR vs. Time



Section 6.3.2 Decrease in Heat Removal By The Secondary System

Section 6.3.2.1 Loss of External Load

As described in UFSAR Section 15.2.1, the disconnection of the main generator causes the loss of external load event from the electrical distribution grid. A loss of external load generates a turbine trip which results in a reduction in steam flow to the turbine, due to the closure of the turbine stop valves. The SBCS and RPCS would be available to accommodate the load rejection without a reactor trip or opening of MSSVs. Should a turbine trip occur with these systems in the manual mode, a complete termination of main steam flow results and reactor trip would occur on high pressurizer pressure. If no credit is taken for operator action, the MSSVs will open to limit the secondary pressure and provide a heat sink for the NSSS. An operator can initiate a controlled system cooldown using the SBCS and/or ADVs after the reactor trip.

The loss of external load event is classified as an AOO. This event was reviewed for the impacts of PUR and remains bounded by the LOCV event consistent with the analysis in UFSAR Section 15.2.1. The LOCV event analysis is provided in Section 6.3.2.3.

Section 6.3.2.2 Turbine Trip

As described in UFSAR Section 15.2.2, a turbine trip may result from a number of conditions that cause the Turbine Generator Control System (TGCS) to initiate a turbine trip signal. A turbine trip initiates the closure of the turbine stop valves, and results in a reduction in steam flow to the turbine. The SBCS and RPCS are both normally in the automatic mode and would upon turbine trip accommodate the load rejection without necessitating reactor trip or the opening of MSSVs. Should a turbine trip occur with these systems in the manual mode, a complete termination of main steam flow and a reactor trip would occur on high pressurizer pressure. If no credit is taken for operator action, the MSSVs will open to limit the secondary pressure increase and provide a heat sink for the NSSS. The operator can initiate a controlled system cooldown using the SBCS and/or ADVs after the reactor trip.

The turbine trip event is classified as an AOO. This event was reviewed for the impacts of PUR and remains bounded by the LOCV event consistent with the analysis in UFSAR Section 15.2.2. The LOCV event analysis is provided in Section 6.3.2.3.

Section 6.3.2.3 Loss of Condenser Vacuum

Section 6.3.2.3.1 Identification of Event and Causes

As described in UFSAR Section 15.2.3, a LOCV may occur due to the failure of the circulating water system, failure of the main condenser evacuation system, or excessive in-leakage of air through a turbine gland. The turbine is assumed to trip immediately coincident with the LOCV.

The turbine trip that occurs due to the LOCV closes the turbine stop valves. The LOCV will cause the main FW pumps to trip on high backpressure. As no credit is taken for the SBCS, closure of the turbine stop valves and coastdown of the main FW pumps causes the primary and secondary temperatures and pressures to increase rapidly and a reactor trip will occur on high pressurizer pressure. Lifting of the PSVs and MSSVs limit the pressure increase in the primary and secondary systems.

A reactor trip on low SG level could occur immediately following a LOCV when a SG pressure spike causes the steam bubbles in the SG to collapse. This level trip was not credited in the analysis.

The PLCS and Pressurizer Pressure Control System (PPCS) may reduce over-pressurization of the RCS. These systems are assumed to be in manual mode and credit was not taken for their functioning.

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

Consideration of single failures is addressed in Section 6.3.2.3.3.1.

Section 6.3.2.3.2 Acceptance Criteria

LOCV is the most limiting moderate frequency event that results in an unplanned decrease in secondary system heat removal. As defined in the SRP Section 15.2.3, the specific acceptance criteria are:

a. Pressure in the RCS and main steam system should be maintained below 110% of the design.
b. Fuel cladding integrity should be maintained by ensuring that Acceptance Criterion 1 of SRP Section 4.4 is satisfied. Demonstrating that the minimum DNBR is larger than the DNBR SAFDL during the transient ensures this.
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Section 6.3.2.3.3 Description of Analysis

The NSSS response to a LOCV was simulated using the CENTS code. The initial and transient DNBR was calculated using the CETOP-D code which uses the CE-1 CHF correlation.

Two cases were analyzed for a LOCV event:

1. the maximum RCS pressure case; and
2. the maximum secondary system pressure case.

In the first case, the input parameters and initial conditions were selected to maximize the RCS pressure and to demonstrate that the peak RCS pressure remains within 110% of the design pressure. In the second case, parameters were selected to maximize the secondary system pressurization and to demonstrate that the peak secondary system pressure remains within 110% of the design pressure.

Both the existing configuration and PUR were analyzed in order to compare the NSSS response to the LOCV event.

<u>Section 6.3.2.3.3.1</u> <u>Transient Simulation</u>

The system was initialized at 102% power, using the most limiting initial parameters selected for the criteria being challenged. At time equal to zero, the LOCV was simulated by a turbine trip, TAV closure (valve closes in 0.2 seconds), and main FW ramp to zero (in 0.1 seconds). The mismatch of the core power production and secondary system heat removal results in pressurization of the secondary side, and a resulting heatup and pressurization of the primary side. Reactor trip occurs on High Pressurizer Pressure Trip (HPPT). Although an earlier Low Steam Generator Level Trip (LSGLT) may occur due to sudden collapse of bubbles, no credit was taken for LSGLT. The lifting of PSVs and MSSVs at their most adverse lift setpoints followed the HPPT, and both primary and secondary cooldown was provided by the heat removal through the PSVs and MSSVs.

Although it does not affect the RCS and secondary system peak pressures and DNBR, an AFAS and a SIAS occur as the plant begins to cooldown and depressurize. AFW flow was initiated after a time delay that accounts for start of the pumps and the flow delivery to the SGs, and SI flow was initiated by the SIAS actuation.

Active single failures were also considered in the analysis. For the peak pressure criteria, the only mechanisms for mitigation of the RCS and secondary system pressurization are the PSVs, the RCS flow, and the MSSVs. There are no credible failures that can degrade PSVs or MSSVs. A decrease in RCS-to-SG heat transfer due to RCS flow coastdown can be caused by a LOP following turbine trip. RCP coastdown results in a reactor trip generated by the CPC. Due to the rapid reactor trip, this failure reduces the peak pressure relative to the LOCV itself. The results of the parametric study show that LOP with the HPPT does not make the primary and secondary peak pressures more adverse. Therefore, it was concluded that there is no single failure to make maximum primary and secondary peak pressure worse.

With regard to fuel performance, decreased RCS flow is the only parameter that could significantly affect the minimum DNBR. LOP is the only failure which may affect coolant flow. LOCV causes increasing RCS pressure. This pressure increase compensates for the elevated RCS temperatures and the available thermal margin does not degrade before the LOP. Thus, the overall DNBR degradation experienced during an LOCV with LOP event would be bounded by that of the loss of RCS flow event (Section 6.3.3.1).

Section 6.3.2.3.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-23 and Table 6.3-24 contain the initial conditions used for the peak primary and secondary pressure events, respectively.

The following assumptions were made in these analyses:

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. Only the HPPT was enabled. Although a low SG level trip may occur earlier, no credit was taken for this trip.
3. LOCV was assumed to result in almost immediate TAV closure (valve closes in 0.2 seconds) and main FW trip (ramp to zero flow in 0.1 seconds).
4. There was no operator action for the first 30 minutes of the event.

Table 6.3-23
Parameters Used for LOCV Primary Peak Pressure Case

Parameter	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	554	555
Initial pressurizer pressure (psia)	2100	2100
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	11.4	11.4
Initial SG level (ft)	31.2	32.8
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)	16	10
PSV tolerance	+3%	+3%
MSSV tolerance	+3%	+3%
Single failure	none	none
LOP	no	no

Table 6.3-24
Parameters Used for LOCV Secondary Peak Pressure Case

Parameter	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	562	566
Initial pressurizer pressure (psia)	2100	2100
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	11.4	11.4
Initial SG level (ft)	31.2	32.8
MTC ($\Delta\rho$/°F)	0.0E-04	0.0E-04
FTC	most negative	most 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)	0	0
PSV tolerance	+3%	+3%
MSSV tolerance	+3%	+3%
Single failure	none	none
LOP	no	no

Section 6.3.2.3.5 Results

Table 6.3-25 and Table 6.3-26 present a sequence of events following the LOCV until operator action is initiated for the primary and secondary peak pressure cases, respectively. LOCV analyses were performed separately for primary and secondary system peak pressure limits since the selection of worst parameters for these events are not mutually conservative.

The behaviors of the NSSS parameters following the LOCV are presented in Figure 6.3-67 to Figure 6.3-92.

The sequence of events during the transient indicates 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 and secondary pressures due to the higher initial core power level and RCS temperature.

Primary Peak Pressure Case

Figure 6.3-67 through Figure 6.3-79 show the response for the LOCV event under most adverse initial and transient conditions that maximize the RCS peak pressure for 3990 MW_t and 3876 MW_t units.

The sudden reduction of steam flow caused by the LOCV leads to a reduction of the primary-to-secondary heat transfer and the RCS and secondary system temperature and pressure increases. The rapid heatup of the RCS results in a reactor trip on high pressurizer pressure. The PSVs open maintaining primary pressure below 110% of the design value. The DNBR value increases above the initial value, and remains well above the SAFDL limit throughout the entire transient.

Secondary Peak Pressure Case

Figure 6.3-80 through Figure 6.3-92 show the response for the LOCV event under the most adverse initial and transient conditions that maximize the secondary peak pressure for 3990 MW_t and 3876 MW_t units.

The sudden reduction of steam flow caused by the LOCV leads to a reduction of the primary-to-secondary heat transfer and results in a RCS and secondary system temperature and pressure increase. The rapid pressurization of the SGs results in opening of the first and the second bank of MSSVs. The pressurization of the RCS results in a reactor trip on high pressurizer pressure. The effects of the reactor trip, opening of the PSVs, and opening of the third bank MSSVs maintain secondary pressure below 110% of the design value. The DNBR value increases above the initial value, and remains well above the SAFDL limit throughout the entire transient.

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

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOCV, turbine trip, main FW pump trip	---	---
7.32	7.20	Pressurizer pressure reaches reactor trip setpoint (psia)	2415	2415
7.32	7.20	HPPT signal generated	---	---
7.82	7.70	Reactor trip breakers open	---	---
8.42	8.30	Scram CEAs begin falling	---	---
9.05	8.88	PSVs open (psia)	2549	2549
9.67	9.53	Maximum RCS pressure (psia)	2712	2733
10.1	9.18	MSSV bank 1 open (psia)	1303	1303
11.4	11.4	PSVs close (psia)	2422	2422
12.2	11.3	MSSV bank 2 open (psia)	1344	1344
13.0	12.9	Maximum pressurizer water volume (ft^3)	811	824
14.5	13.4	MSSV bank 3 open (psia)	1370	1370
14.5	13.5	Maximum SG pressure (psia)	1377	1377
18.6	17.8	SG level reaches AFAS setpoint (%WR)	20	20
18.6	17.8	AFAS generated	---	---
19.0	21.3	MSSV bank 3 close (psia)	1301	1301
33.0	31.1	MSSV bank 2 close (psia)	1277	1277
58.0	59.9	MSSV bank 1 close (psia)	1238	1238
>60	>60	AFW flow initiated (gpm)	186	186
1800	1800	Operator initiates the cooldown (min)	30	30

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

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOCV, turbine trip, main FW pump trip	---	---
6.26	4.17	MSSV bank 1 open (psia)	1303	1303
7.26	7.01	Pressurizer pressure reaches reactor trip setpoint (psia)	2415	2415
7.26	7.01	HPPT signal generated	---	---
7.76	7.51	Reactor trip breakers open	---	---
8.03	5.84	MSSV bank 2 open (psia)	1344	1344
8.36	8.11	Scram CEAs begin falling	---	---
9.19	9.62	PSVs open (psia)	2549	2549
9.66	9.95	Maximum RCS pressure (psia)	2657	2631
9.74	7.49	MSSV bank 3 open (psia)	1370	1370
9.78	14.5	Maximum SG pressure	1377	1390
10.9	10.9	PSVs close (psia)	2422	2422
12.1	11.9	Maximum pressurizer water volume (ft^3)	764	719
16.5	12.9	SG level reaches AFAS setpoint (%WR)	20	20
16.5	12.9	AFAS generated	---	---
21.8	23.6	MSSV bank 3 close (psia)	1301	1301
34.8	35.6	MSSV bank 2 close (psia)	1277	1277
>60	58.9	AFW flow initiated (gpm)	186	186
>60	>60	MSSV bank 1 close (psia)	1238	1238
1800	1800	Operator initiates the cooldown (min)	30	30

Section 6.3.2.3.6 Conclusions

For the LOCV event, all acceptance criteria are met. The peak primary and secondary pressures remain below 110% of design at all times thus ensuring the integrity of the RCS or the main steam system. The minimum DNBR remains above the safety analysis limit thus ensuring fuel cladding integrity.

Figure 6.3-67
LOCV Primary Peak Pressure Case - Core Power vs. Time

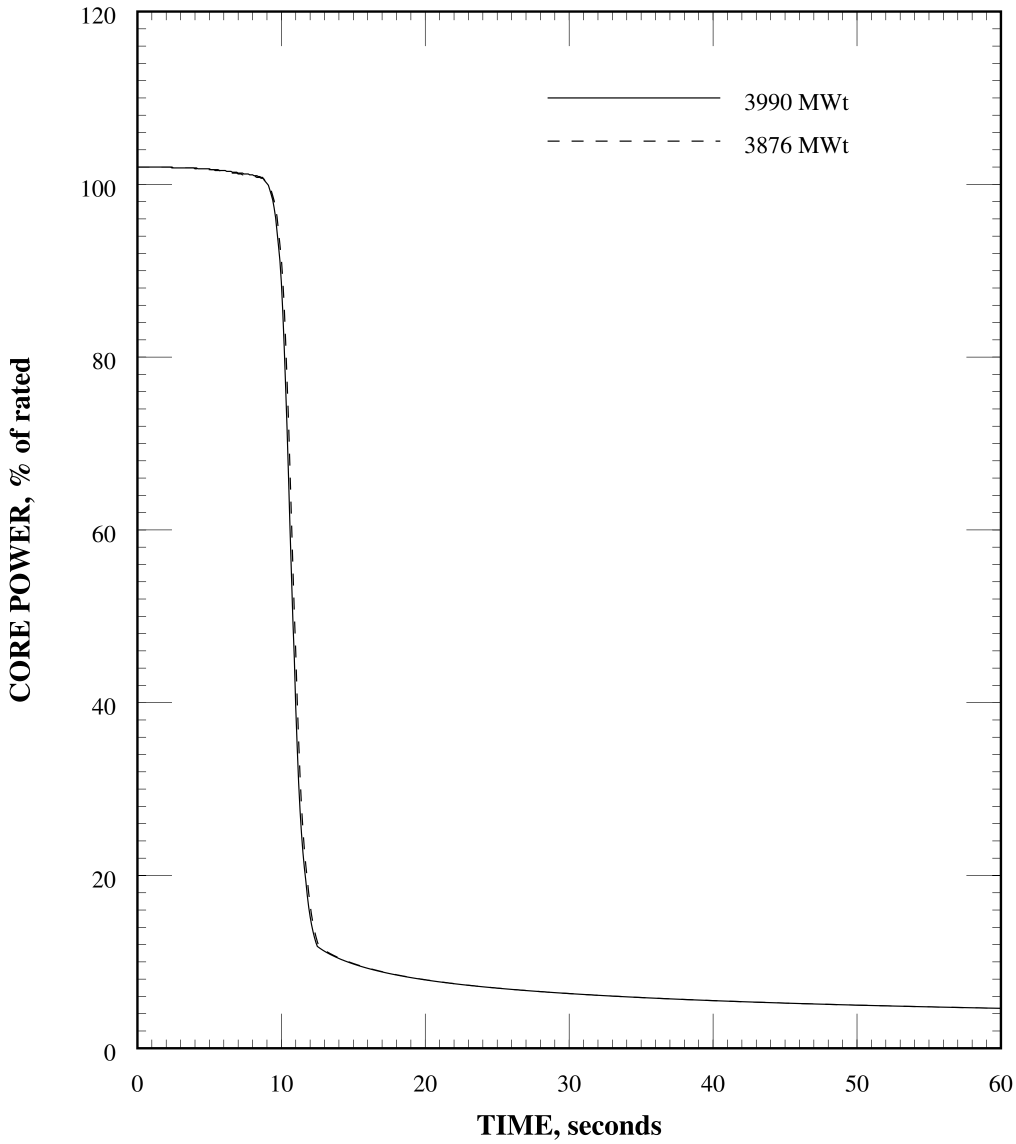


Figure 6.3-68
LOCV Primary Peak Pressure Case - Core Heat Flux vs. Time

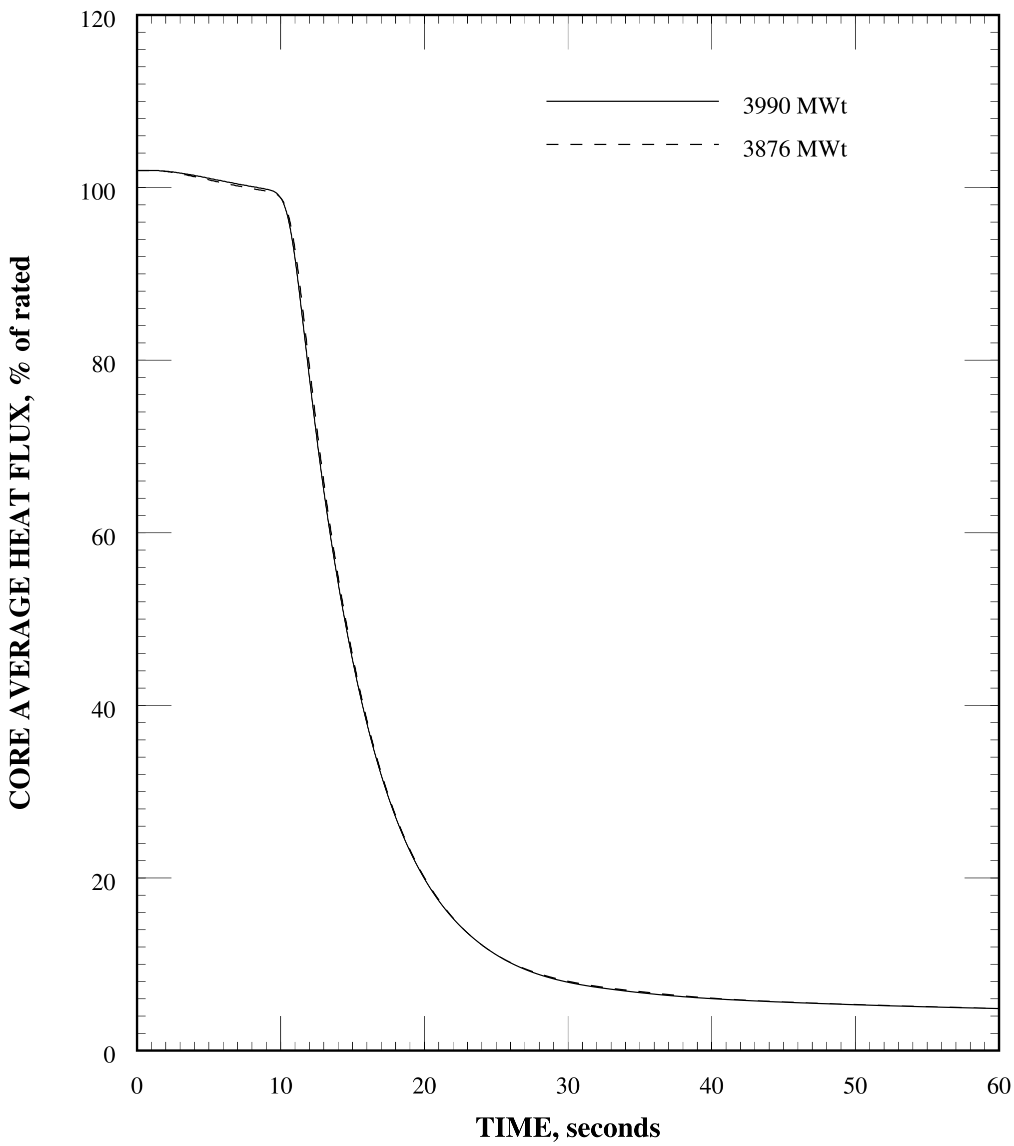


Figure 6.3-69
LOCV Primary Peak Pressure Case - Core Reactivities vs. Time

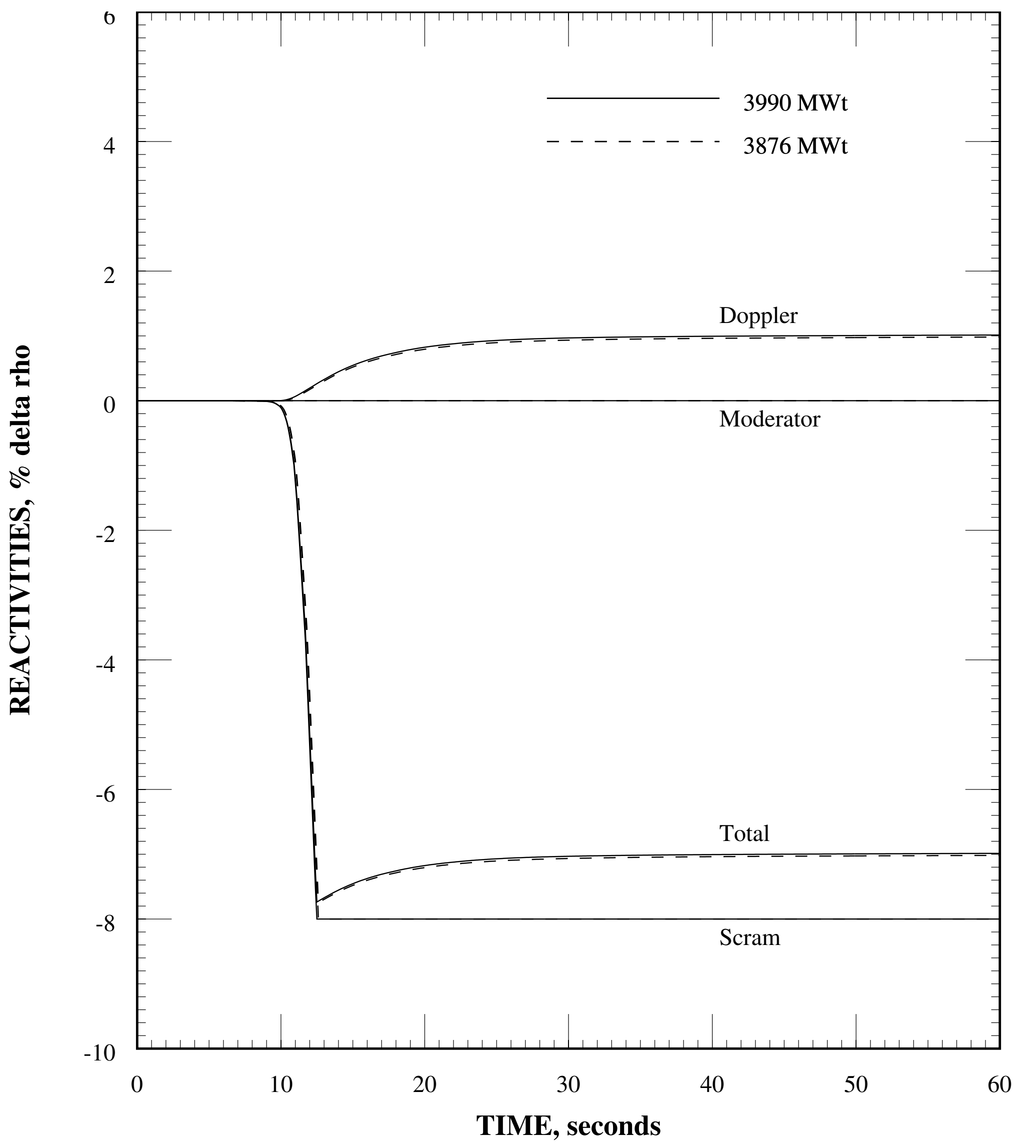


Figure 6.3-70
LOCV Primary Peak Pressure Case - RCS Temperatures vs. Time

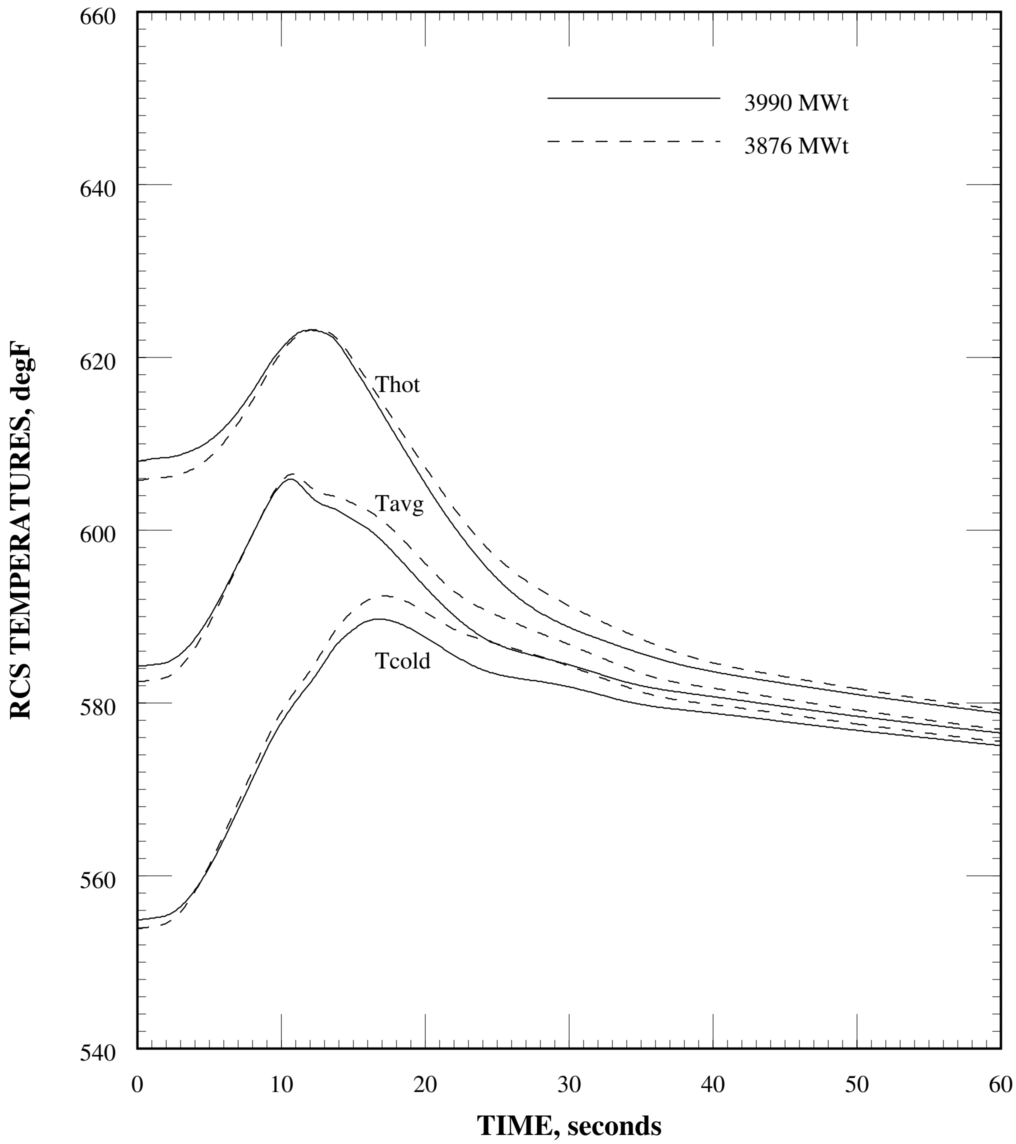


Figure 6.3-71
LOCV Primary Peak Pressure Case - RCS Pressure vs. Time

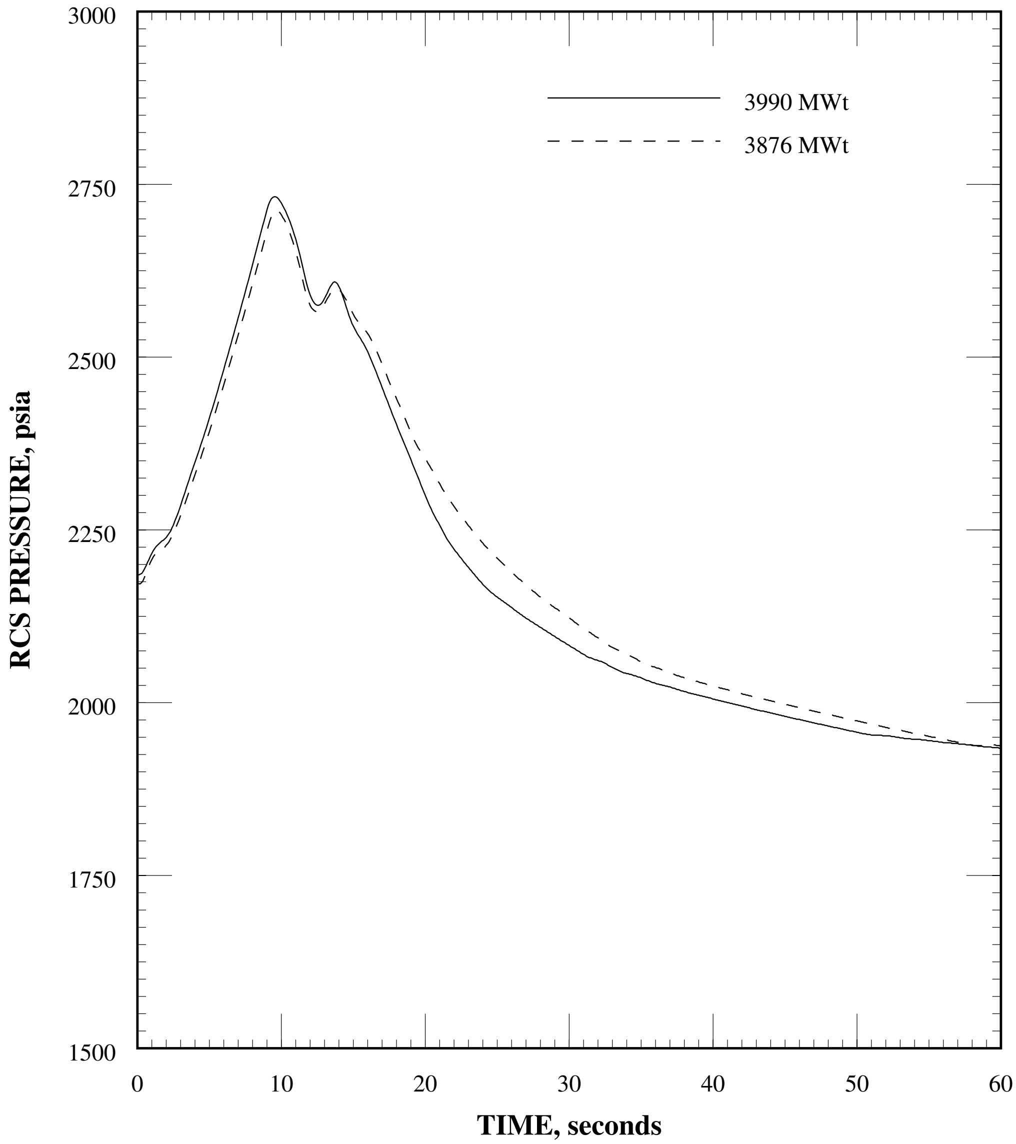


Figure 6.3-72
LOCV Primary Peak Pressure Case - Pressurizer Pressure vs. Time

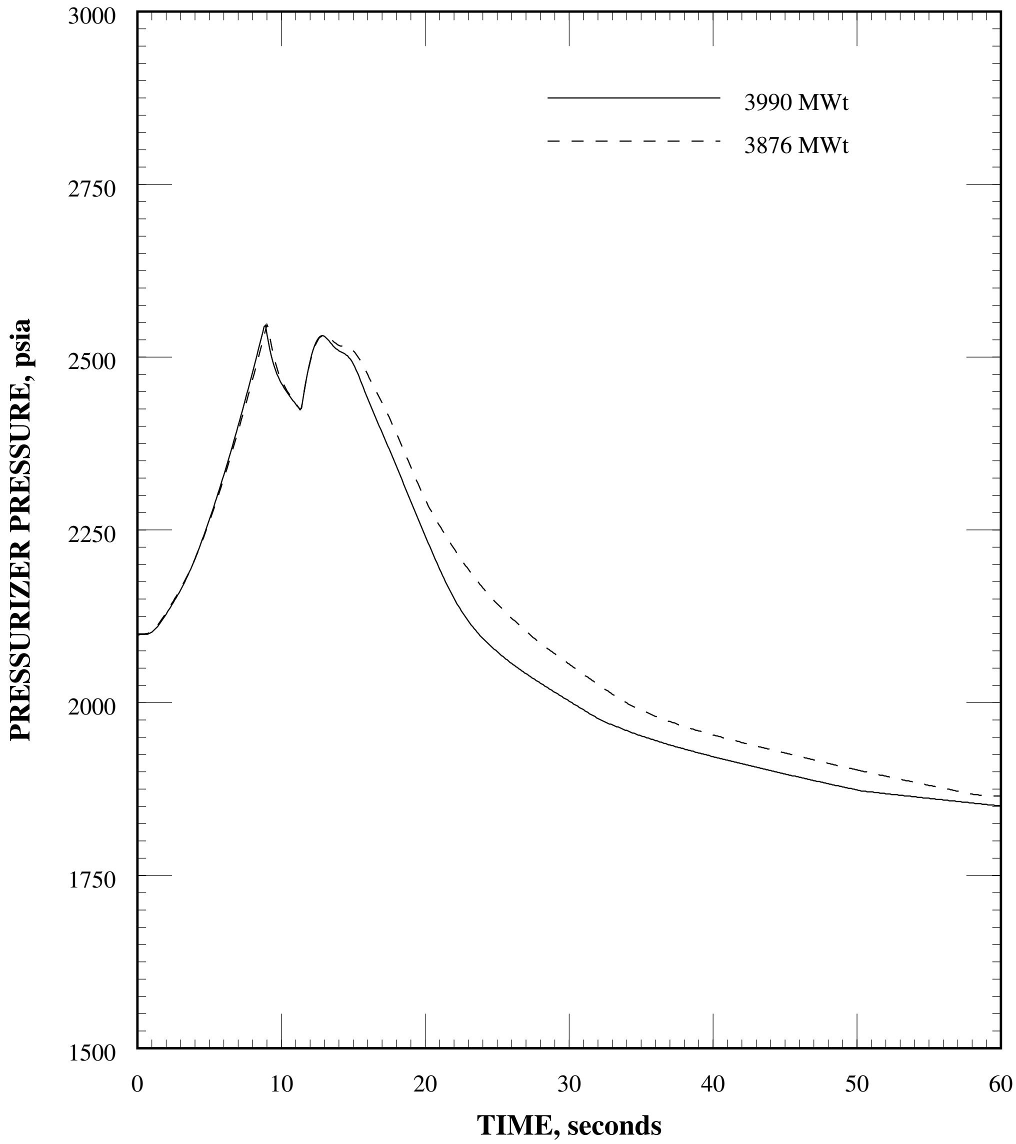


Figure 6.3-73
LOCV Primary Peak Pressure Case - Pressurizer Water Volume vs. Time

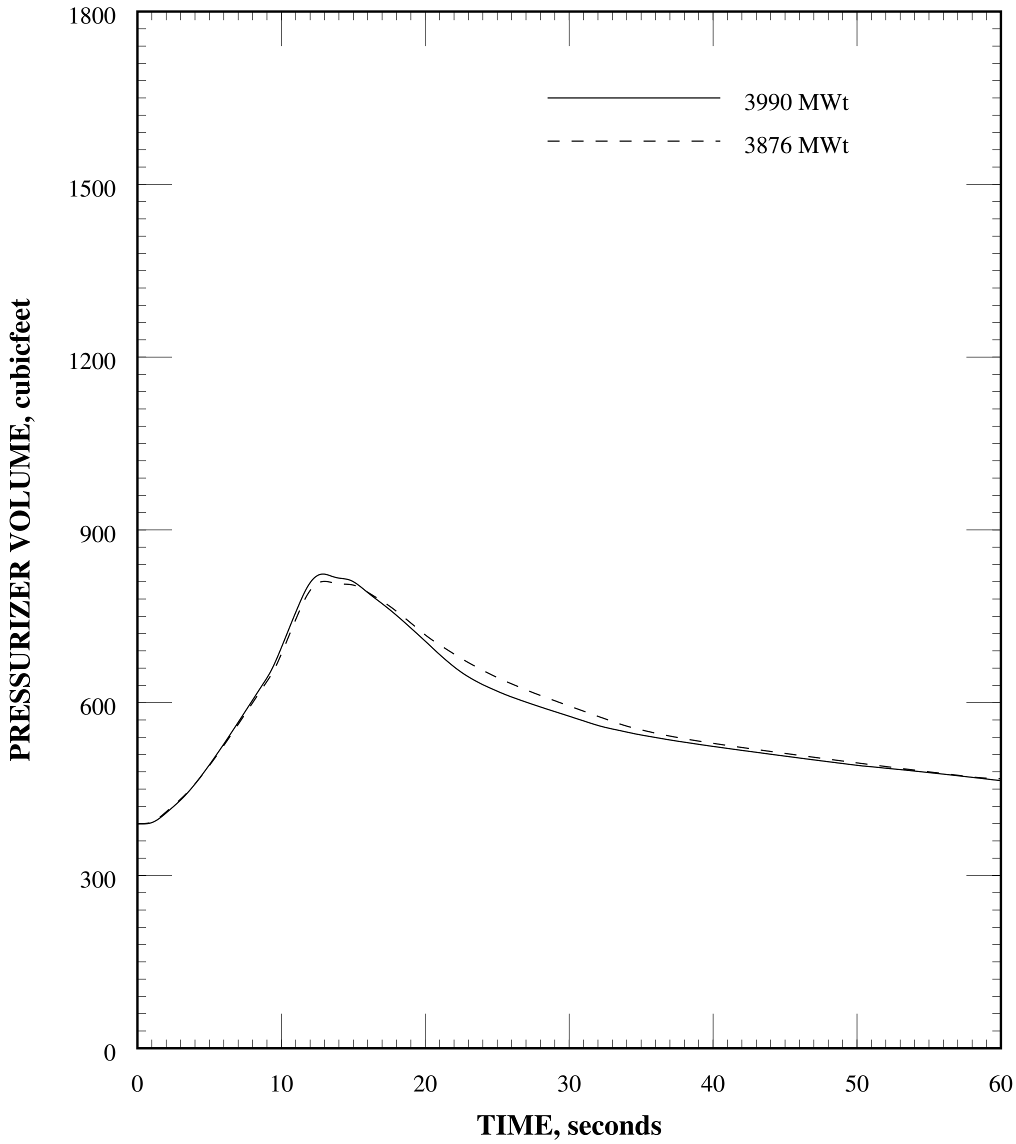


Figure 6.3-74
LOCV Primary Peak Pressure Case - SG Pressure vs. Time

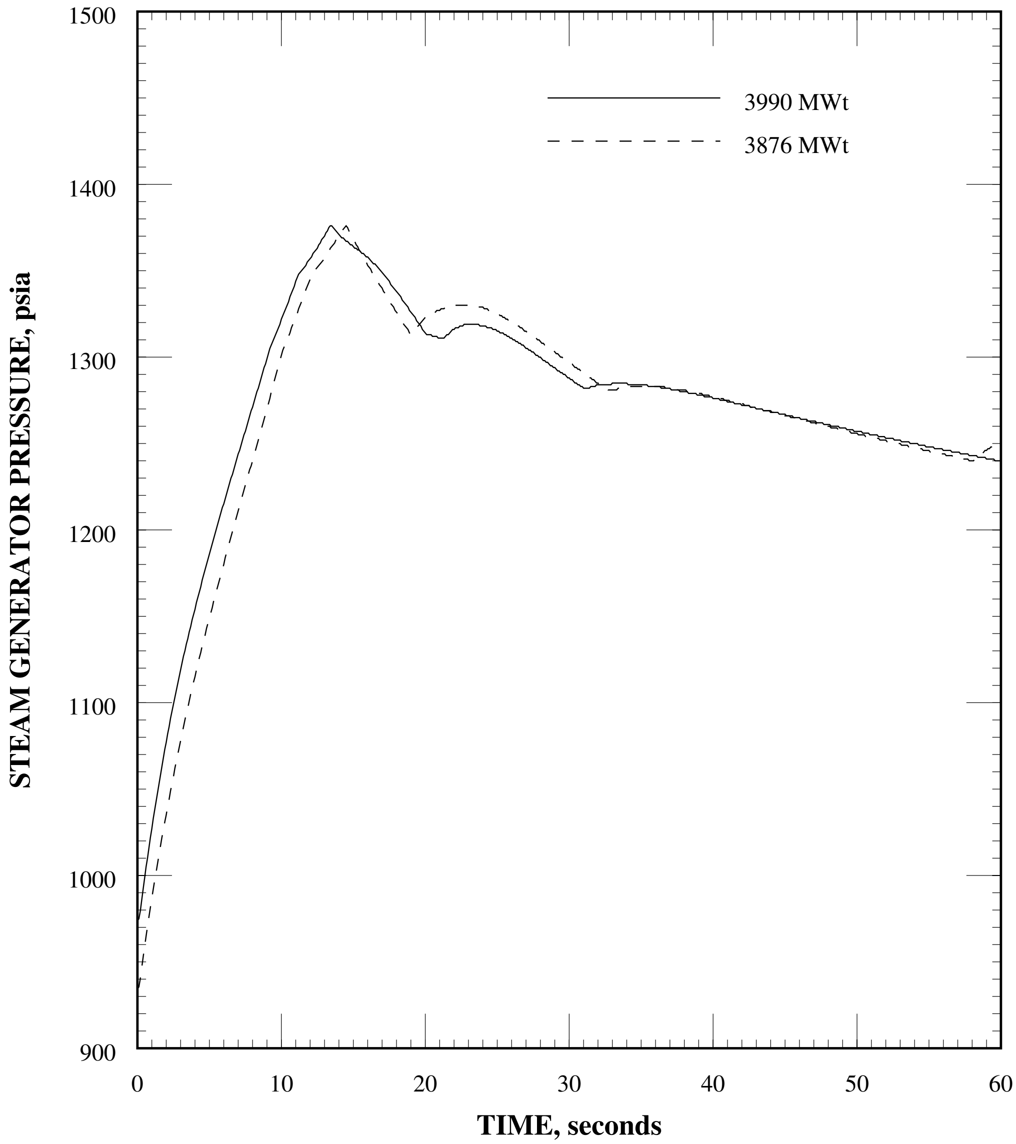


Figure 6.3-75
LOCV Primary Peak Pressure Case - SG Level vs. Time

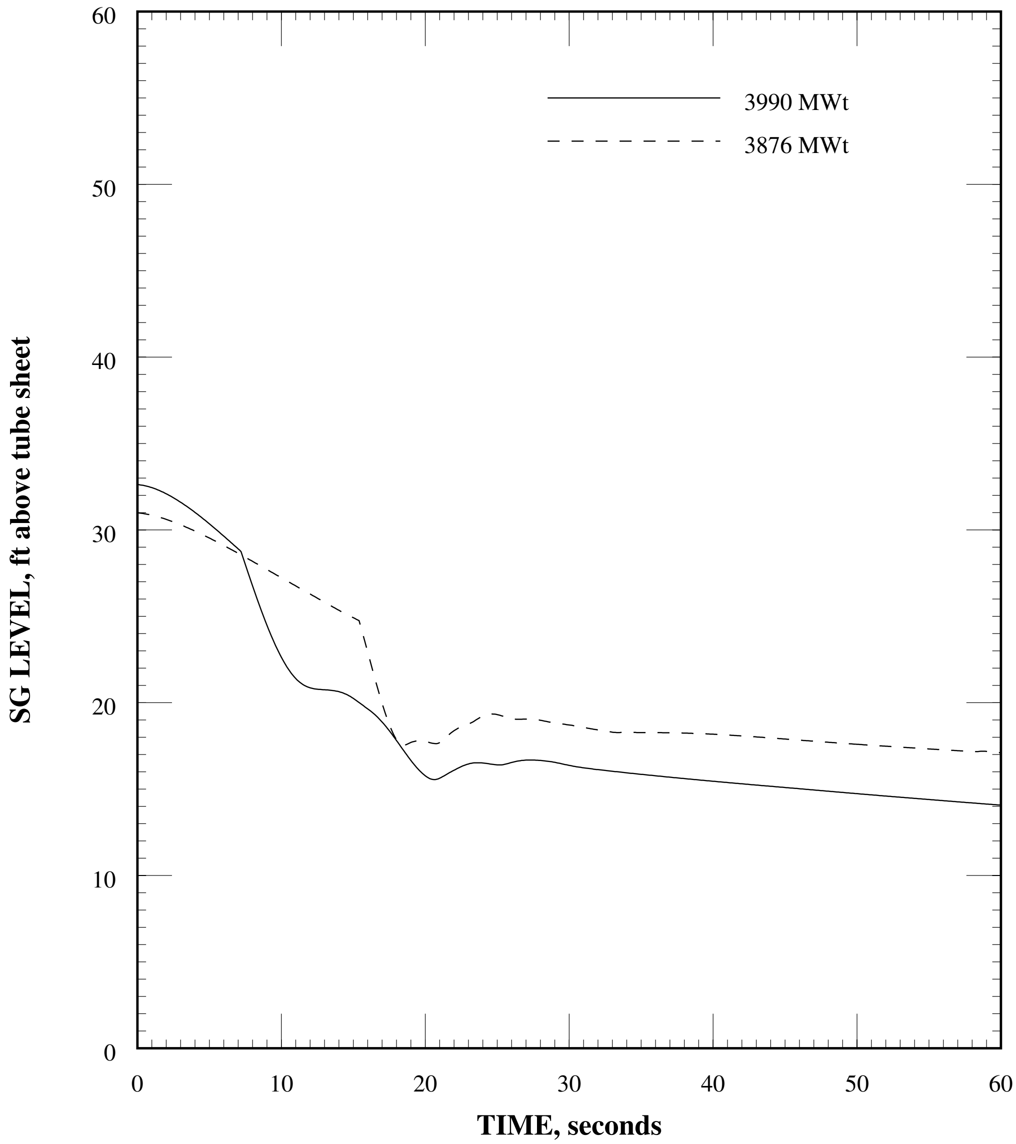


Figure 6.3-76
LOCV Primary Peak Pressure Case - SG Liquid Inventory vs. Time

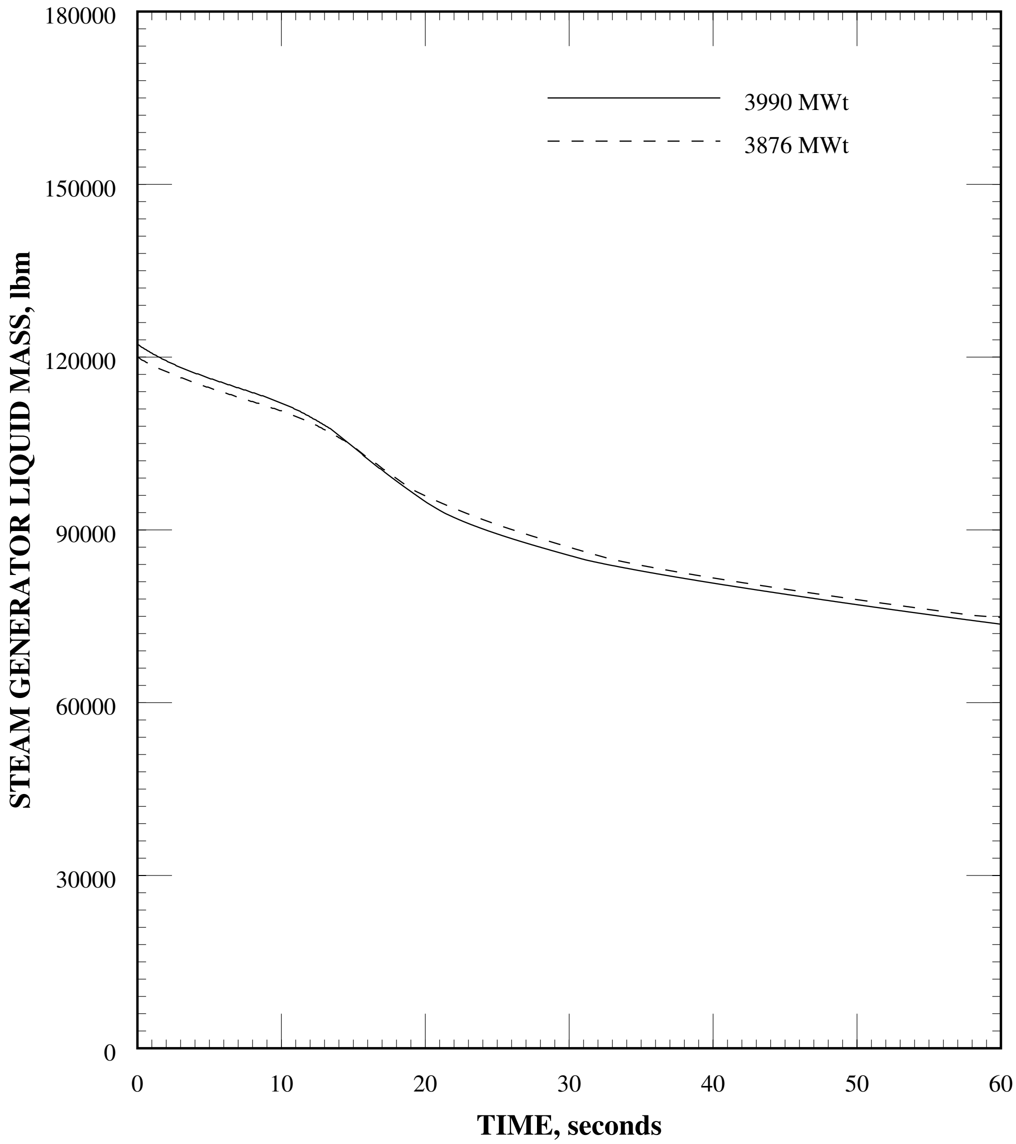


Figure 6.3-77
LOCV Primary Peak Pressure Case - Integrated Steam Flow vs. Time

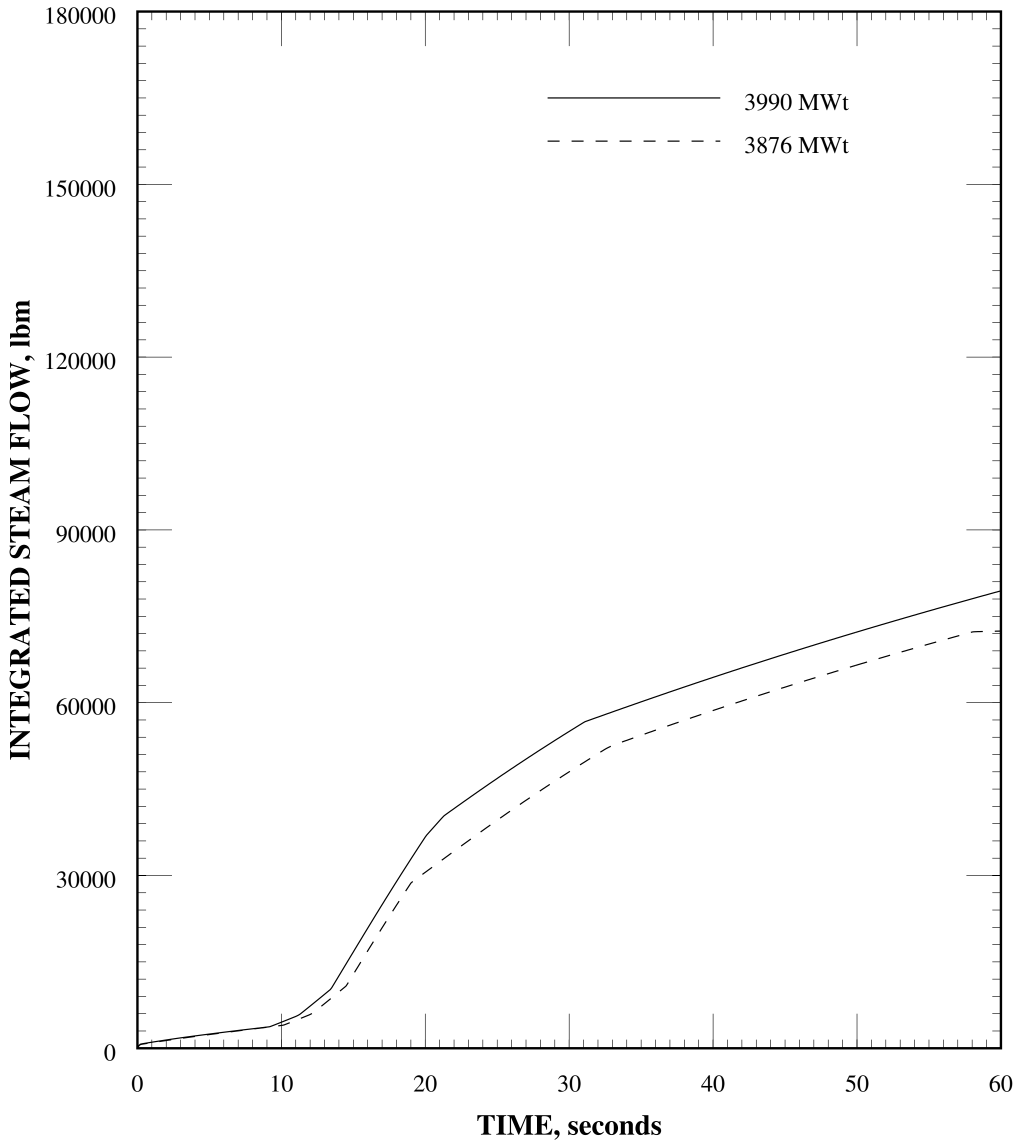


Figure 6.3-78
LOCV Primary Peak Pressure Case - Total FW Flow vs. Time

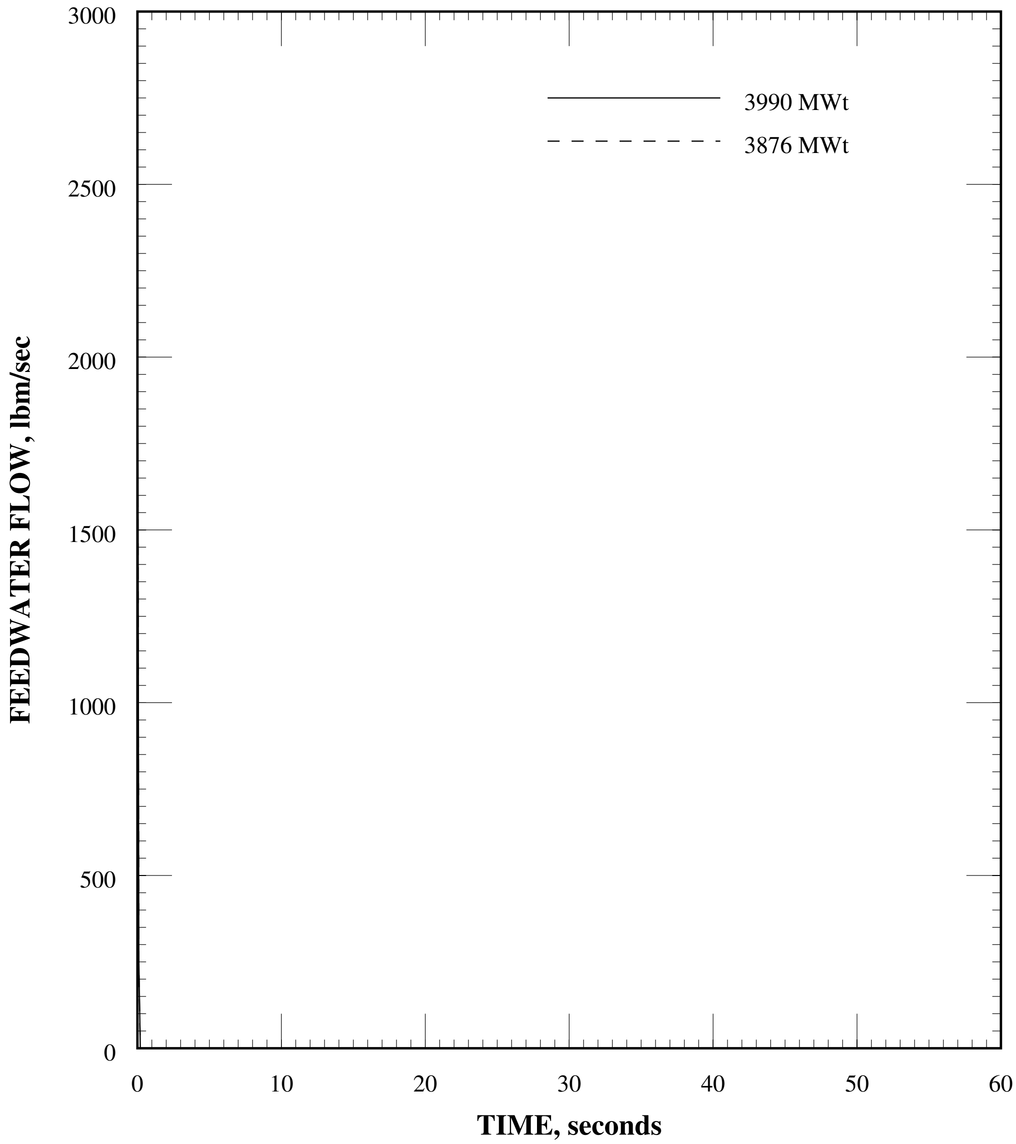


Figure 6.3-79
LOCV Primary Peak Pressure Case - Minimum DNBR vs. Time

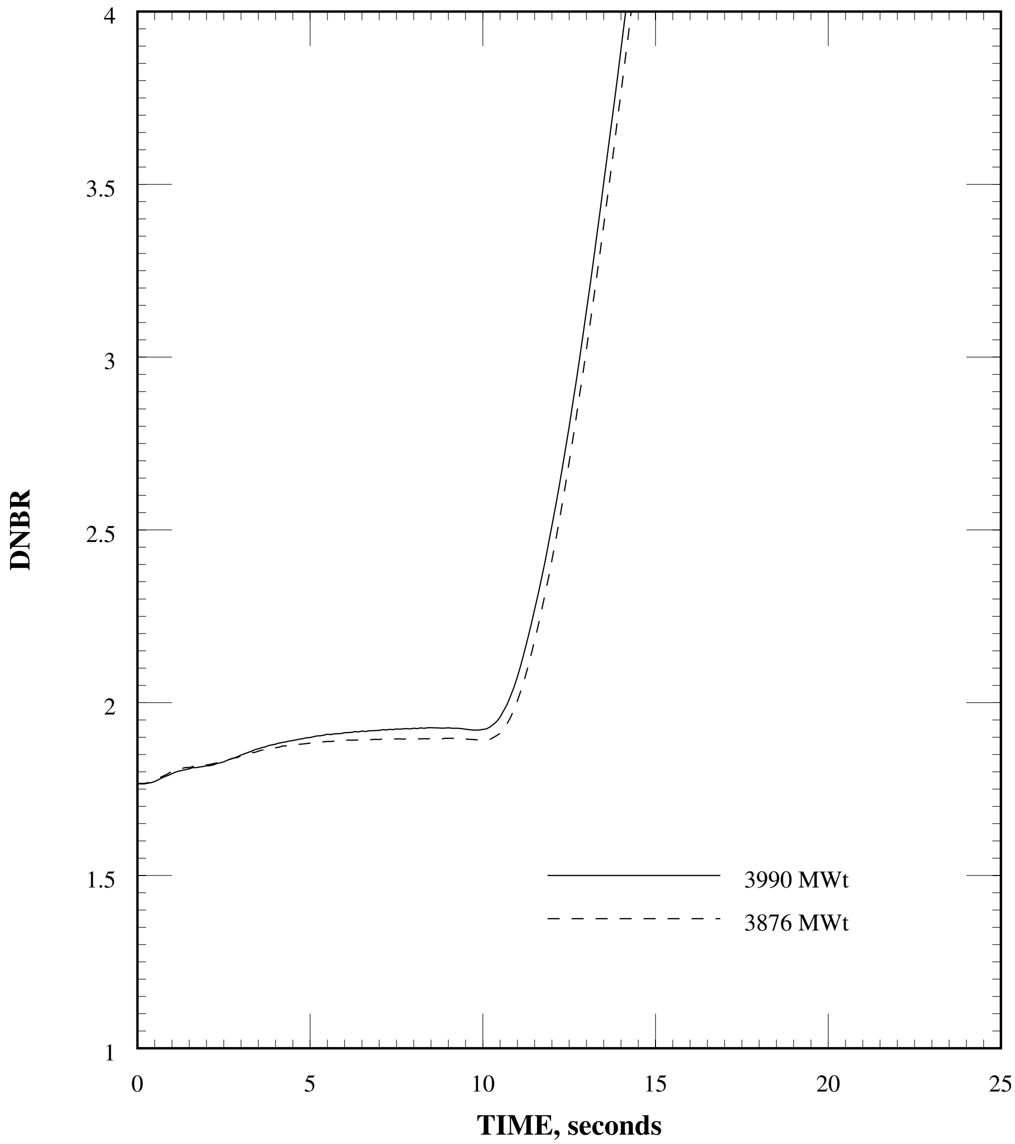


Figure 6.3-80
LOCV Secondary Peak Pressure Case - Core Power vs. Time

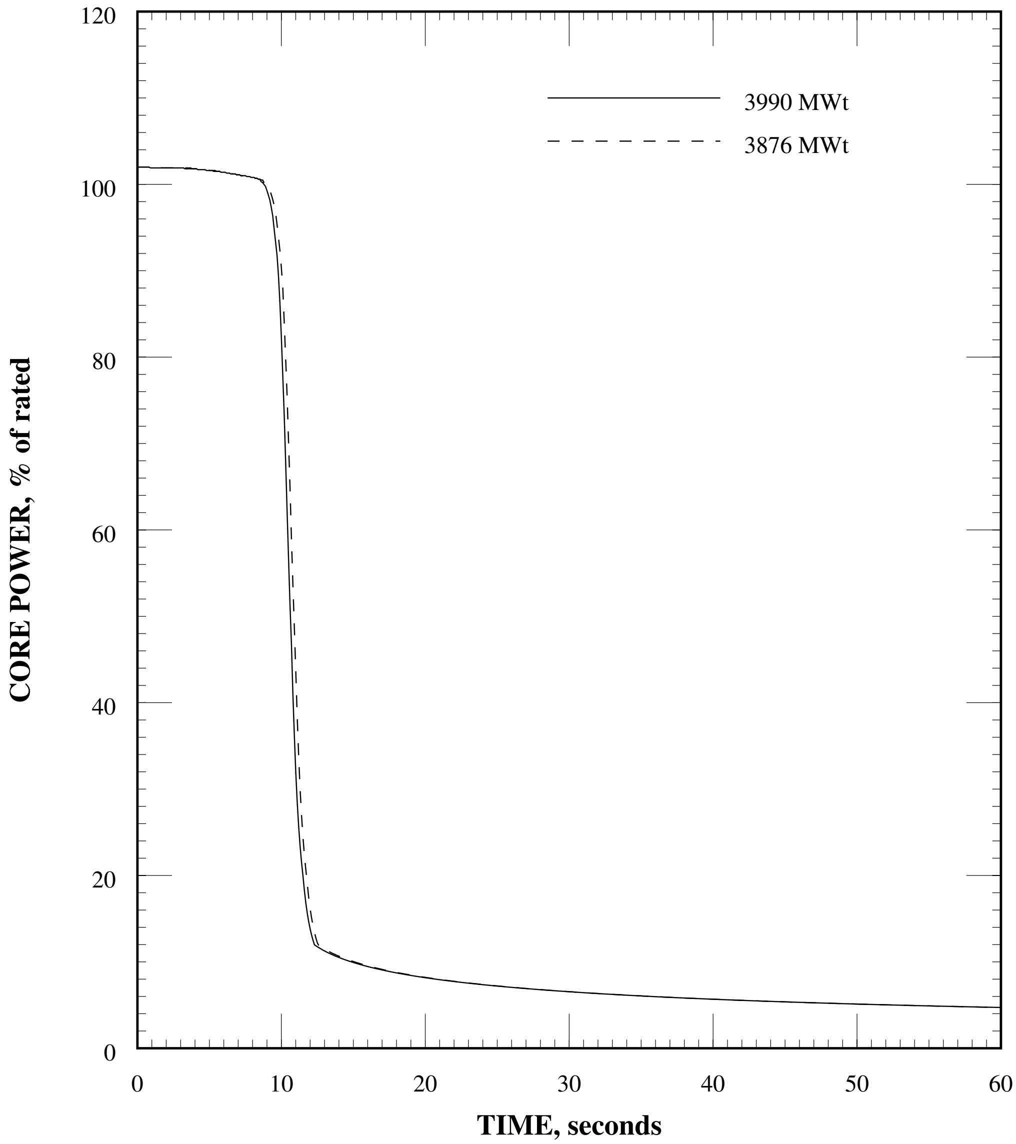


Figure 6.3-81
LOCV Secondary Peak Pressure Case - Core Heat Flux vs. Time

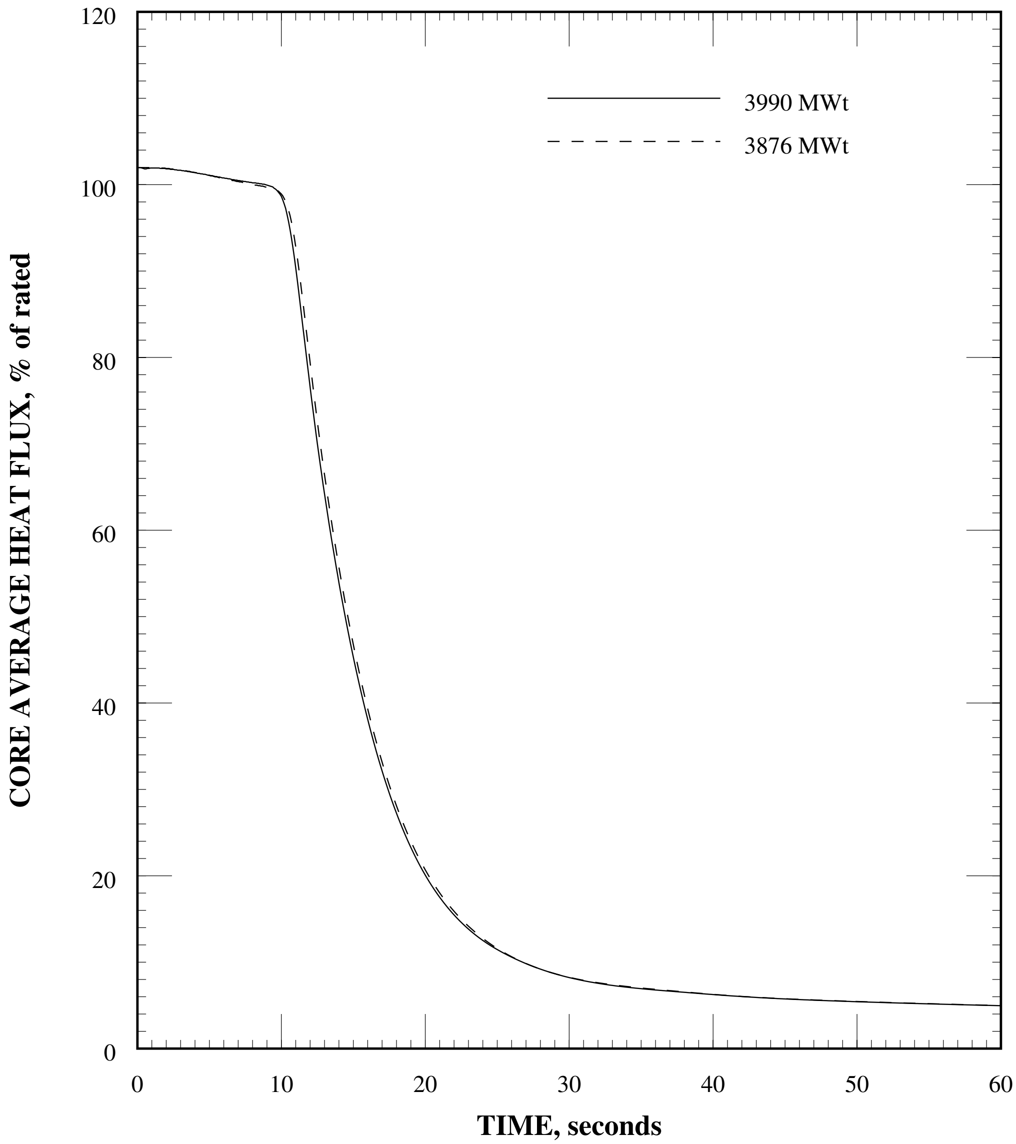


Figure 6.3-82
LOCV Secondary Peak Pressure Case - Core Reactivities vs. Time

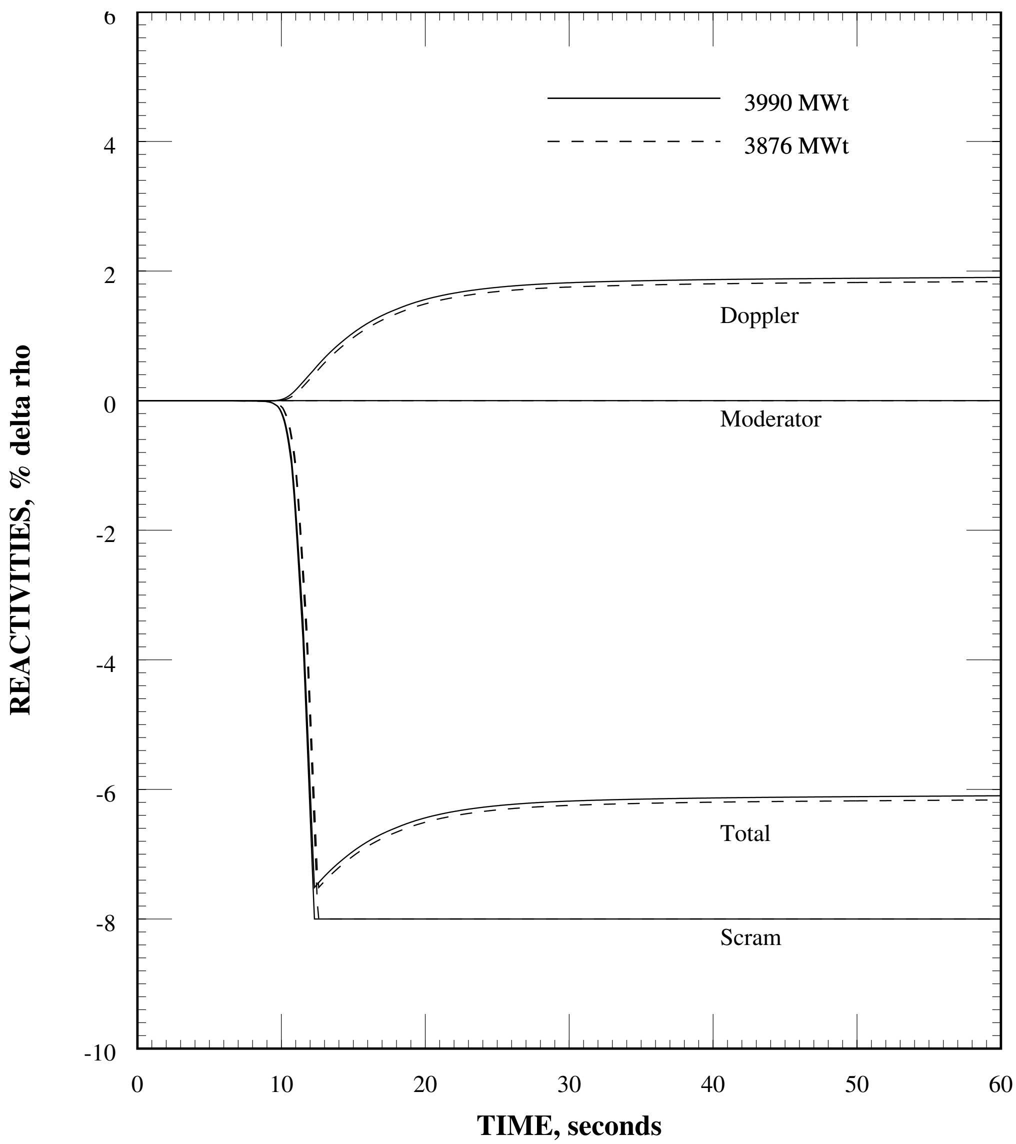


Figure 6.3-83
LOCV Secondary Peak Pressure Case - RCS Temperatures vs. Time

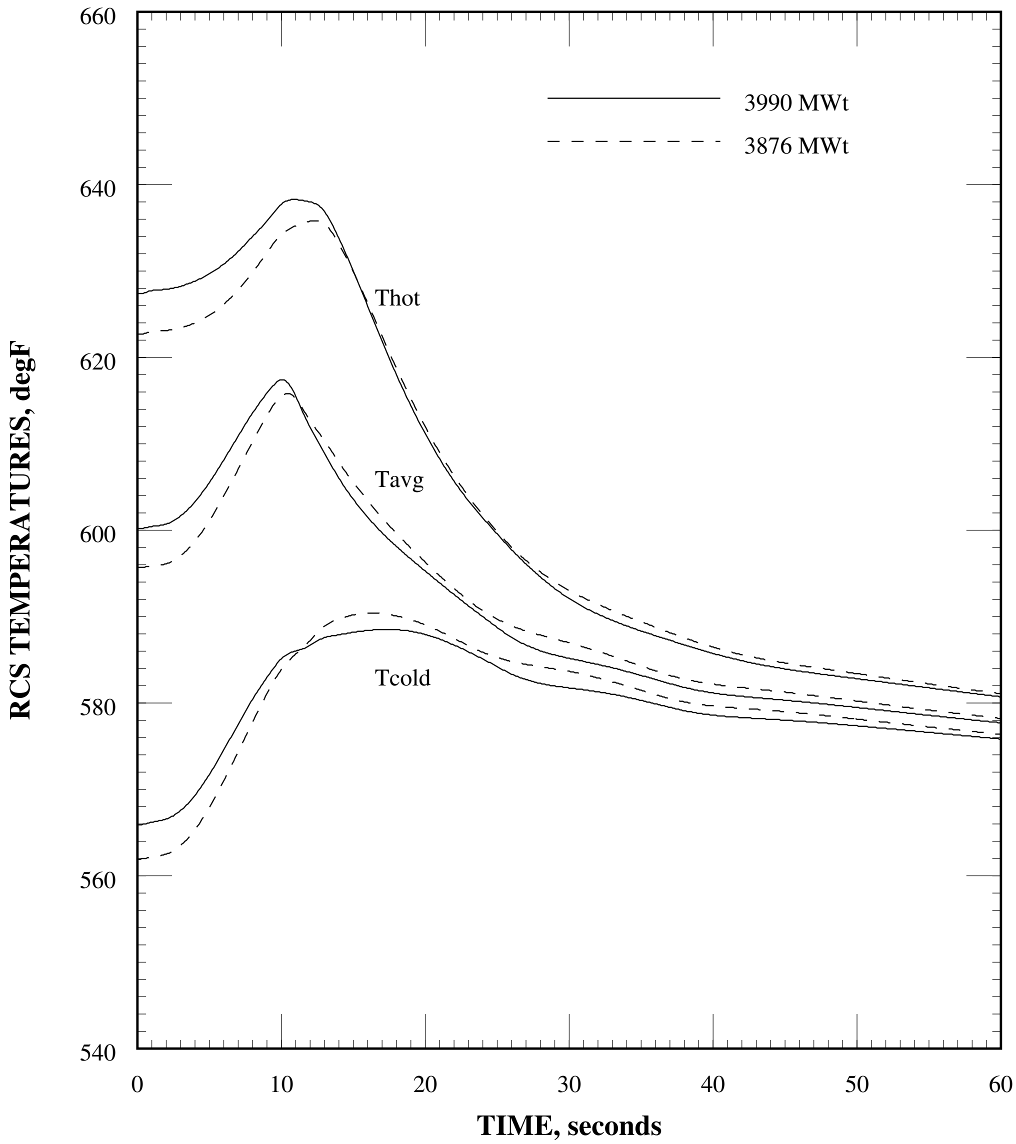


Figure 6.3-84
LOCV Secondary Peak Pressure Case - RCS Pressure vs. Time

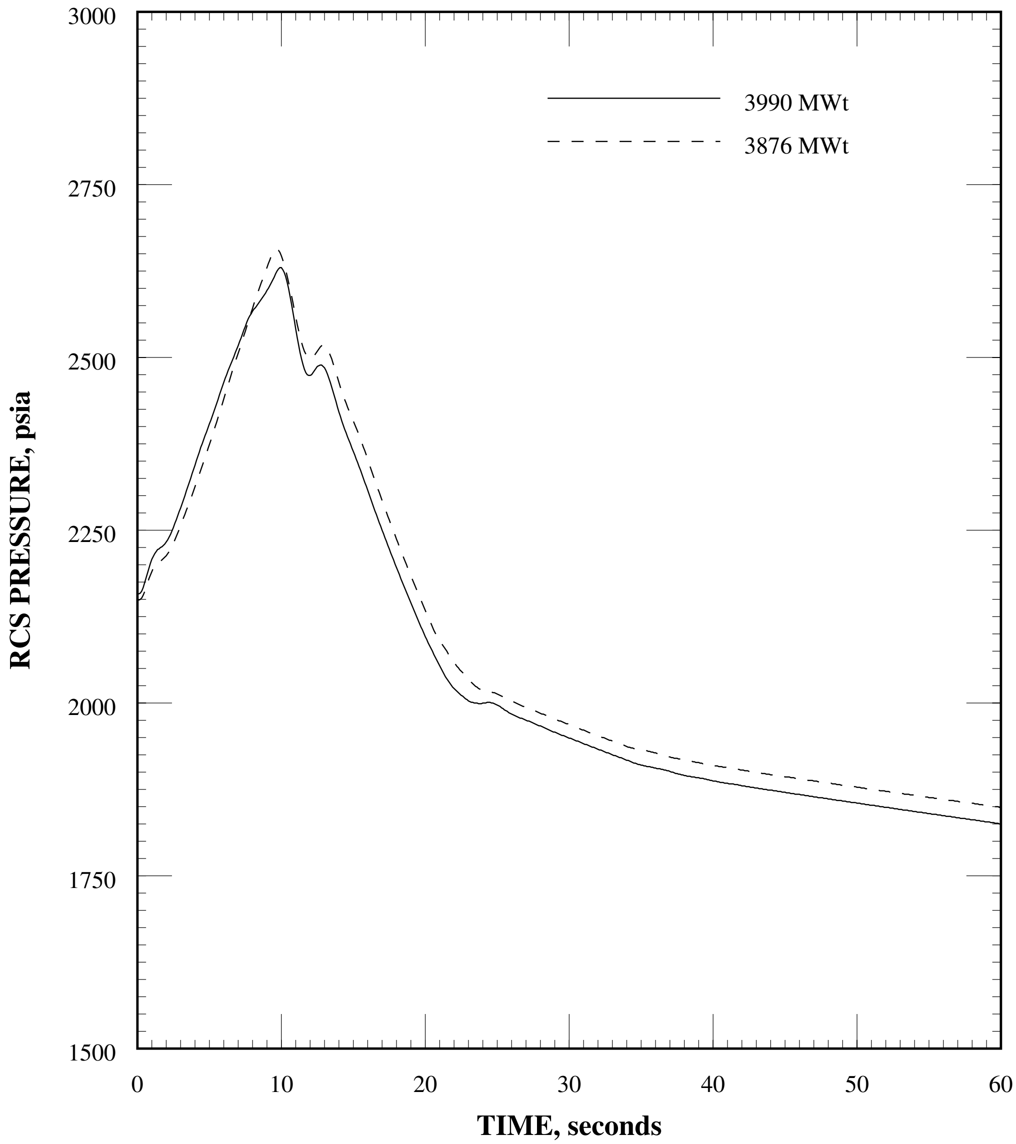


Figure 6.3-85
LOCV Secondary Peak Pressure Case - Pressurizer Pressure vs. Time

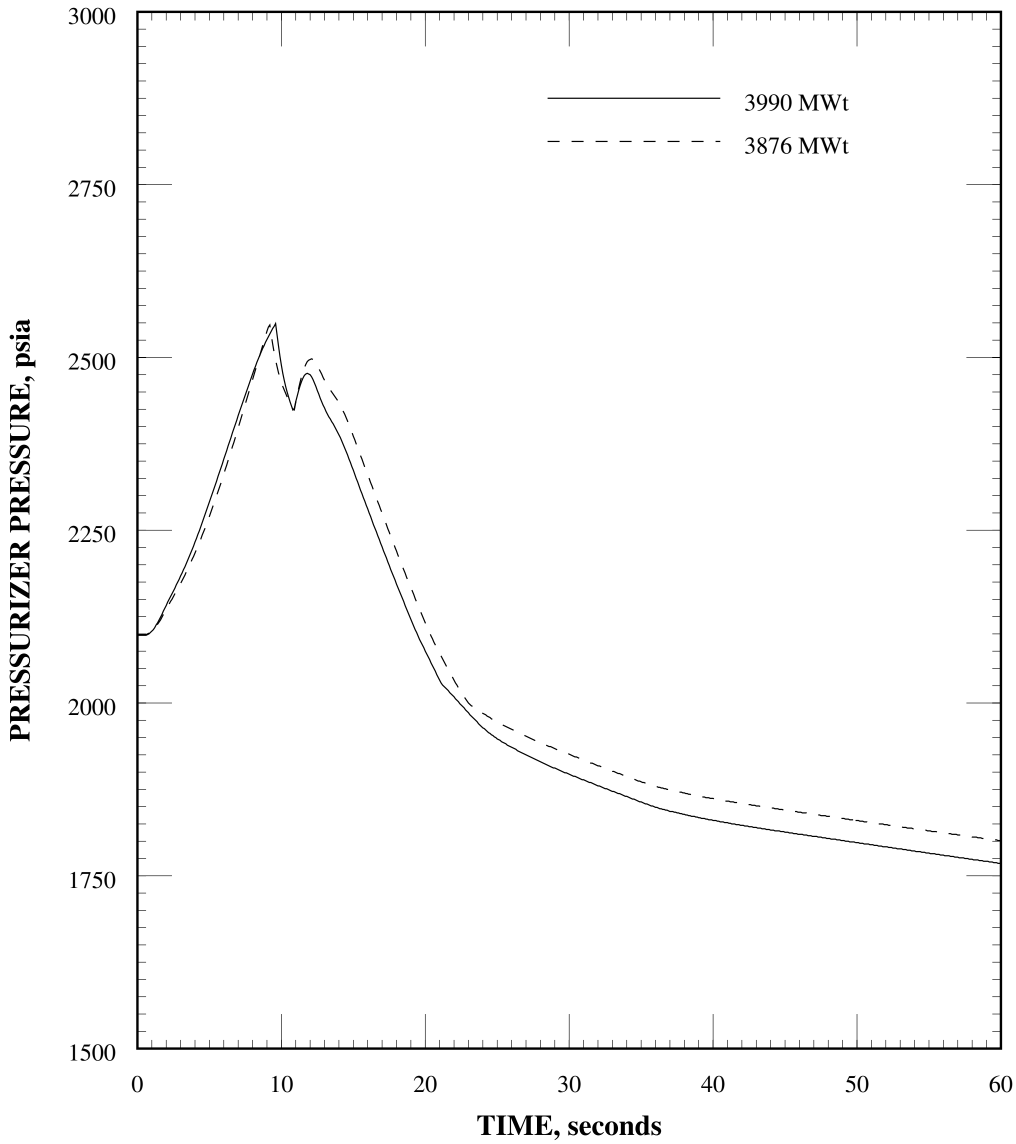


Figure 6.3-86
LOCV Secondary Peak Pressure Case - Pressurizer Water Volume vs. Time

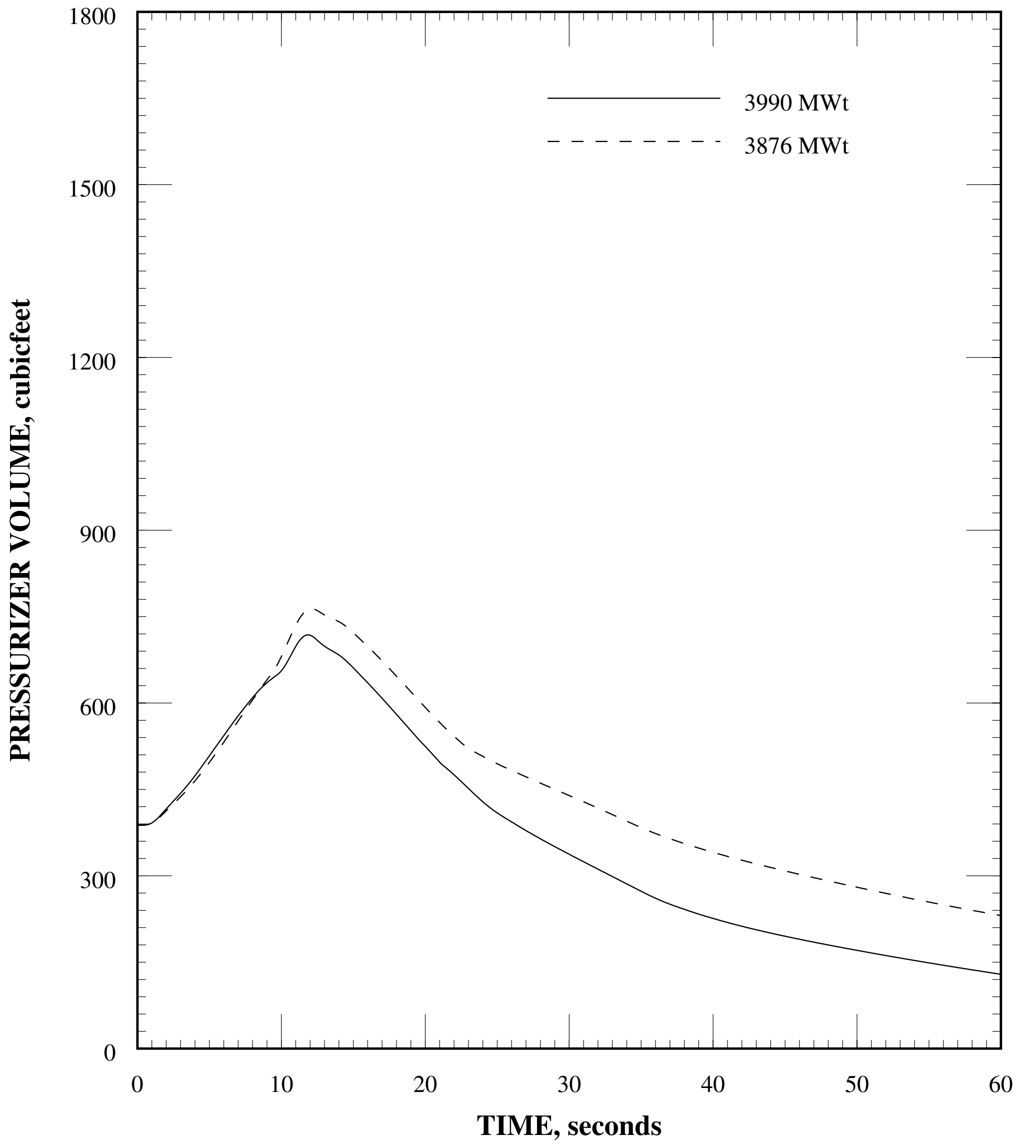


Figure 6.3-87
LOCV Secondary Peak Pressure Case - SG Pressure vs. Time

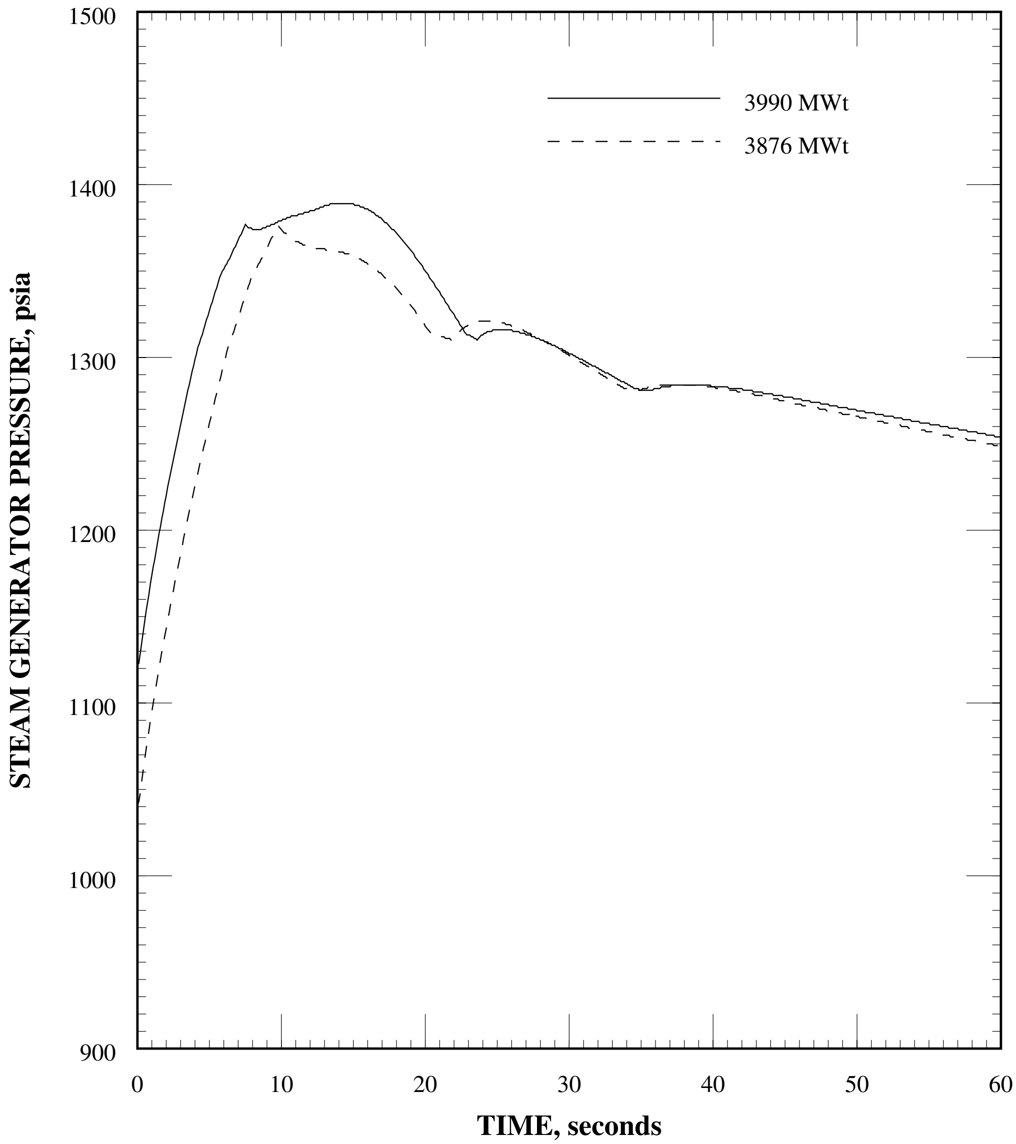


Figure 6.3-88
LOCV Secondary Peak Pressure Case - SG Level vs. Time

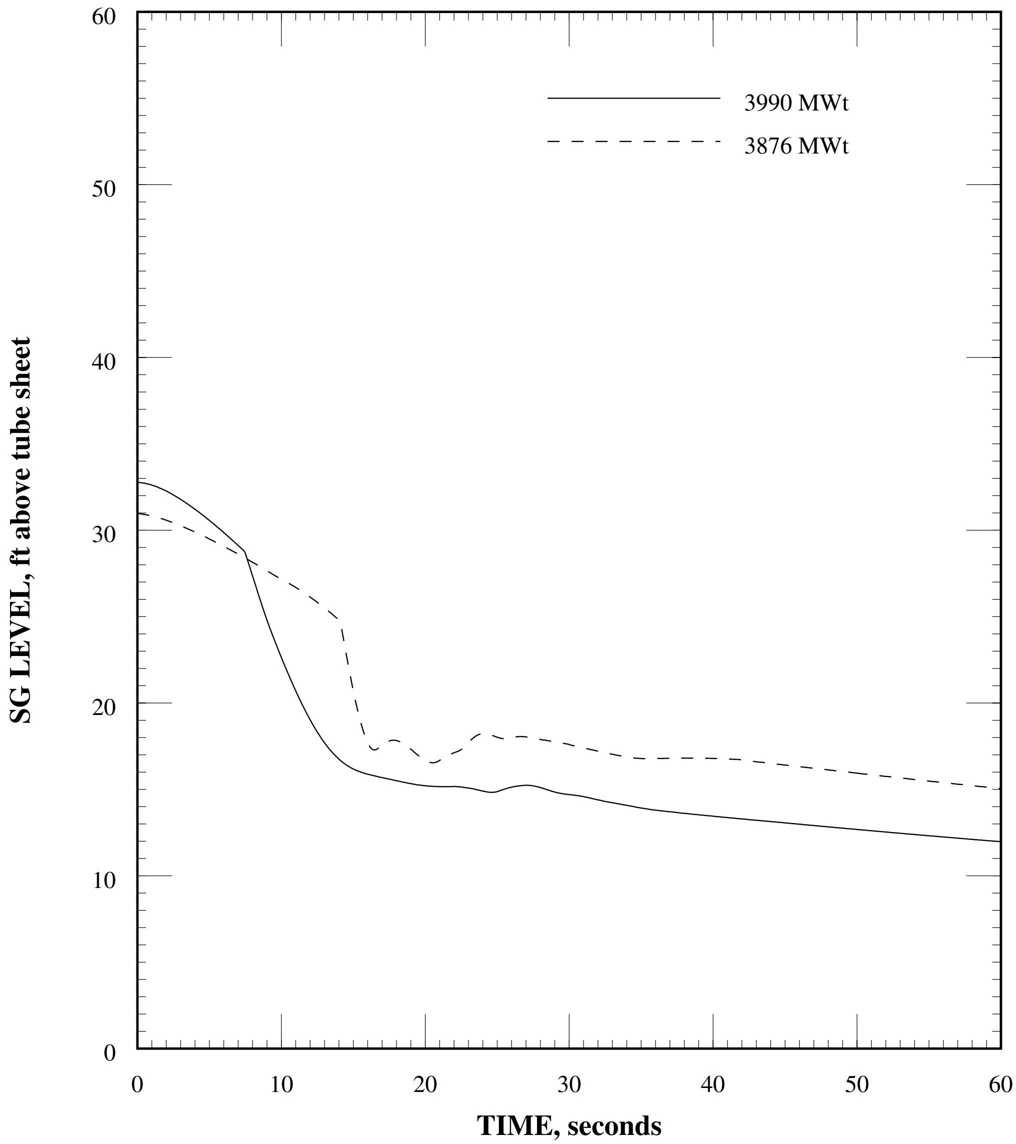


Figure 6.3-89
LOCV Secondary Peak Pressure Case - SG Liquid Inventory vs. Time

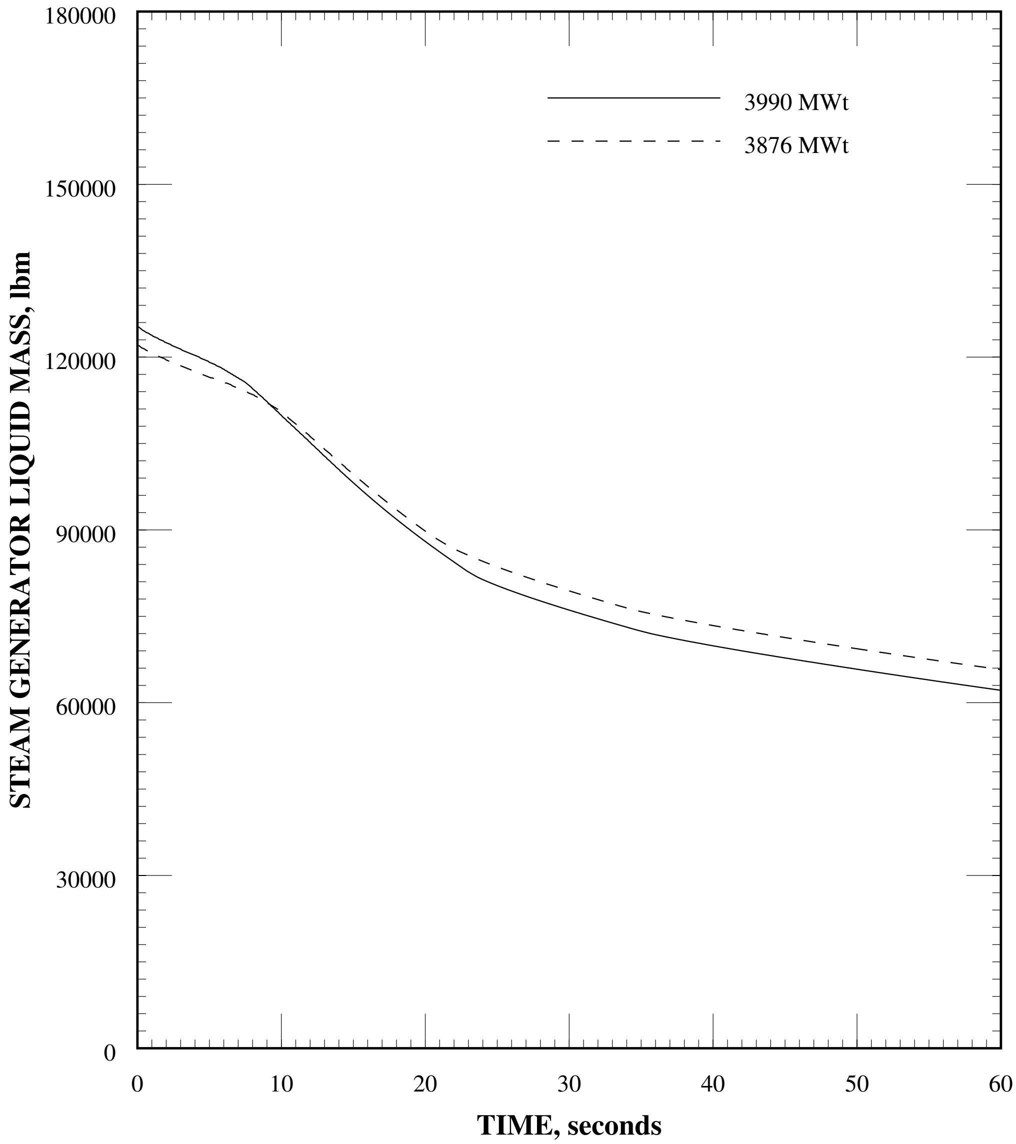


Figure 6.3-90
LOCV Secondary Peak Pressure Case - Integrated Steam Flow vs. Time

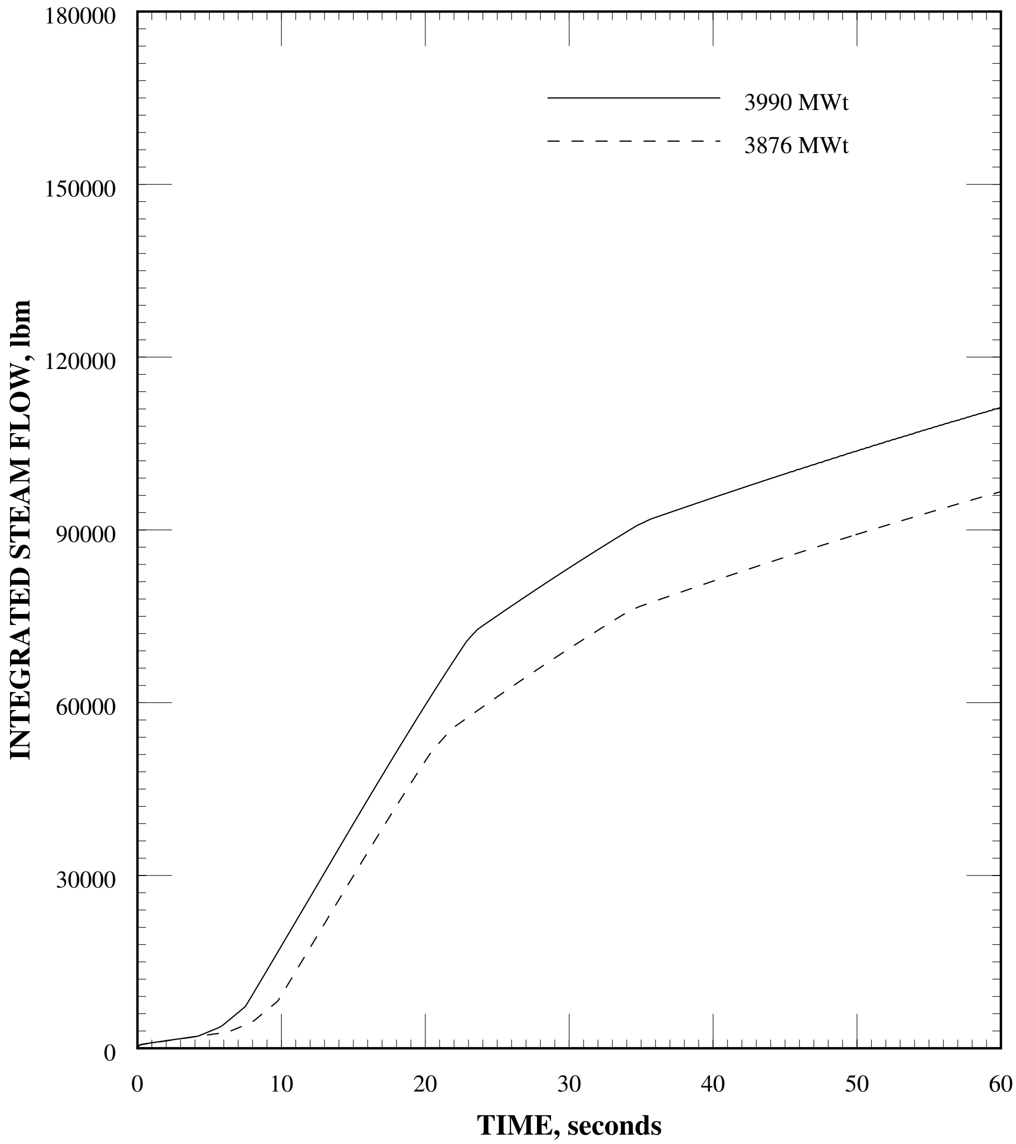


Figure 6.3-91
LOCV Secondary Peak Pressure Case - Total FW Flow vs. Time

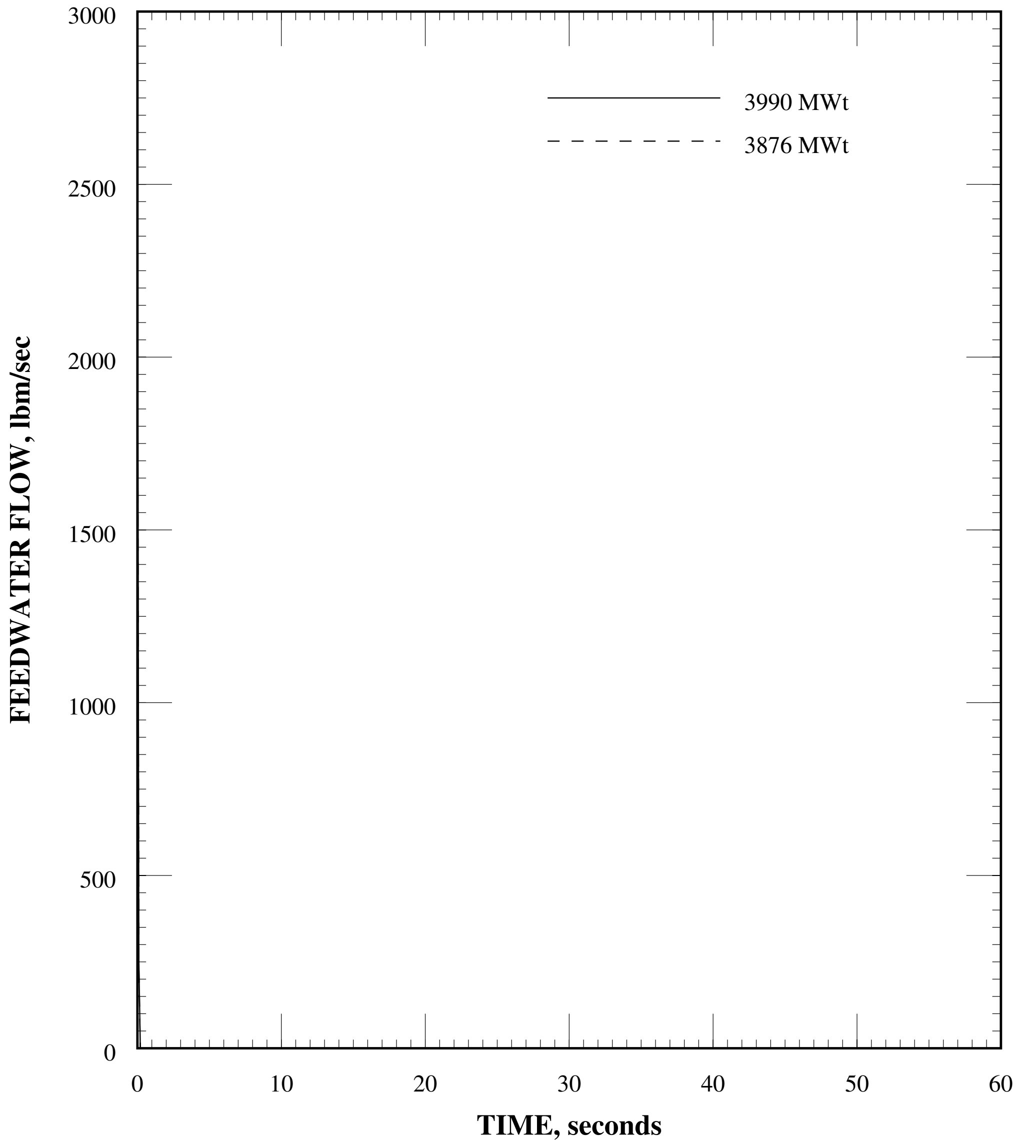
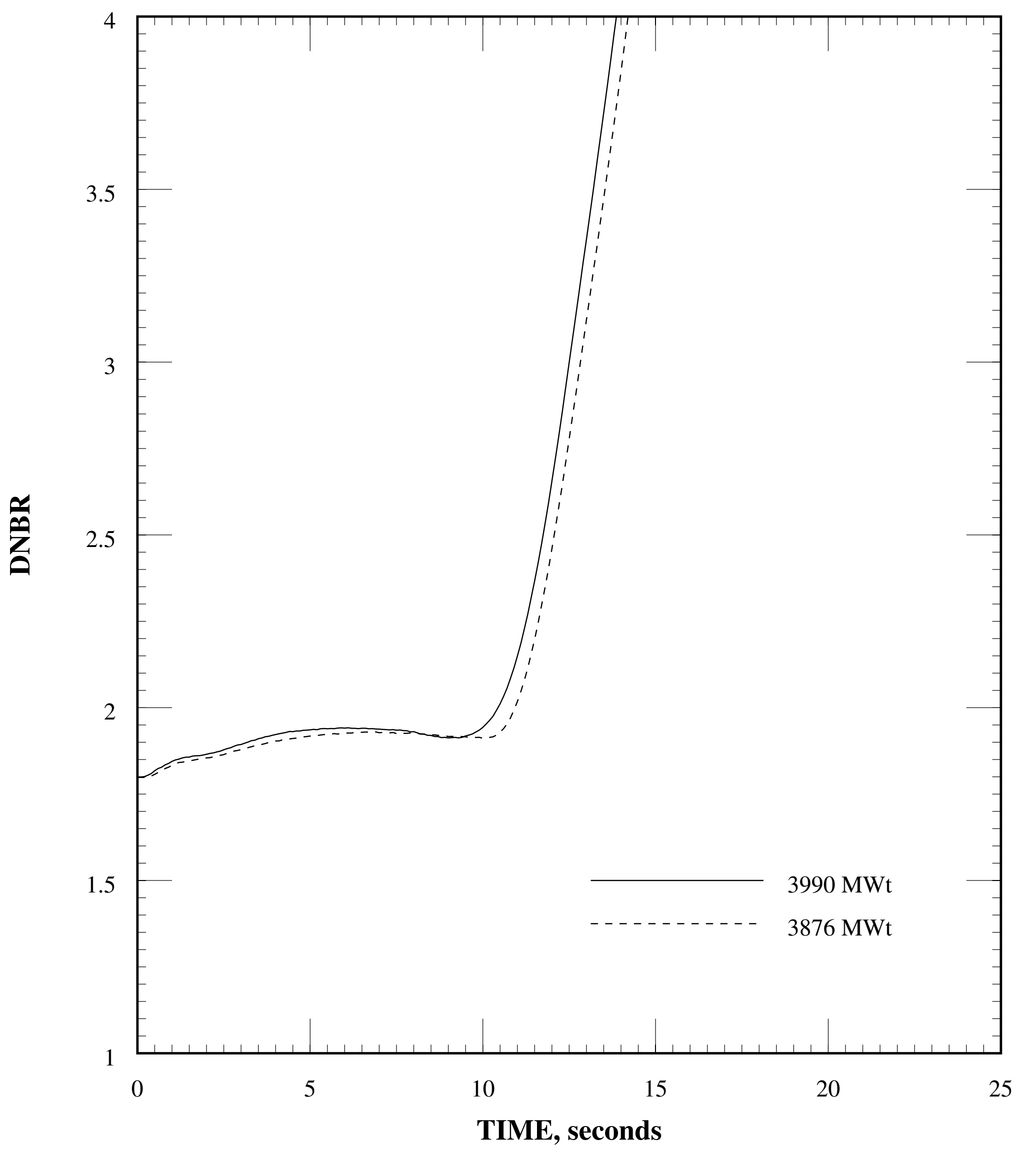


Figure 6.3-92
LOCV Secondary Peak Pressure Case - Minimum DNBR vs. Time



Section 6.3.2.4 Main Steam Isolation Valve Closure

As described in UFSAR Section 15.2.4 the MSIV closure event is initiated by the closure of all MSIVs due to a spurious closure signal. The closure of all MSIVs results in the termination of all main steam flow. The decreased heat removal results in increasing primary and secondary temperatures and pressure. Reactor trip occurs on high pressurizer pressure. The pressure increase in the RCS and secondary system pressure is limited by the PSVs and MSSVs. The operator can initiate a controlled system cooldown using the SBCS and ADVs any time after reactor trip occurs.

The MSIV closure event is classified as an AOO. This event was reviewed for the impacts of PUR and is bounded by the LOCV event. The LOCV event analysis is provided in Section 6.3.2.3.

Section 6.3.2.5 Steam Pressure Regulator Failure

As described in UFSAR Section 15.2.5, this event does not apply to the CE SYSTEM 80 design and therefore is not presented.

Section 6.3.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

As described in UFSAR Section 15.2.6, the loss of non-emergency AC power to the station auxiliaries may result from either a complete loss of the external grid or a loss of the onsite AC distribution system. The loss of AC power is presented as the initiating event for the four-RCP LOF event. When all normal AC power is lost, the turbine stop valves close. In addition, the FW flow to both SGs is assumed to go to zero. The RCPs coast down and the flow decreases. A reactor trip will occur because of a DNBR condition as the flow coastdown begins. The PSVs and MSSVs limit the pressure increase in the RCS and SGs.

The loss of AC power is followed by automatic start of the standby EDGs, the power output is sufficient to supply electrical power to all necessary ESF systems and to provide the capability of maintaining the plant in a safe shutdown condition. After the reactor trip, stored and fission product decay energy must be dissipated by the RCS and main steam system. Without forced RCS flow, convective heat transfer coolant flow occurs. Initially, residual water inventory in the SGs is used as a heat sink, and the resultant steam is released to atmosphere by the MSSVs. With the standby EDG power, AFW is automatically initiated on a low SG water level signal. Plant cooldown is operator controlled via the ADVs.

The loss of AC power event is classified as an AOO. This event was reviewed for the impacts of PUR. For peak RCS and secondary pressure, the loss of AC power event remains bounded by the LOCV event described in Section 6.3.2.3. For fuel performance (approach to DNBR SAFDL), the loss of AC power event remains identical to the LOF event.

Section 6.3.2.7 Loss of Normal Feedwater Flow

As described in UFSAR Section 15.2.7, the Loss of Normal Feedwater flow (LOFW) event may be initiated by losing one or both main FW pumps or by a spurious signal being generated by the Digital Feedwater Control System (DFWCS) resulting in a closure of the FWCV(s). LOFW results in decreasing level and increasing pressure and temperature in the SGs. The RCS pressure and temperature also rise until a reactor trip occurs either due to low SG level or high pressurizer pressure. Assuming the SBCS is in the manual mode of operation, termination of main steam flow due to closure of the turbine stop valves following reactor trip temporarily causes SG and RCS pressurization. The decrease in core heat rate after insertion of the CEAs and with the MSSVs opening restores the RCS to a new steady state condition. AFW flow is automatically initiated on low SG water level, assuring sufficient SG inventory for core decay heat removal and cooldown to SCS entrance conditions. The cooldown is operator controlled using the SBCS and the condenser.

The LOFW flow event is classified as an AOO. This event was reviewed for the impacts of PUR and remains bounded by the LOCV event. The LOCV event analysis is provided in Section 6.3.2.3.

Section 6.3.2.8 Feedwater System Pipe Breaks

As described in UFSAR Section 15.2.8, FWLB event is initiated by a break in main FW system piping. Depending on the break size and location, and the response of the FW system, the effects of a FWLB can vary from a rapid heatup to a rapid cooldown of the NSSS.

Breaks are categorized as small, if the associated discharge flow is within the excess capacity of the FW system, and otherwise as large. Break sizes that are less than or equal to 0.2 ft^2 area are considered as small breaks in this analysis.

Break locations are identified for the FW line reverse flow check valves that are located between the SG FW nozzles and the FW pumps. Closure of these valves to reverse flow from the nearest SG maintains the integrity of that generator in the presence of a break upstream of the valves.

Breaks upstream of the check valves can initiate one of the following transients. If the FW system is unavailable following the pipe failure, a total LOFW results. If the FW system remains in operation, no reduction in the FW flow occurs for small breaks, while for large breaks either a partial LOFW or a total LOFW occurs, if the break area is sufficient to discharge the entire FW pump flow capacity. In addition, breaks downstream of the check valves have the potential to establish reverse flow from the nearest SG (referred to as the “affected SG”) back through the break. Reverse flow occurs whenever FW is not operating after a pipe break or whenever FW is operating but does not provide sufficient capacity to maintain pressure at the break above the SG pressure. It is only the breaks that develop reverse flow that are of interest in this analysis in order to maximize the transient effects.

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 causes 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.

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. No credit is taken for three-second delay between the turbine trip and a LOP, and a LOP is assumed to occur at the same time of turbine trip. 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. 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.

Table 6.3-27
Parameters Used for FWLB with LOP Peak RCS Pressure Event

Parameter	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	562	566
Initial pressurizer pressure (psia)	2100	2140
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	11.4	11.4
Initial SG level (ft)	31.2	32.8
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)	16	10
PSV tolerance	+3%	+3%
PSV blowdown	5%	5%
MSSV tolerance	+3%	+3%
MSSV blowdown	5%	5%
Single failure	none	none
LOP	yes	yes
FWLB area (ft^2)	0.21	0.14

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)	548	548
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)	31.2	32.8
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	20%	20%
MSSV tolerance	+3%	+3%
MSSV blowdown	5%	5%
Single failure	one AFW pump	one AFW pump
LOP	yes	yes
FWLB area (ft^2)	0.25	0.21

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

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	FWLB and complete LOFW to both SGs, break size (ft^2)	0.21	0.14
24.41	27.90	Pressurizer pressure reaches trip setpoint (psia)	2450	2450
24.41	27.90	HPPT signal generated	---	---
24.59	28.11	Dryout of affected SG; AFAS generated in affected SG	<5000	<5000
24.91	28.40	Reactor trip breakers open	---	---
24.91	28.40	Turbine trip occurs	---	---
24.91	28.40	LOP occurs	---	---
25.51	29.00	Scram CEAs begin falling	---	---
26.11	29.55	PSVs open (psia)	2550	2550
27.17	30.89	Maximum RCS pressure (psia)	2728	2756
29.97	33.82	PSVs close (psia)	2422	2422
30.92	30.11	MSSVs bank 1 open on unaffected SG (psia)	1303	1303
31.23	35.90	PSVs open (psia)	2550	2550
31.35	30.32	MSSVs bank 1 open on affected SG (psia)	1303	1303
---	33.93	MSSVs bank 2 open on unaffected SG (psia)	1344	1344
32.05	36.66	PSVs close (psia)	2422	2422
34.06	33.99	Peak secondary pressure occurs	1317	1355
35.39	37.65	Maximum liquid volume of pressurizer (ft^3)	996	1015
---	45.01	MSSVs bank 2 close on unaffected SG (psia)	1277	1277
46.10	52.85	MSSVs bank 1 close on affected SG (psia)	1238	1238
47.18	55.56	MSSVs bank 1 close on unaffected SG (psia)	1238	1238
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the long-term cooling FWLB event	---	---
1800	1800	Operator initiates the cooldown (min)	30	30

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	0.21
22.07	23.77	Pressurizer pressure reaches trip setpoint (psia)	2450	2450
22.07	23.77	HPPT signal generated	---	---
22.08	23.79	PSVs open (psia)	2451	2451
22.24	23.92	Dryout of affected SG; AFAS generated in affected SG	<5000	<5000
22.57	24.27	Reactor trip breakers open	---	---
22.57	24.27	Turbine trip occurs	---	---
22.57	24.27	LOP occurs	---	---
22.62	24.32	Maximum RCS pressure (psia)	2542	2558
23.17	24.87	Scram CEAs begin falling	---	---
30.45	32.15	PSVs close (psia)	1960	1960
68.24	69.92	AFW initiated (one pump, gpm)	650	650
124.9	142.8	MSIS generated (psia)	810	875
141.9	162.1	AFW lockout (psid)	270	270
366.8	356.8	PSVs open (psia)	2451	2451
368.6	360.4	PSVs close (psia)	1960	1960
495.4	480.2	PSVs open (psia)	2451	2451
500.0	486.1	PSVs close (psia)	1960	1960
523.0	524.5	MSSVs bank 1 open (psia) (1)	1303	1303
530.6	533.3	MSSVs bank 1 close (psia)	1238	1238
716.5	480.2	Maximum liquid volume of pressurizer (ft^3)	1743	1742
743.7	715.6	PSVs open (psia)	2451	2451
751.0	723.5	PSVs close (psia)	1960	1960
1800	1800	Operator initiates the cooldown (min)	30	30

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

Section 6.3.2.8.1.6 Conclusions

For FWLB with a LOP resulting from turbine trip and a single failure, all acceptance criteria are met for a full spectrum of FWLB sizes. The peak primary pressure remains below 120% of design at all times thus ensuring the integrity of the RCS. The secondary system pressure remains below 120% of design pressure ensuring integrity of the main steam system. The minimum DNBR remains above the safety analysis fuel design limit thus ensuring fuel cladding integrity. AFW capacity is adequate to provide removal of the core decay heat until operator action is taken 30 minutes after the initiating event.

Figure 6.3-93
FWLB with LOP Primary Peak Pressure Case - Core Power vs. Time

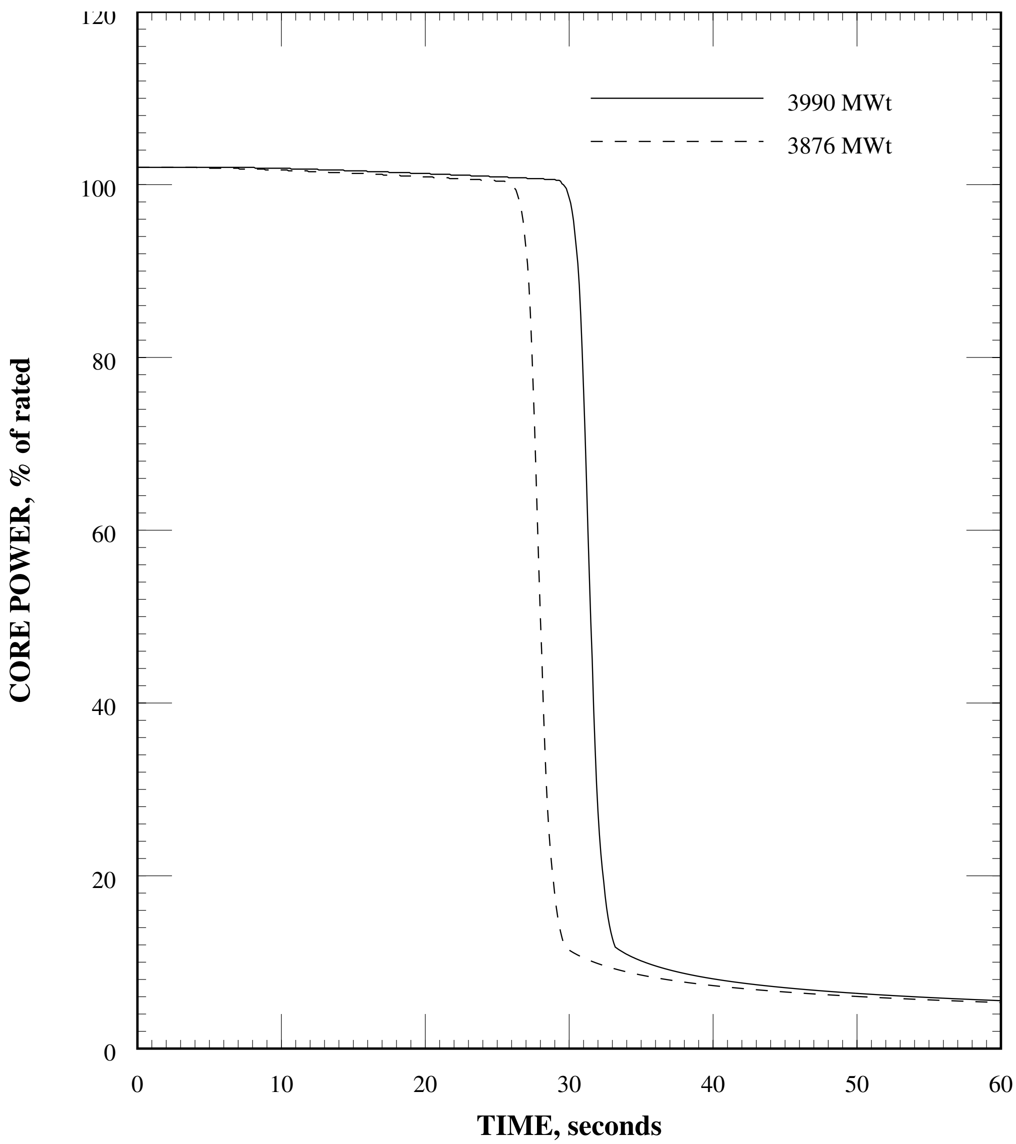


Figure 6.3-94
FWLB with LOP Primary Peak Pressure Case - Core Heat Flux vs. Time

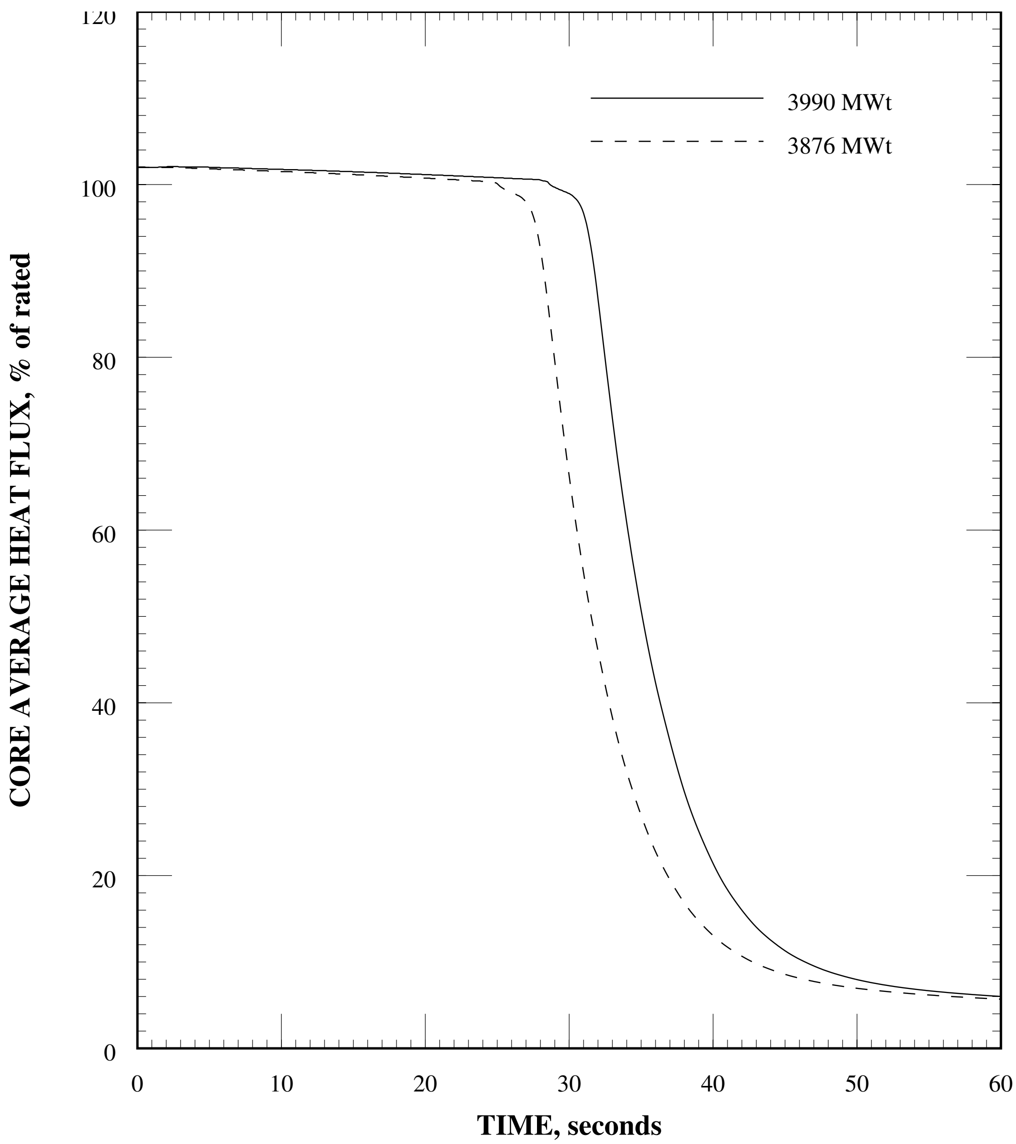


Figure 6.3-95
FWLB with LOP Primary Peak Pressure Case - Reactivity vs. Time

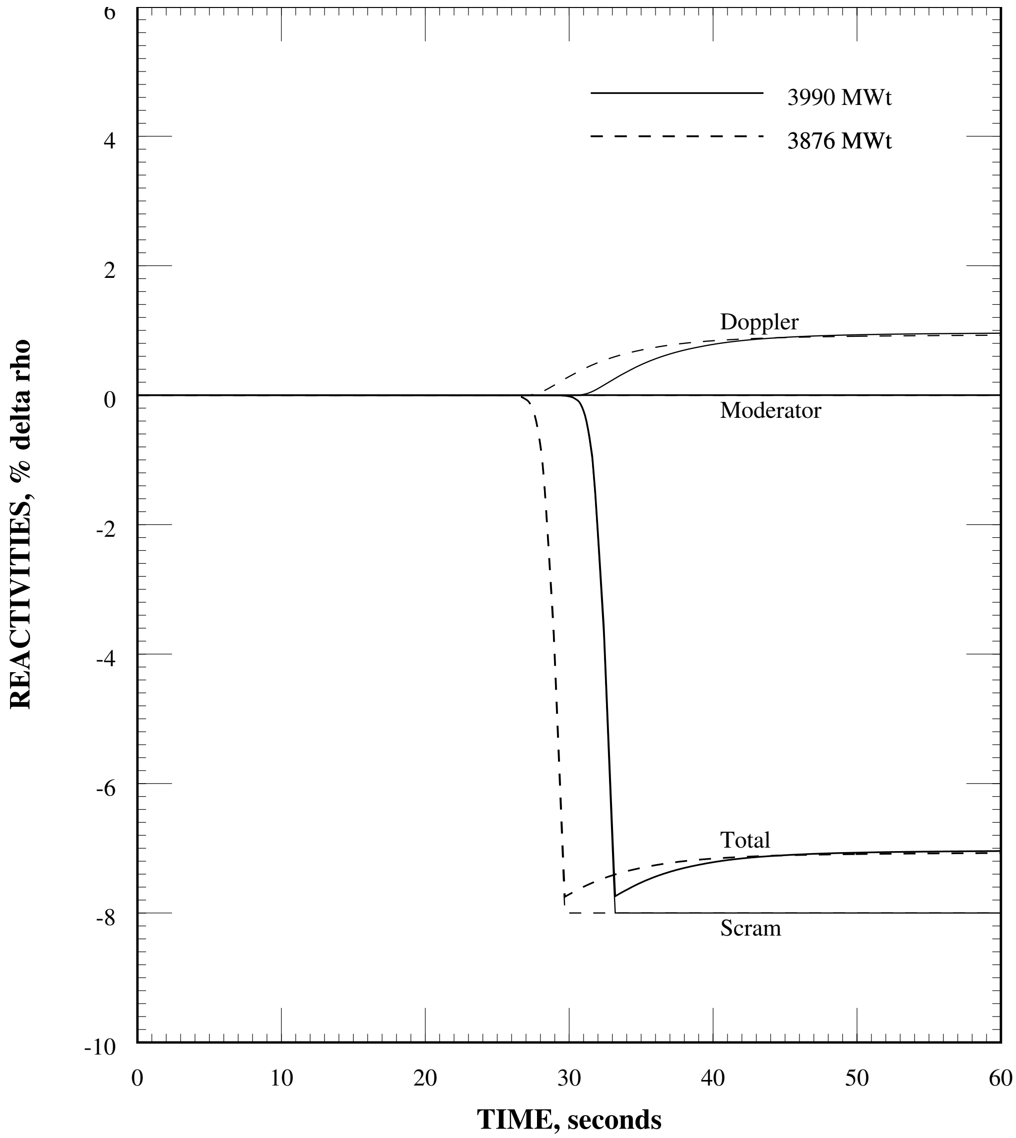


Figure 6.3-96
FWLB with LOP Primary Peak Pressure Case – Affected Loop RCS Temperatures vs. Time

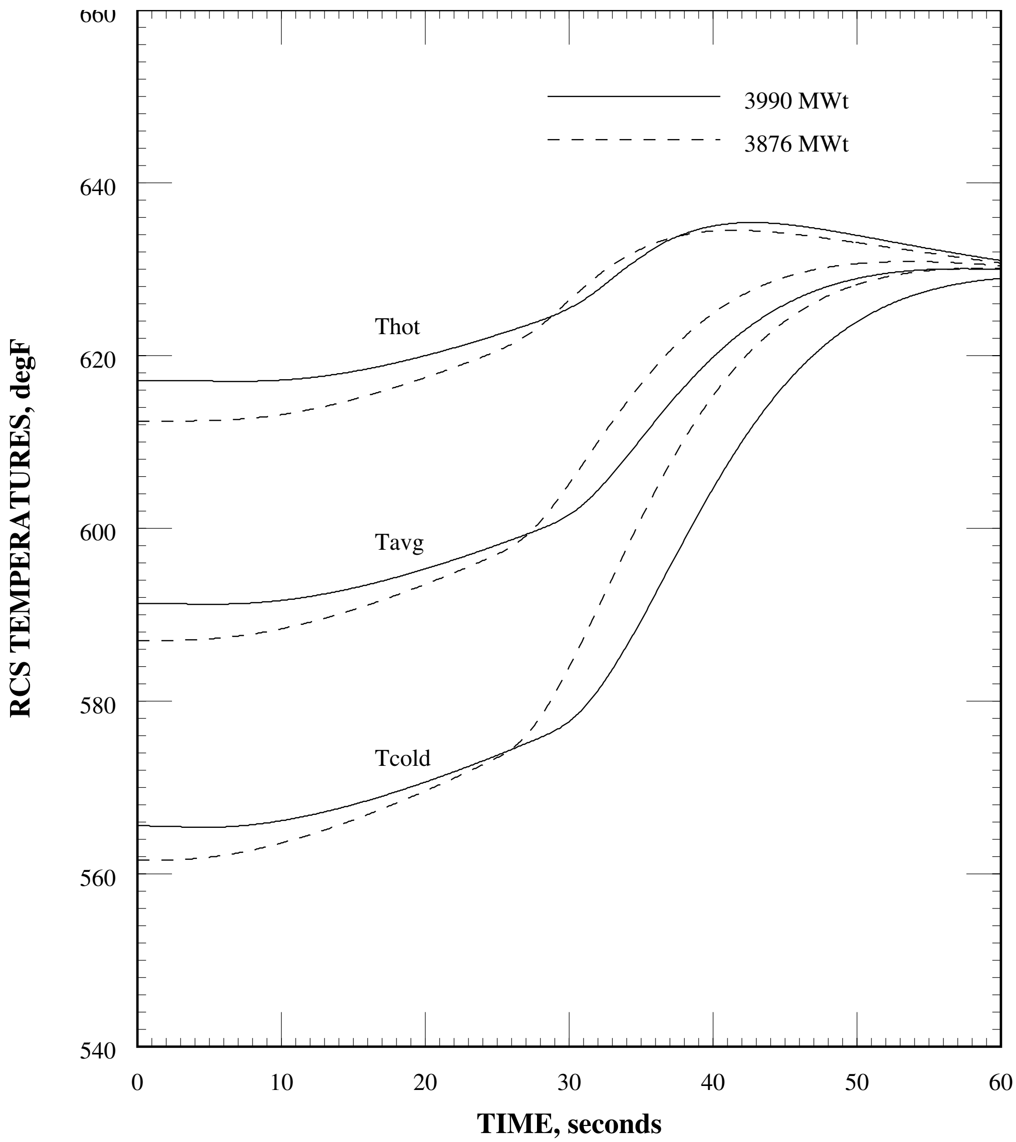


Figure 6.3-97
FWLB with LOP Primary Peak Pressure Case - RCS Pressure vs. Time

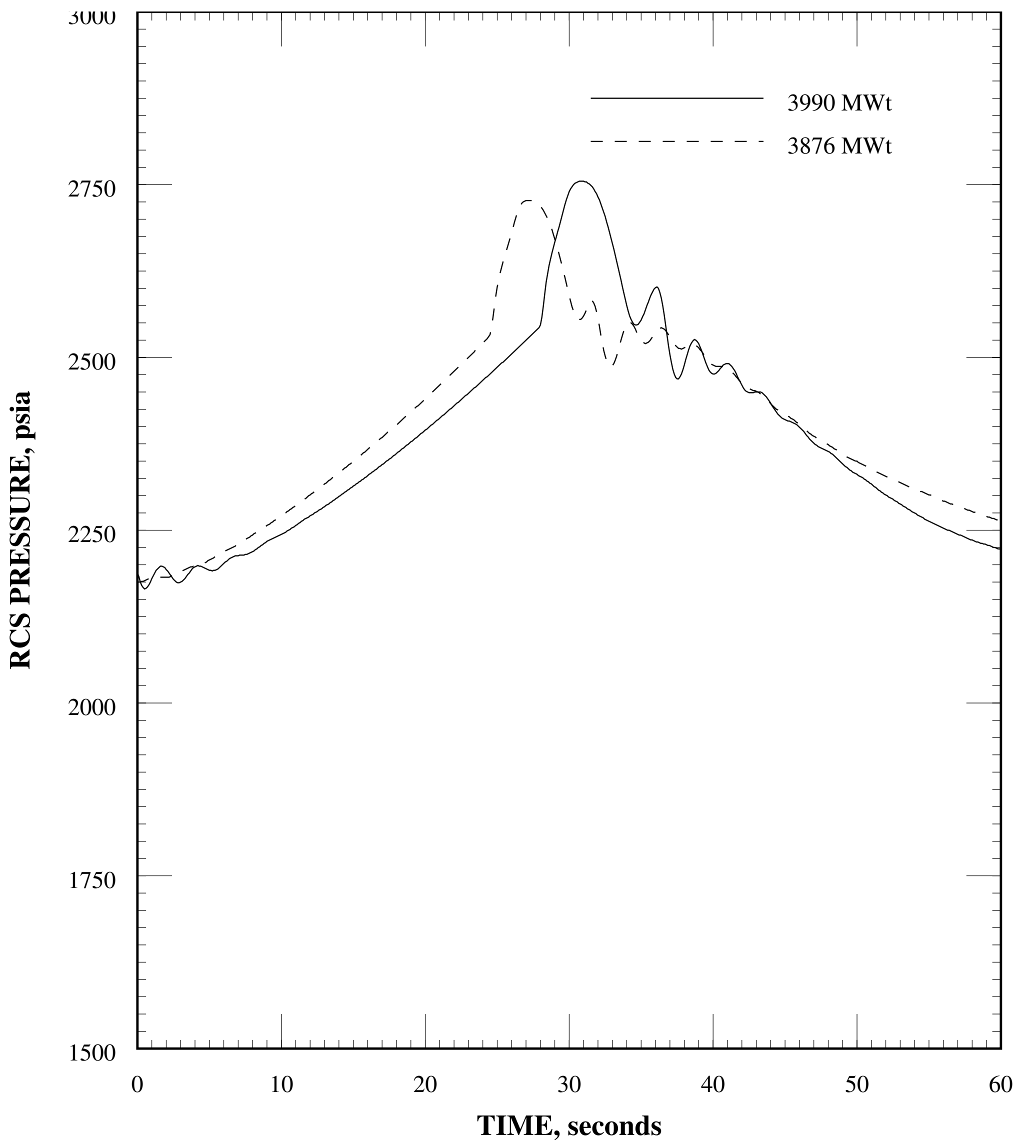


Figure 6.3-98
FWLB with LOP Primary Peak Pressure Case - Pressurizer Pressure vs. Time

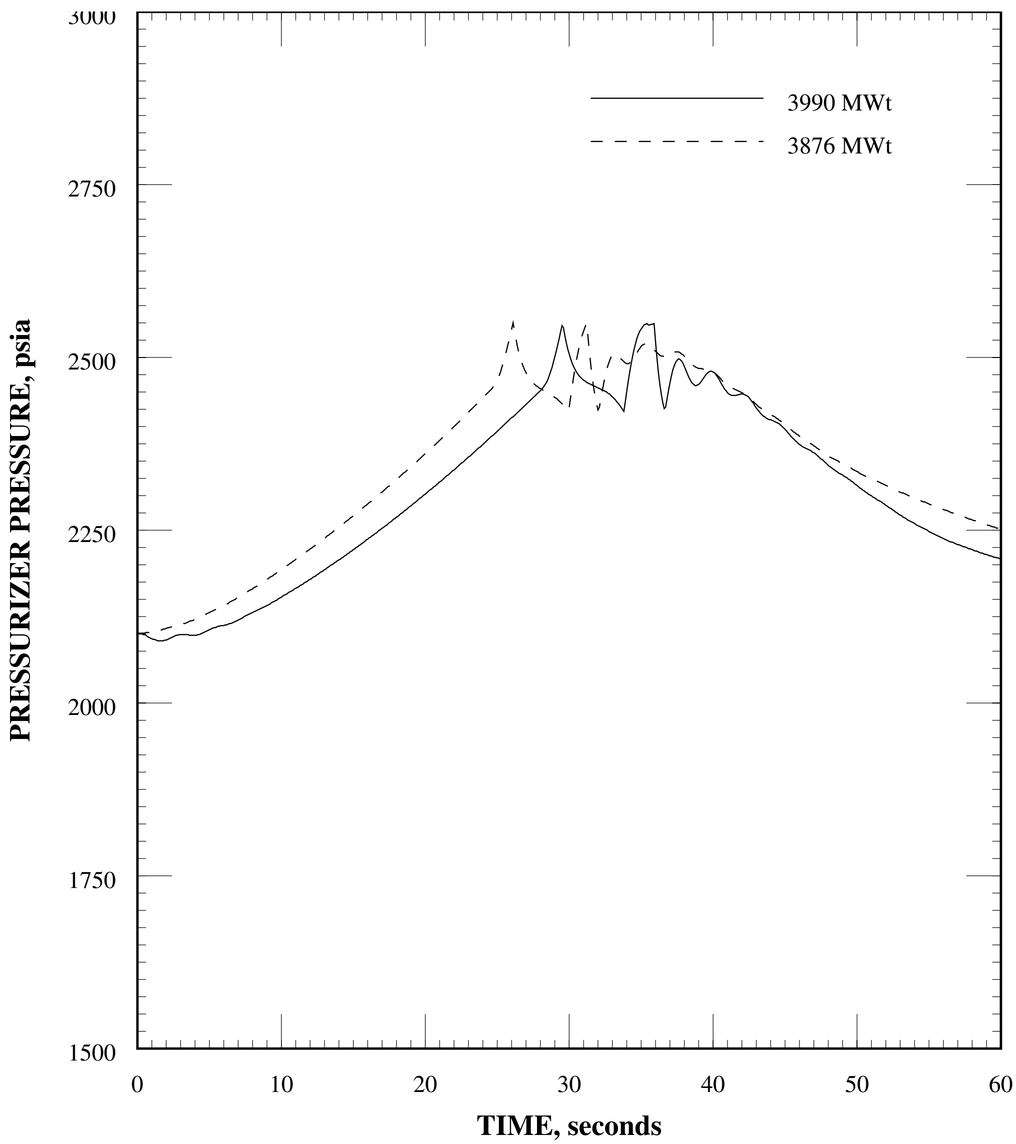


Figure 6.3-99
FWLB with LOP Primary Peak Pressure Case - Pressurizer Water Volume vs. Time

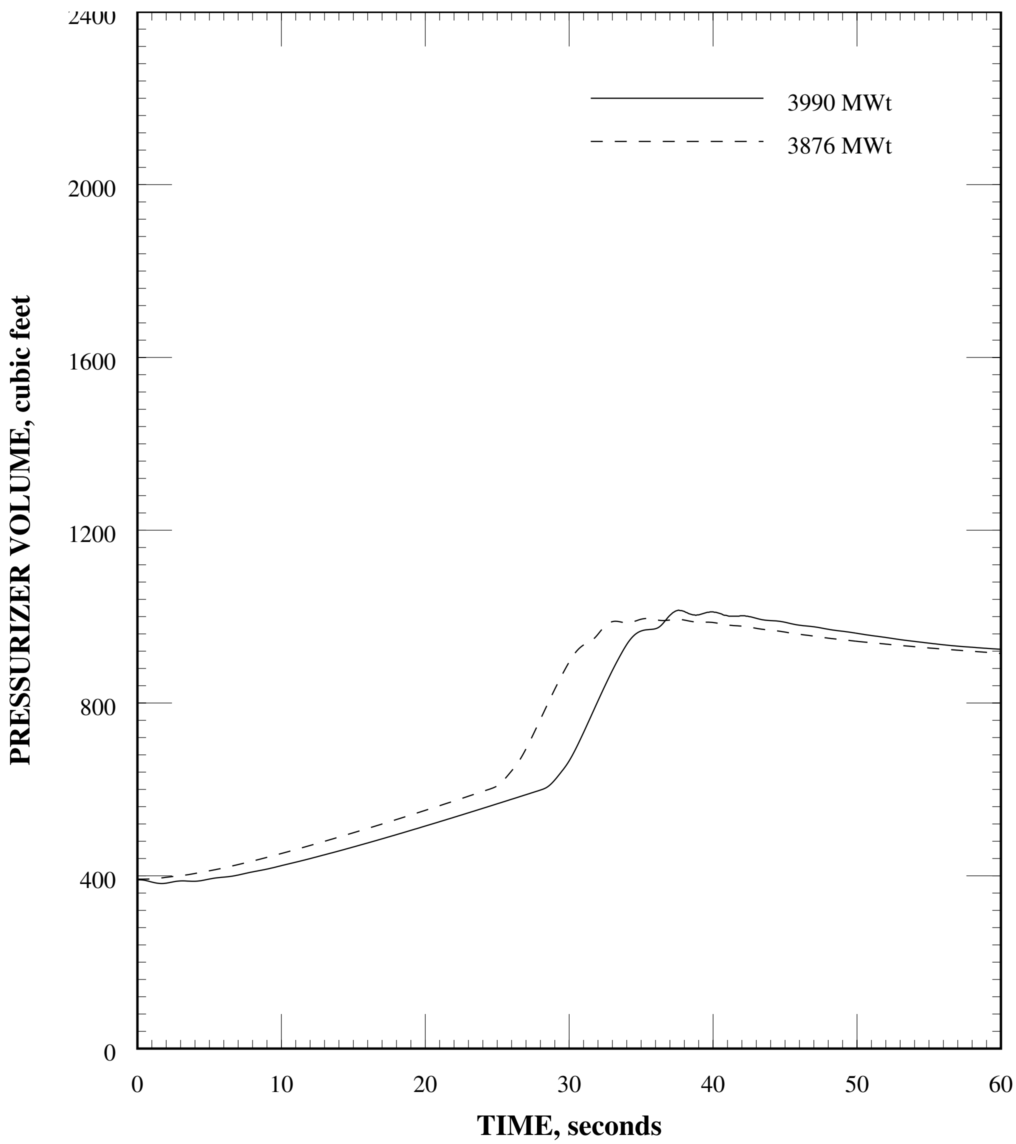


Figure 6.3-100
FWLB with LOP Primary Peak Pressure Case - SG Pressures vs. Time

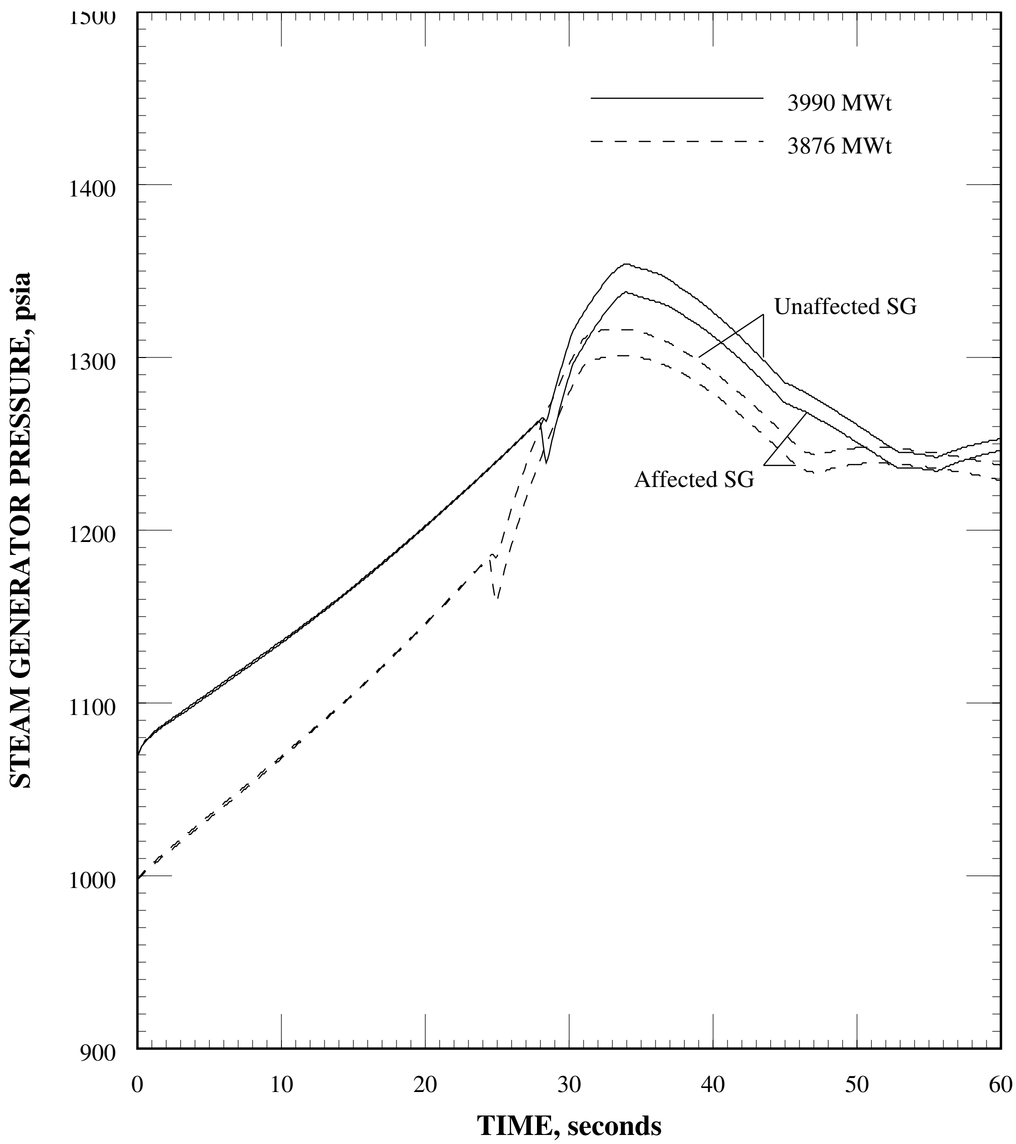


Figure 6.3-101
FWLB with LOP Primary Peak Pressure Case - SG Levels vs. Time

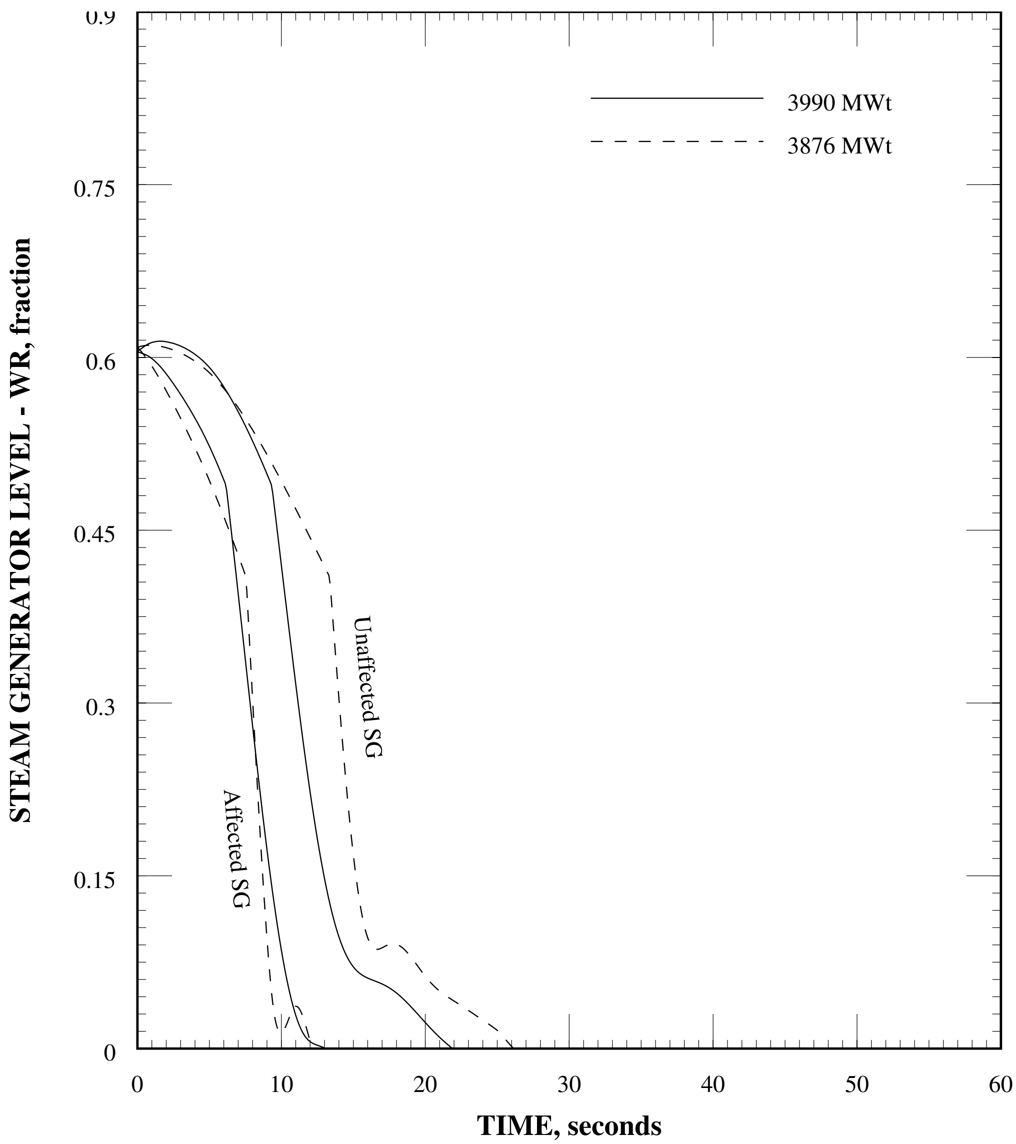


Figure 6.3-102
FWLB with LOP Primary Peak Pressure Case - SG Liquid Inventories vs. Time

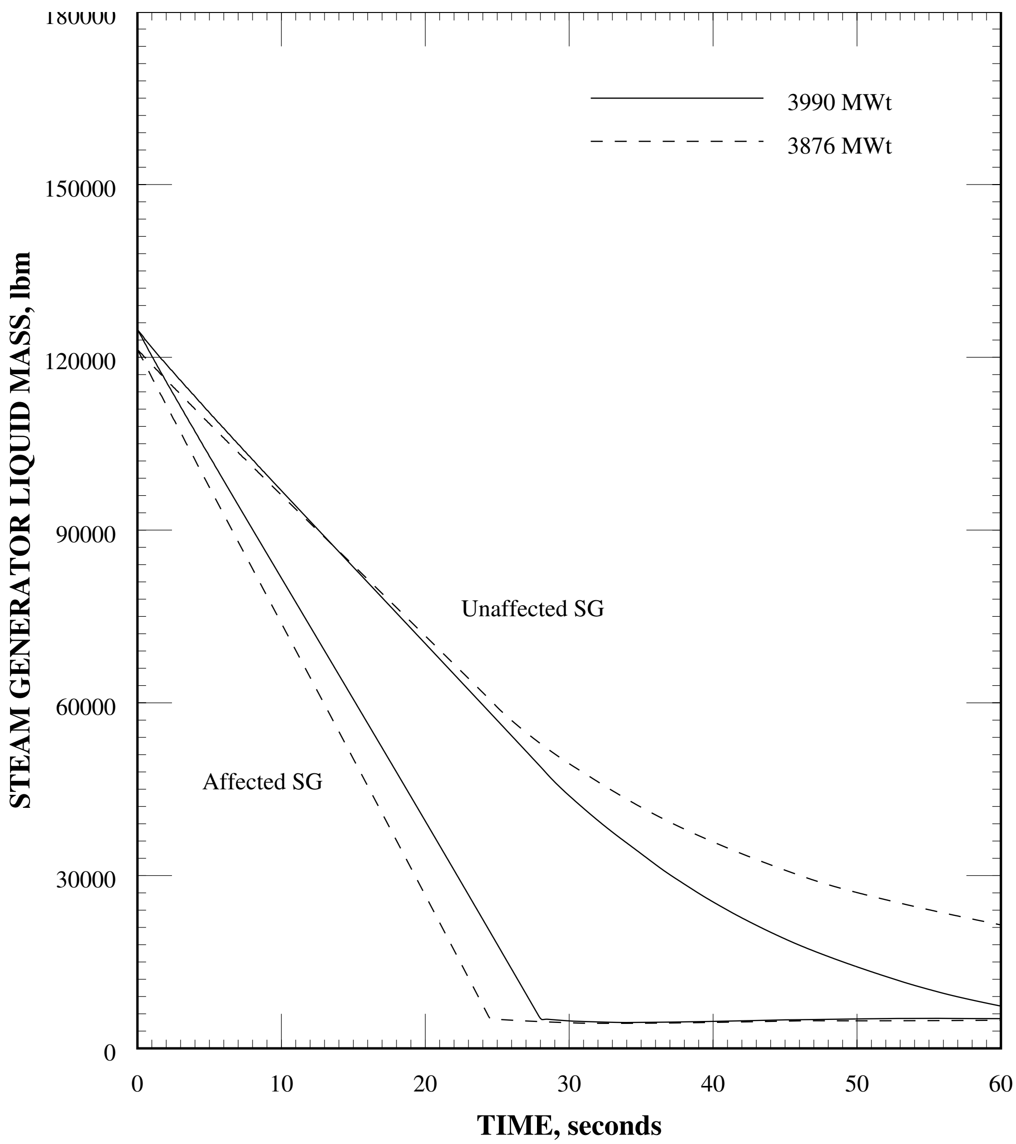


Figure 6.3-103
FWLB with LOP Primary Peak Pressure Case - RCS Loop Flows vs. Time

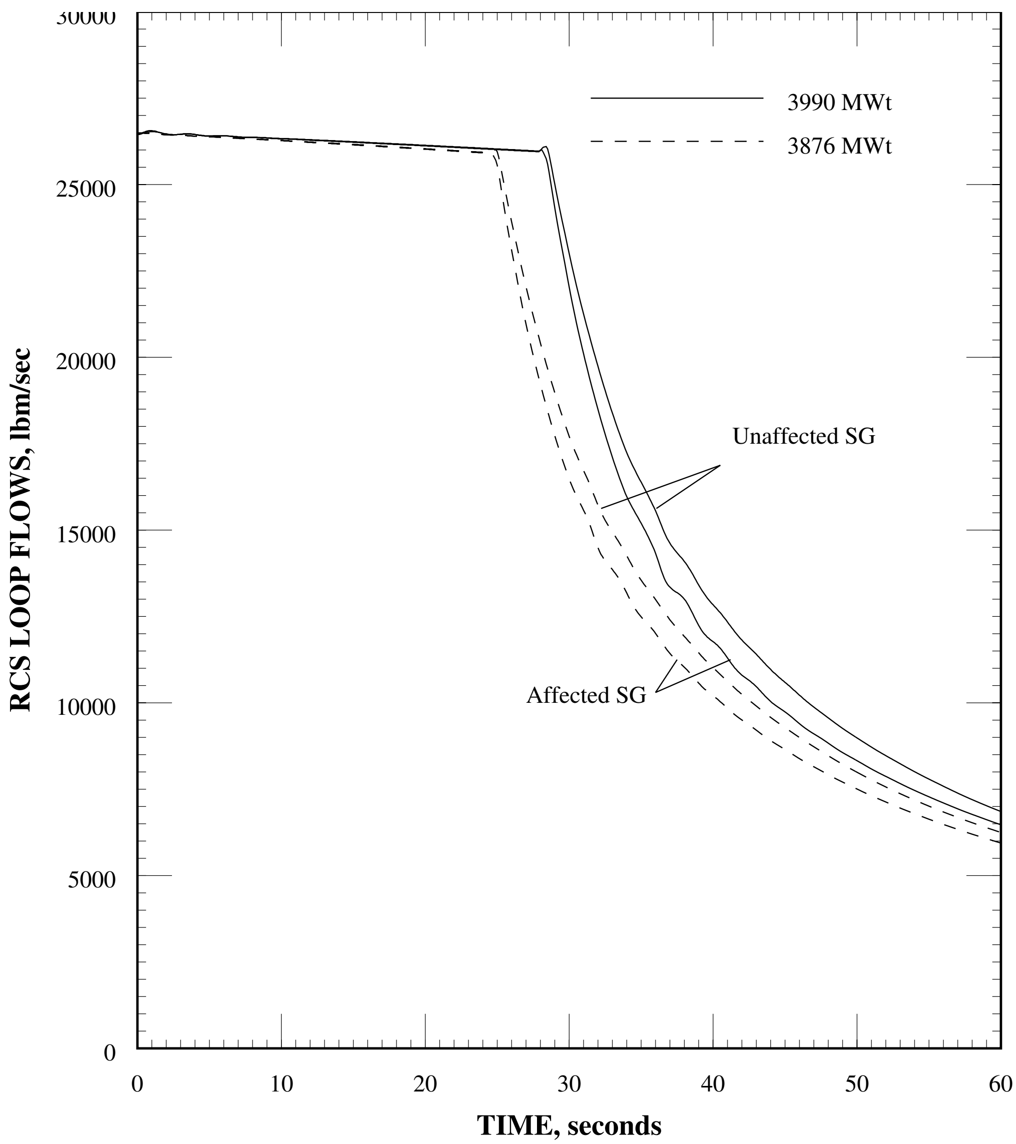


Figure 6.3-104
FWLB with LOP Primary Peak Pressure Case - SG Steam Flows vs. Time

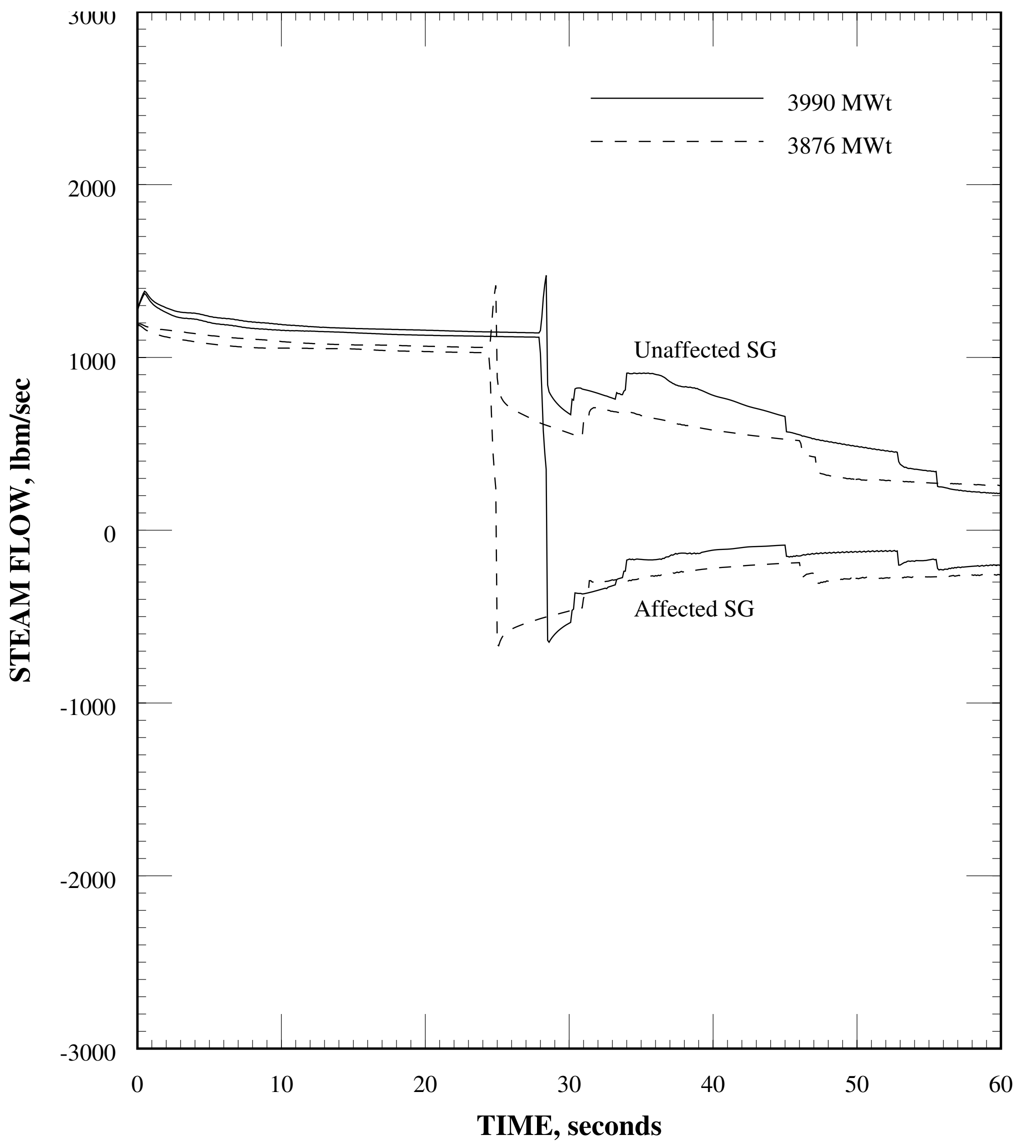


Figure 6.3-105
FWLB with LOP Primary Peak Pressure Case - Break Flow vs. Time

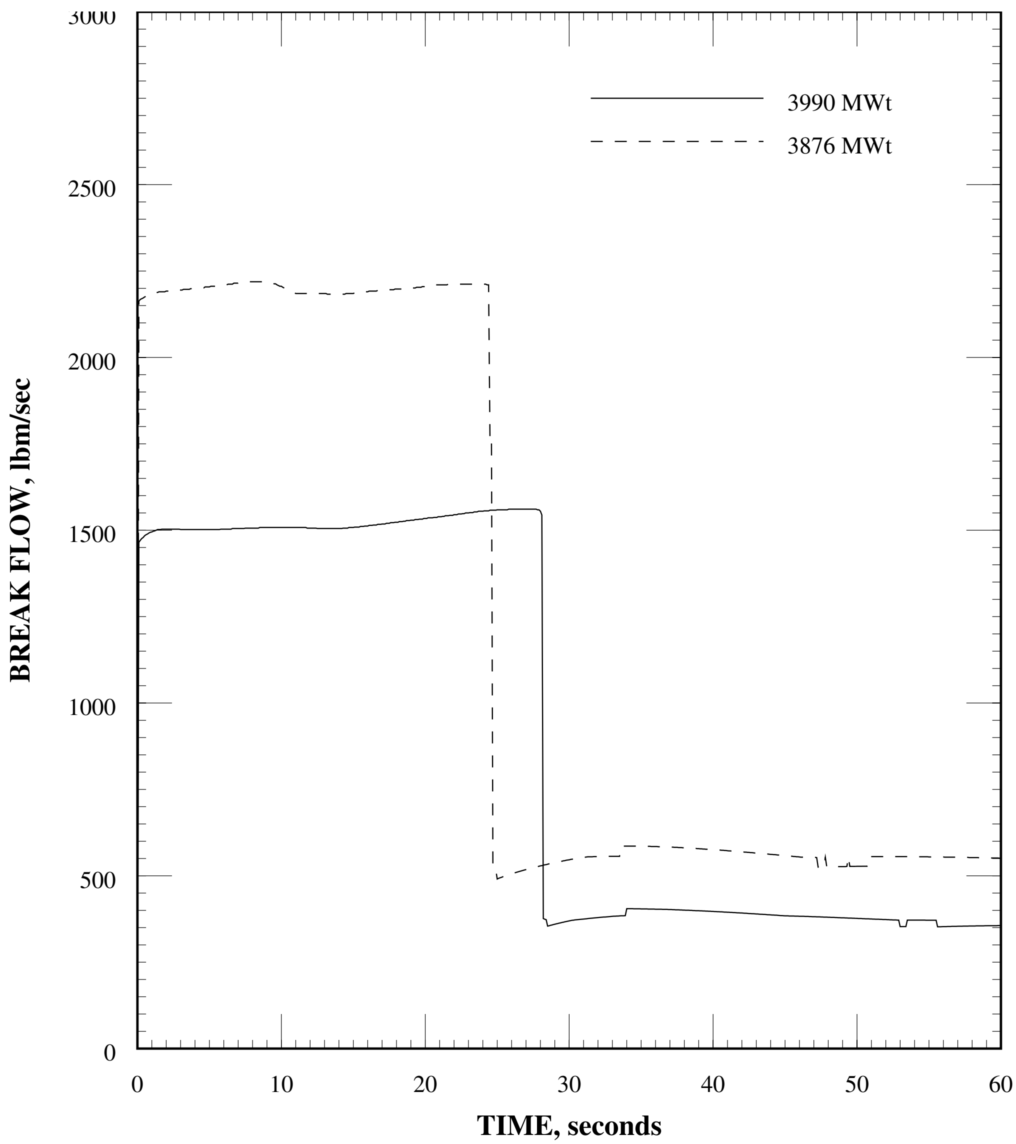


Figure 6.3-106
FWLB with LOP Primary Peak Pressure Case - Break Enthalpy vs. Time

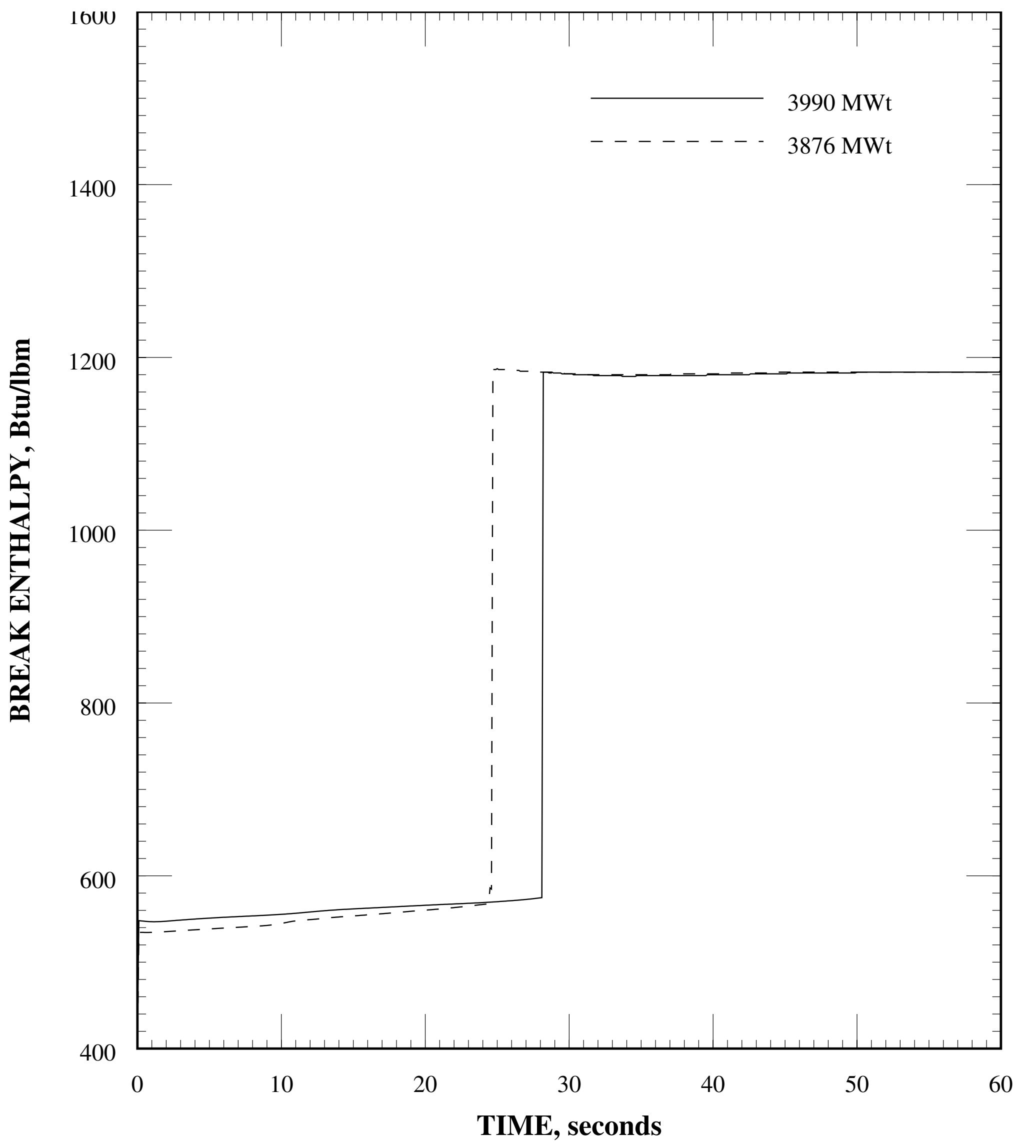


Figure 6.3-107
FWLB with LOP Primary Peak Pressure Case - PSV Flow vs. Time

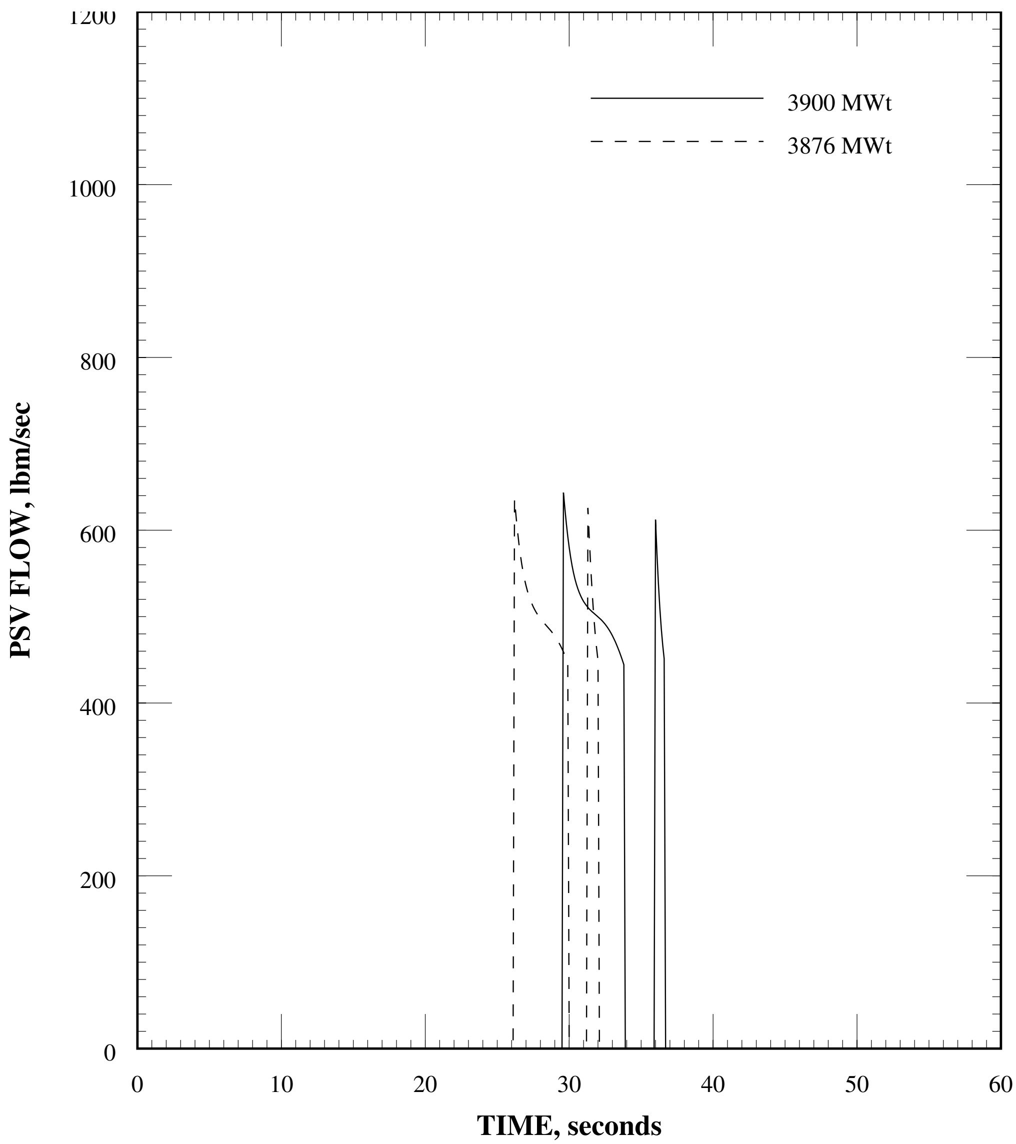


Figure 6.3-108
FWLB with LOP Primary Peak Pressure Case - Surge Flow vs. Time

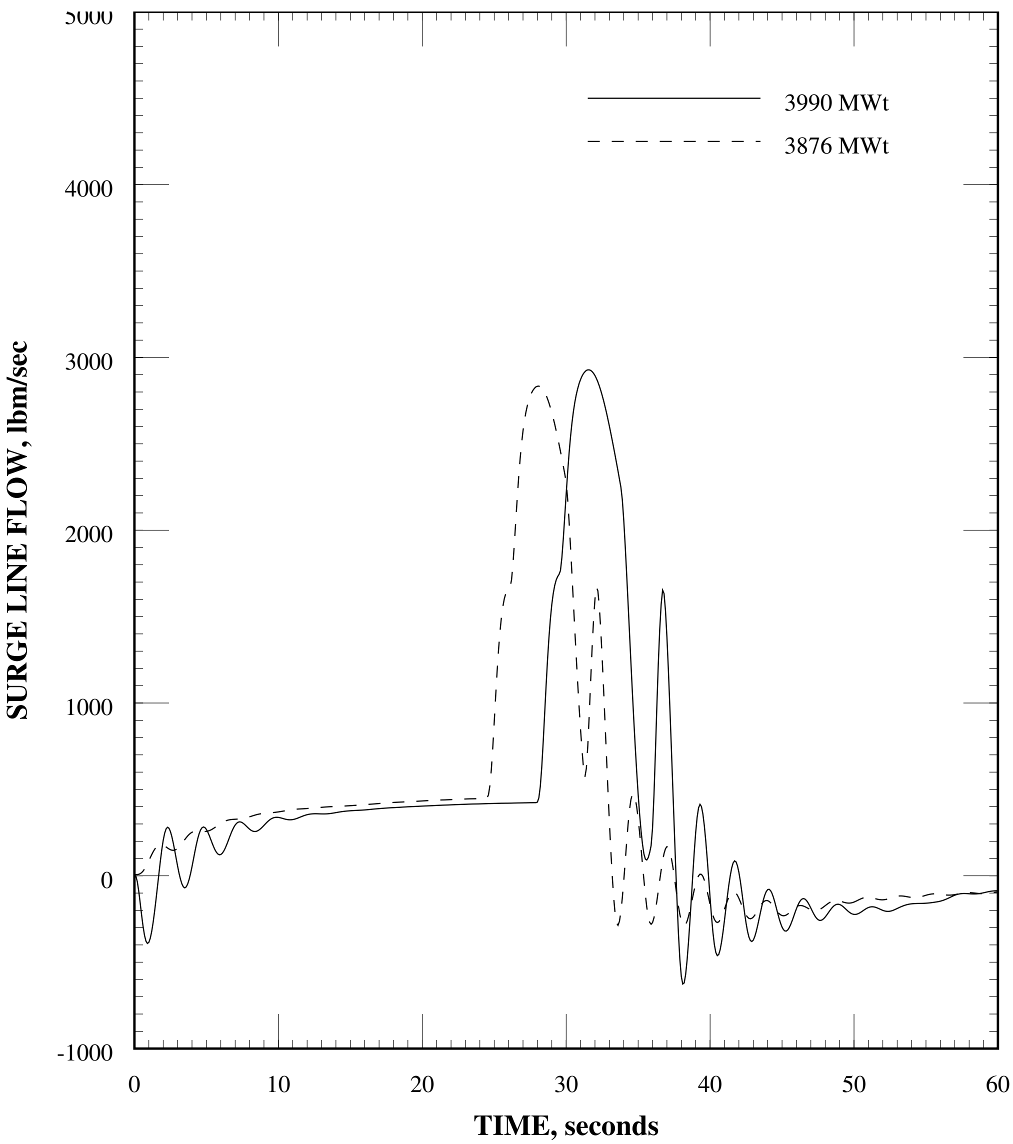


Figure 6.3-109
FWLB with LOP Primary Peak Pressure Case - Minimum DNBR vs. Time

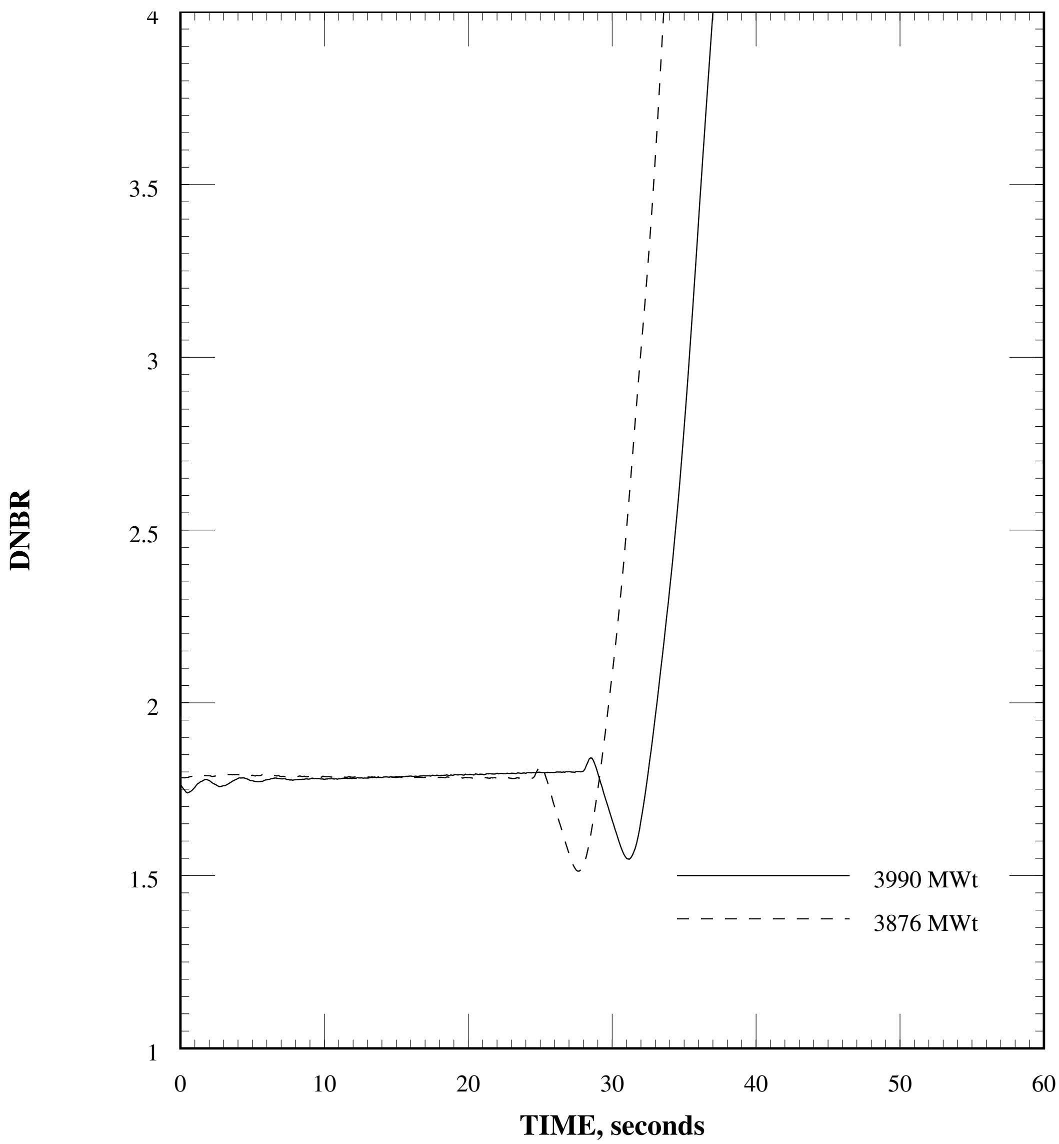


Figure 6.3-110
FWLB with LOP Long-Term Cooling Case - Core Power vs. Time

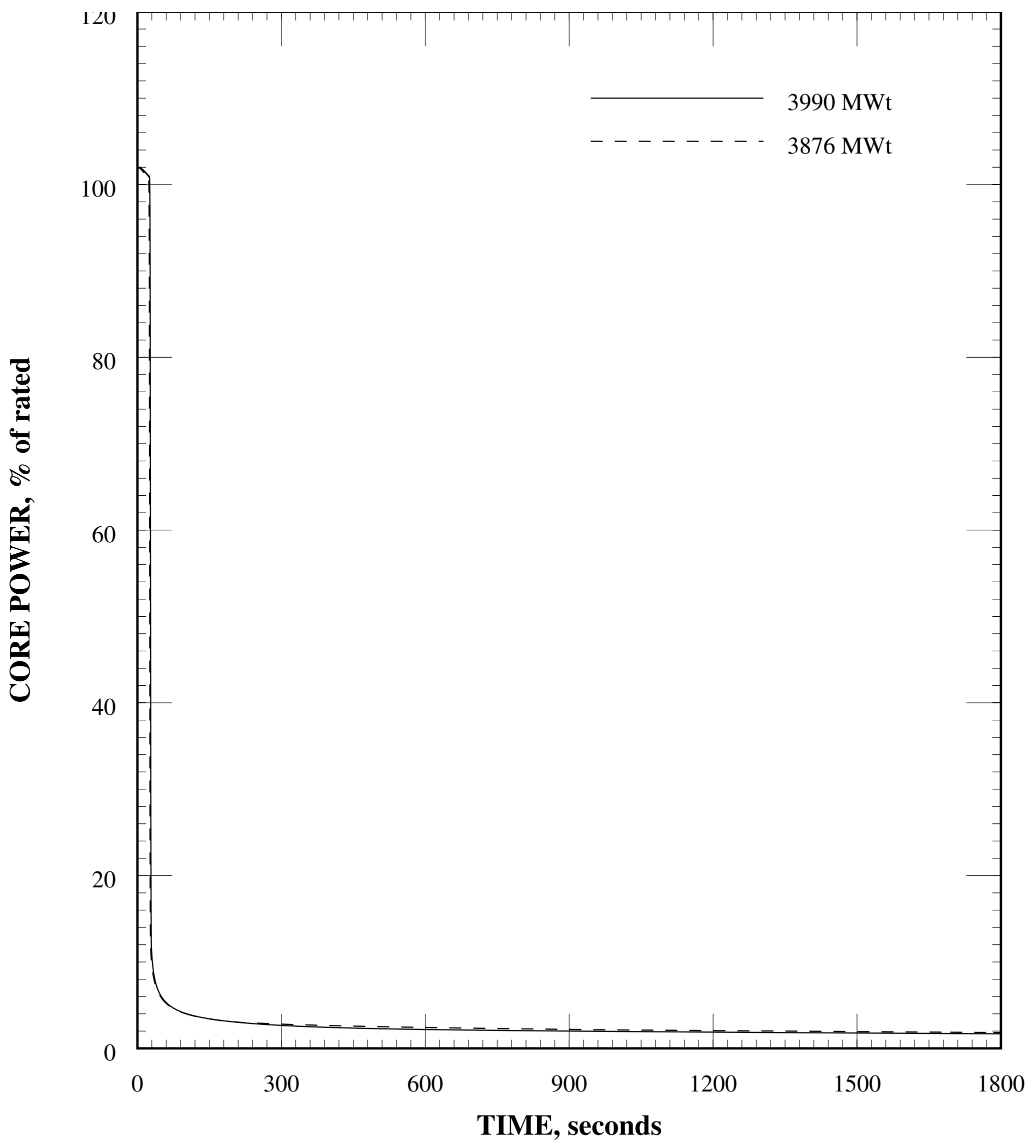


Figure 6.3-111
FWLB with LOP Long-Term Cooling Case – Affected Loop RCS Temperatures vs. Time

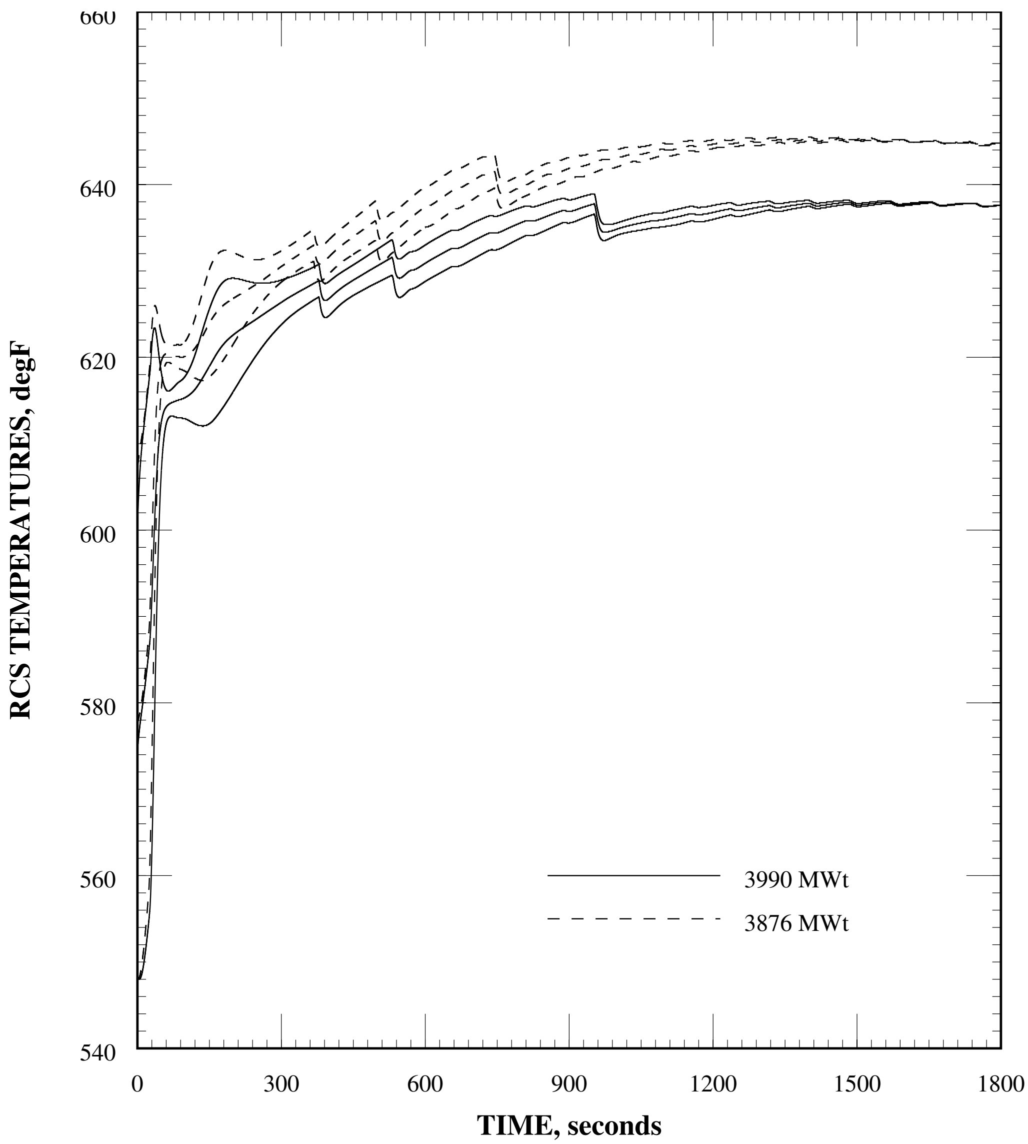


Figure 6.3-112
FWLB with LOP Long-Term Cooling Case - RCS Pressure vs. Time

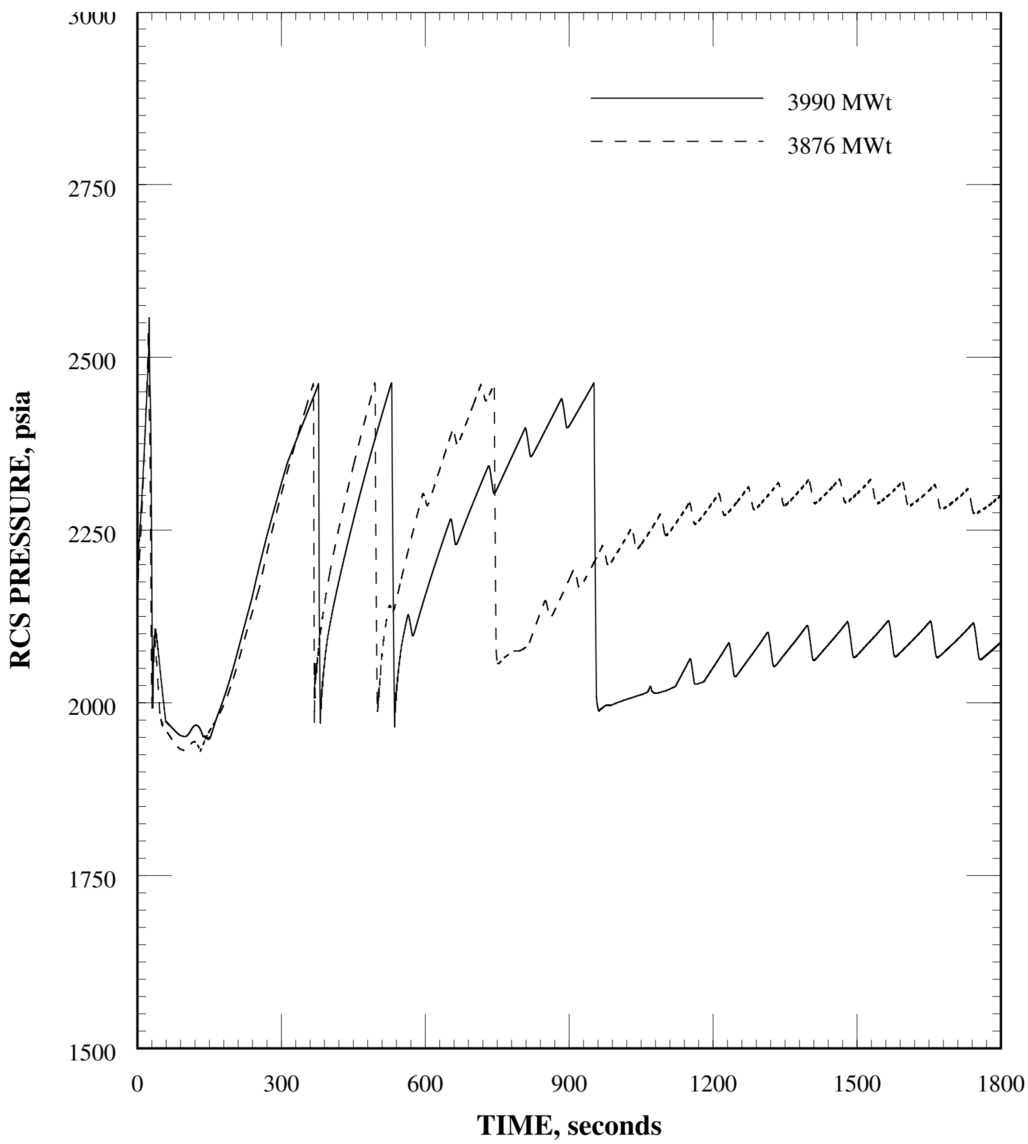


Figure 6.3-113
FWLB with LOP Long-Term Cooling Case - Pressurizer Pressure vs. Time

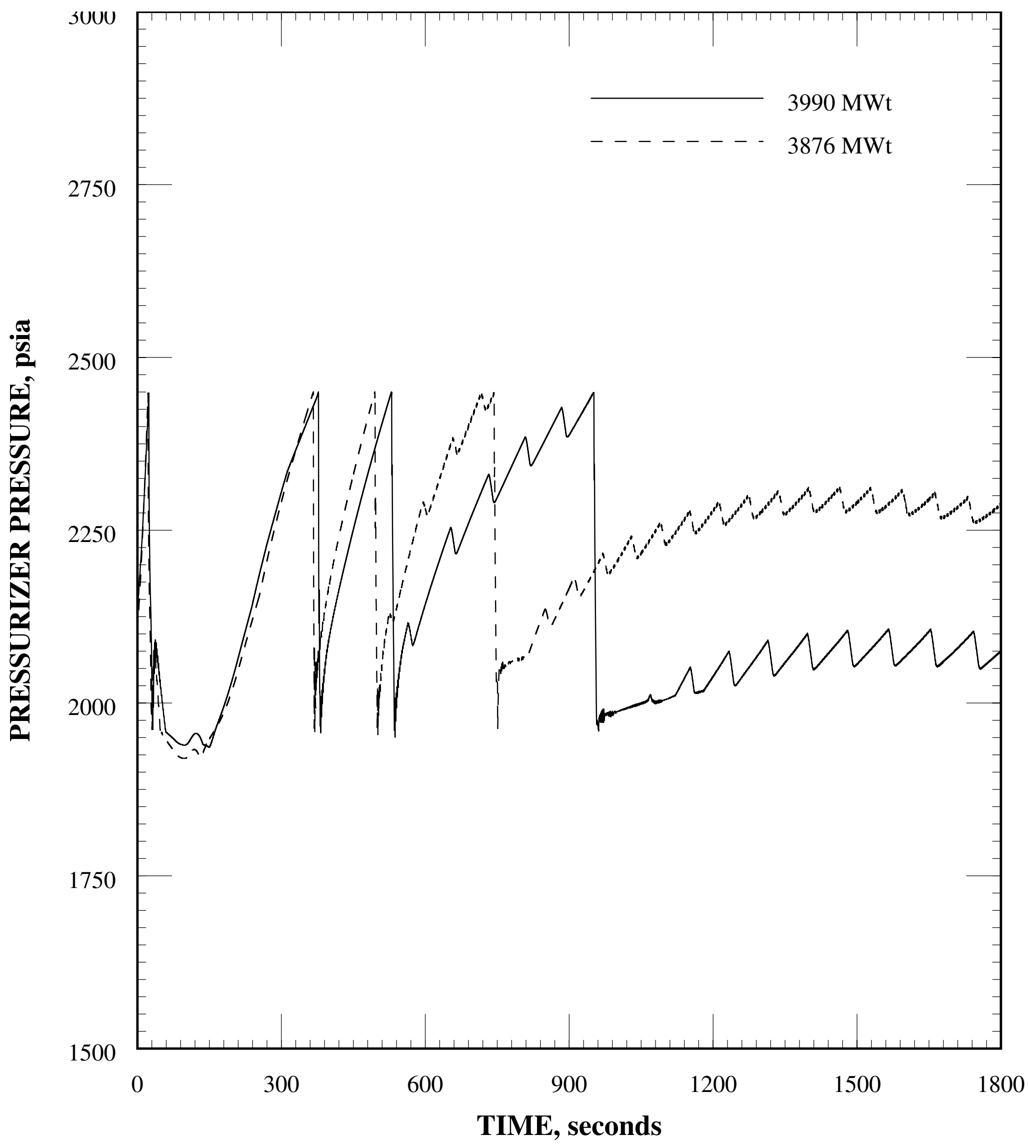


Figure 6.3-114
FWLB with LOP Long-Term Cooling Case - Pressurizer Water Volume vs. Time

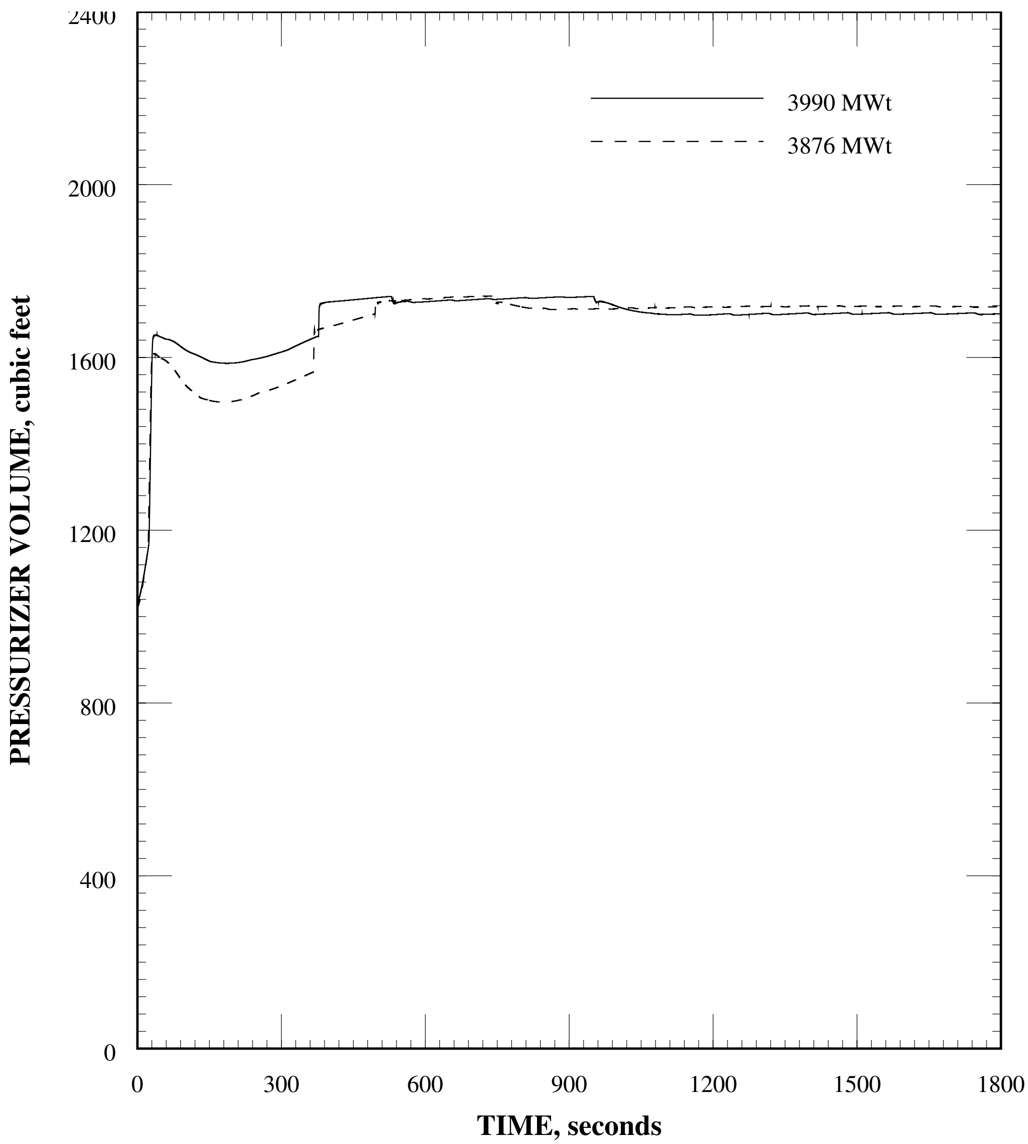


Figure 6.3-115
FWLB with LOP Long-Term Cooling Case - SG Pressures vs. Time

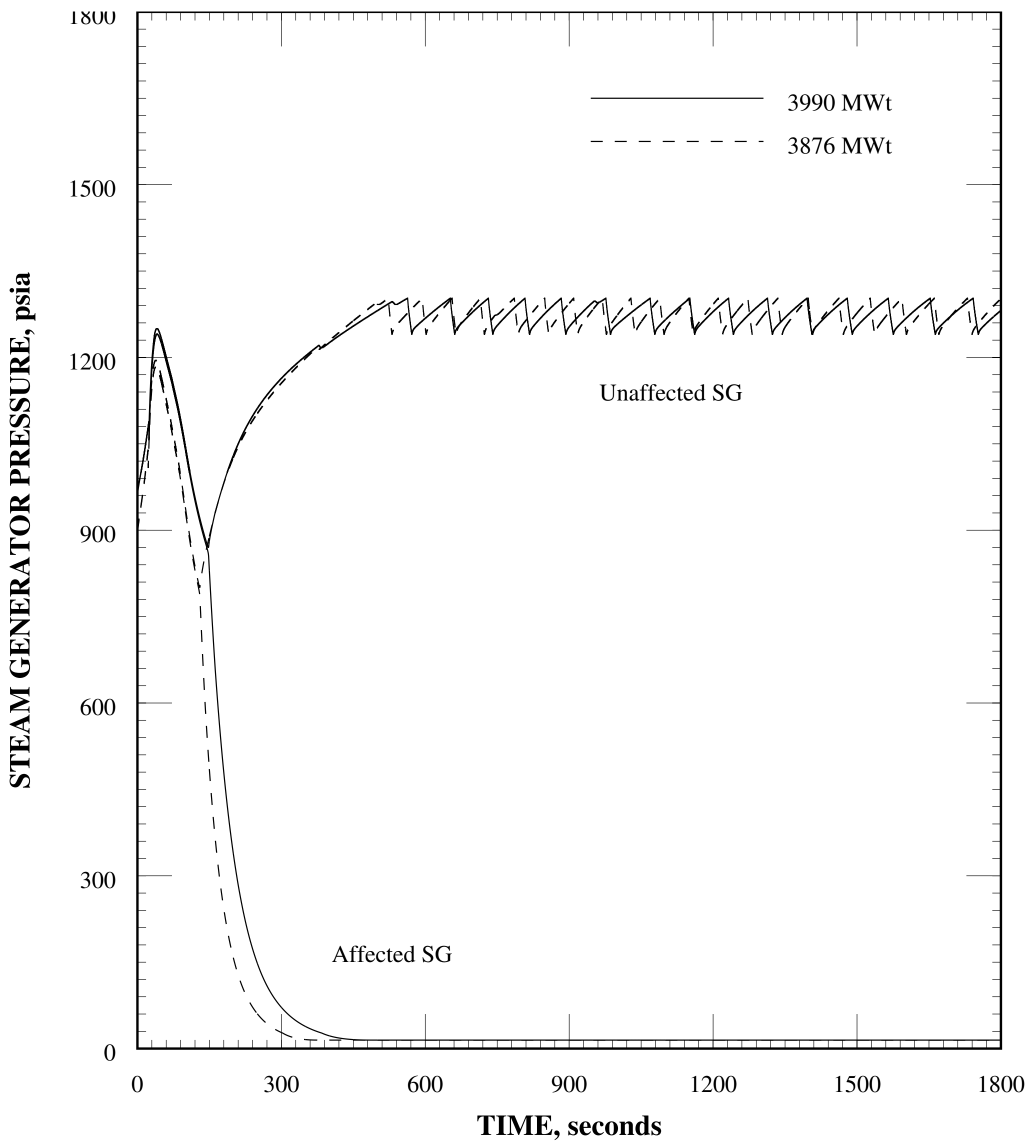


Figure 6.3-116
FWLB with LOP Long-Term Cooling Case - Unaffected SG Levels vs. Time

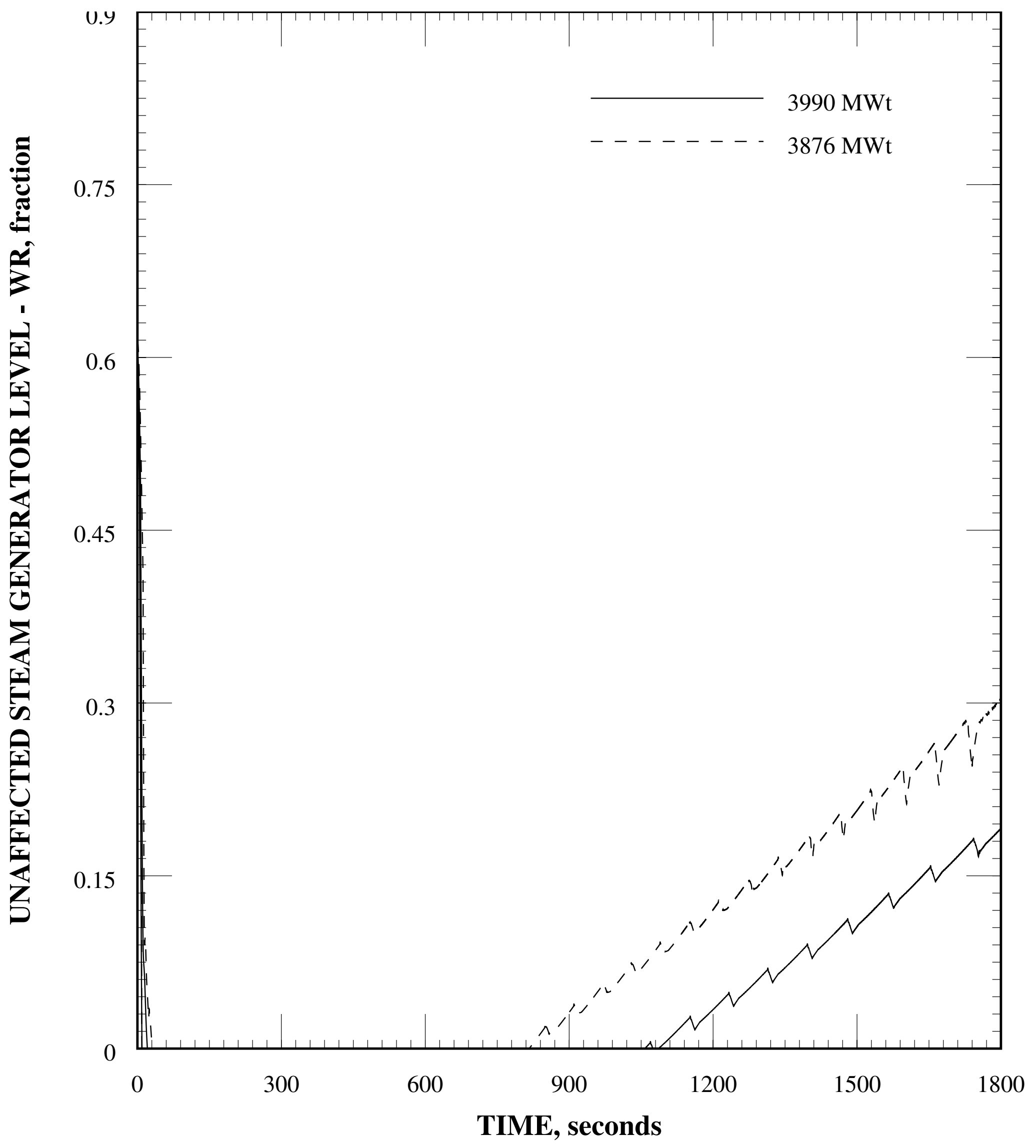


Figure 6.3-117
FWLB with LOP Long-Term Cooling Case - SG Liquid Inventories vs. Time

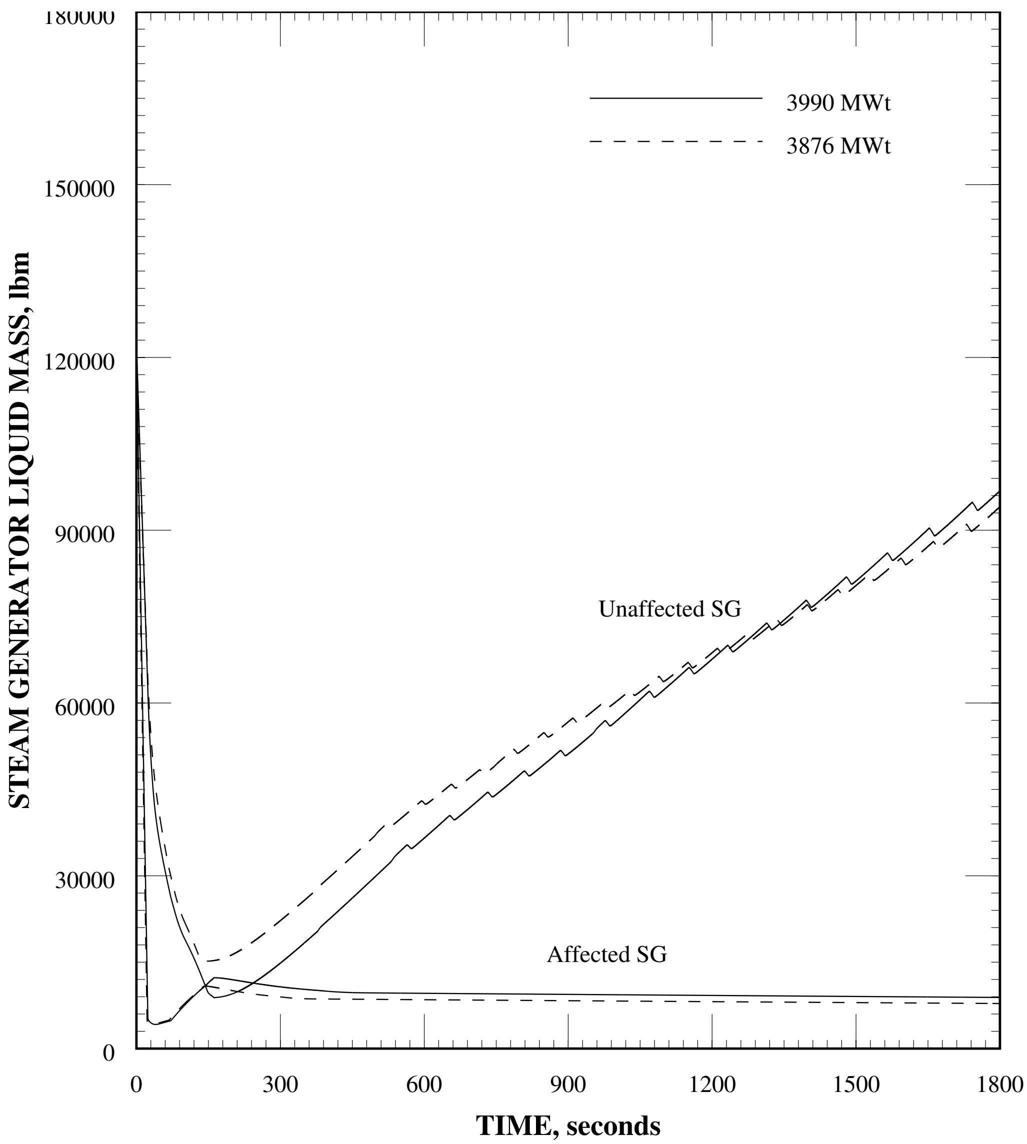


Figure 6.3-118
FWLB with LOP Long-Term Cooling Case - RCS Loop Flows vs. Time

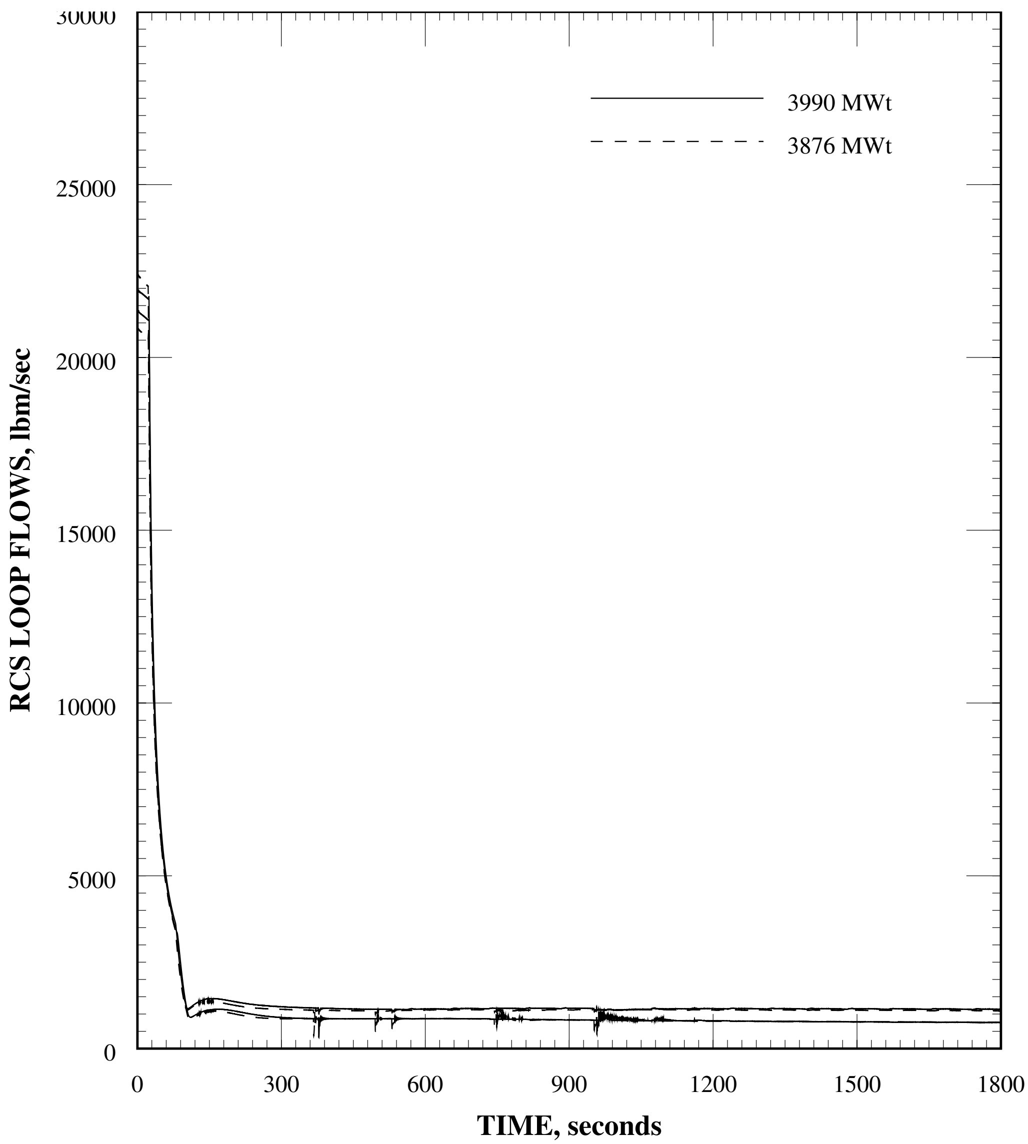


Figure 6.3-119
FWLB with LOP Long-Term Cooling Case - Affected SG AFW Flow vs. Time

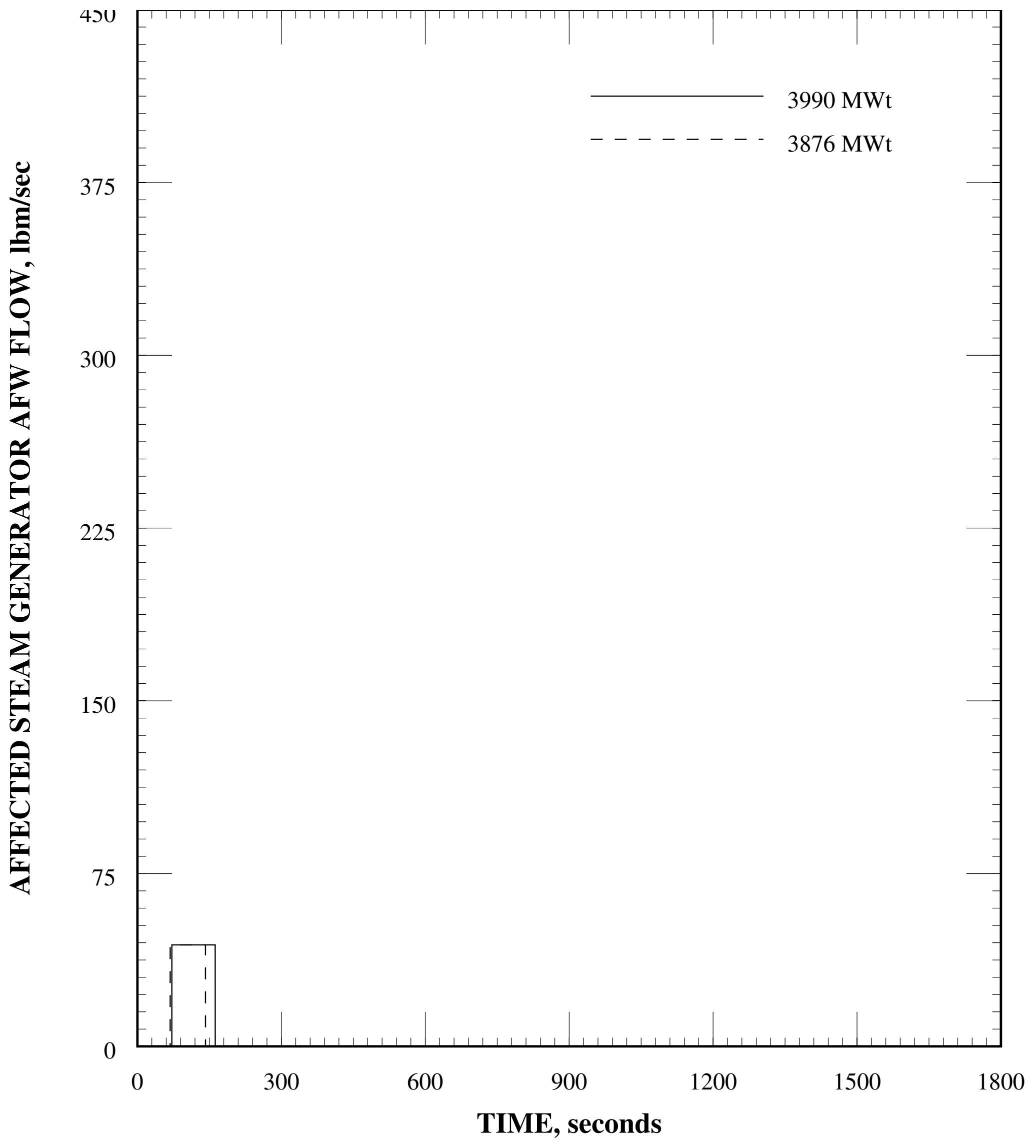


Figure 6.3-120
FWLB with LOP Long-Term Cooling Case - Unaffected SG AFW Flow vs. Time

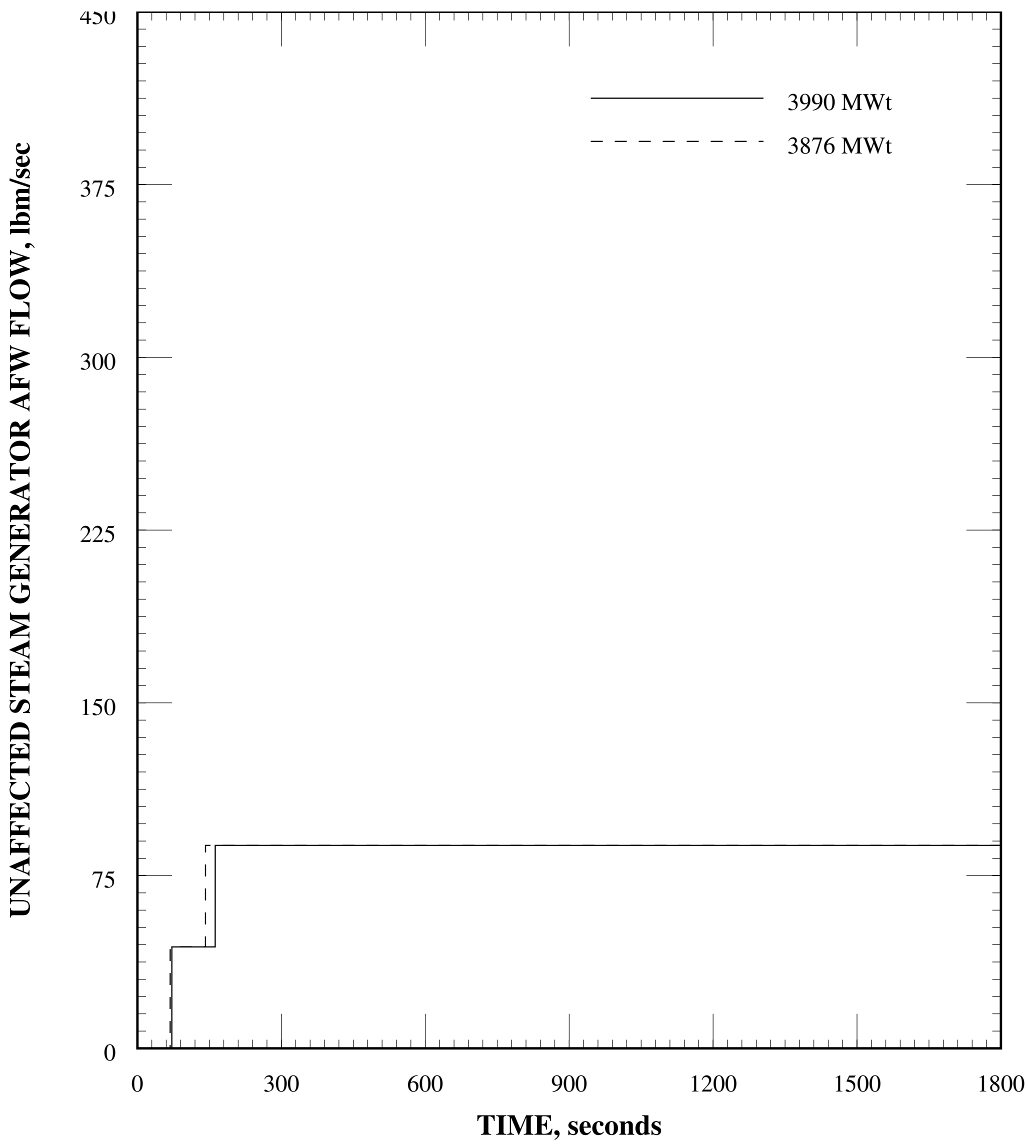


Figure 6.3-121
FWLB with LOP Long-Term Cooling Case - Break Flow vs. Time

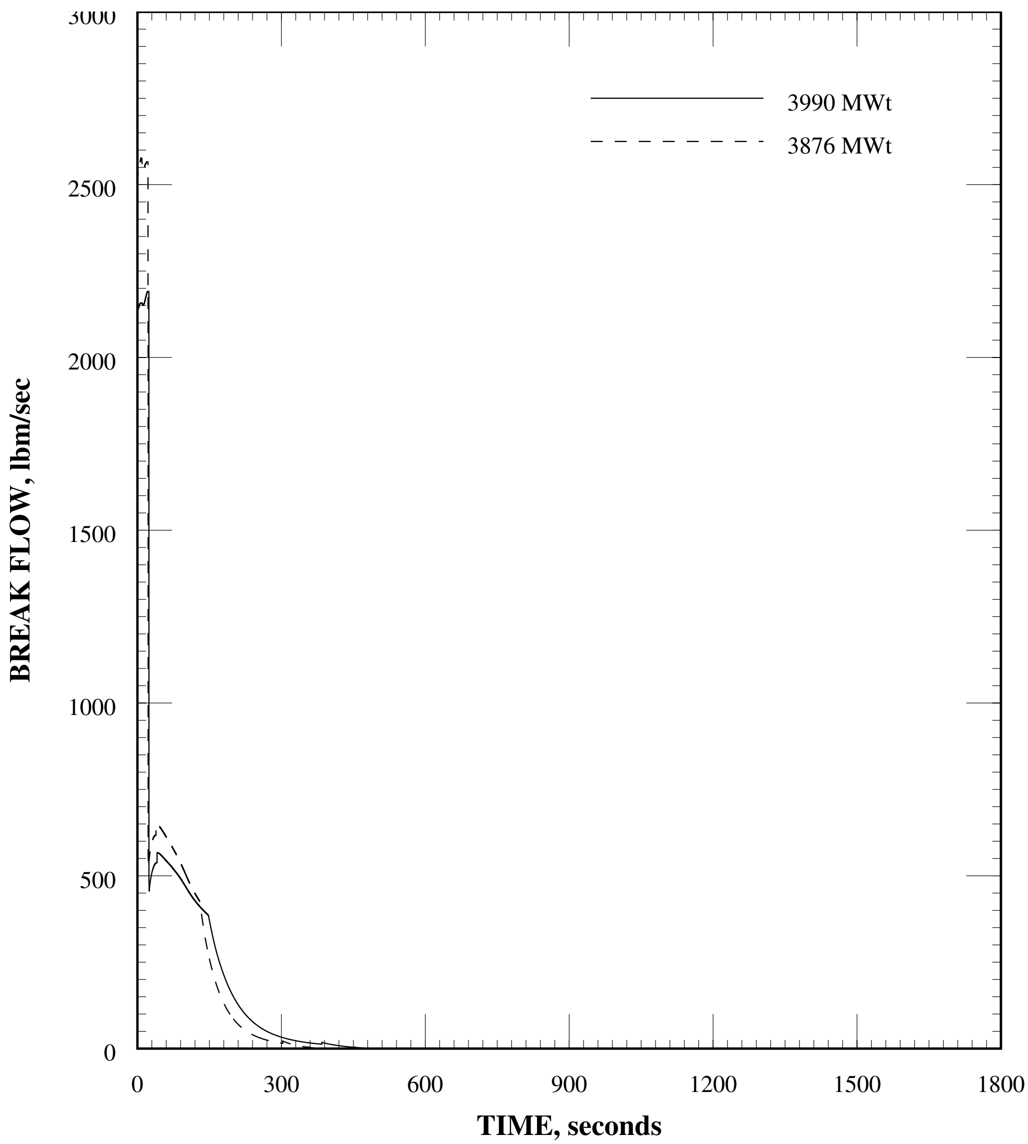
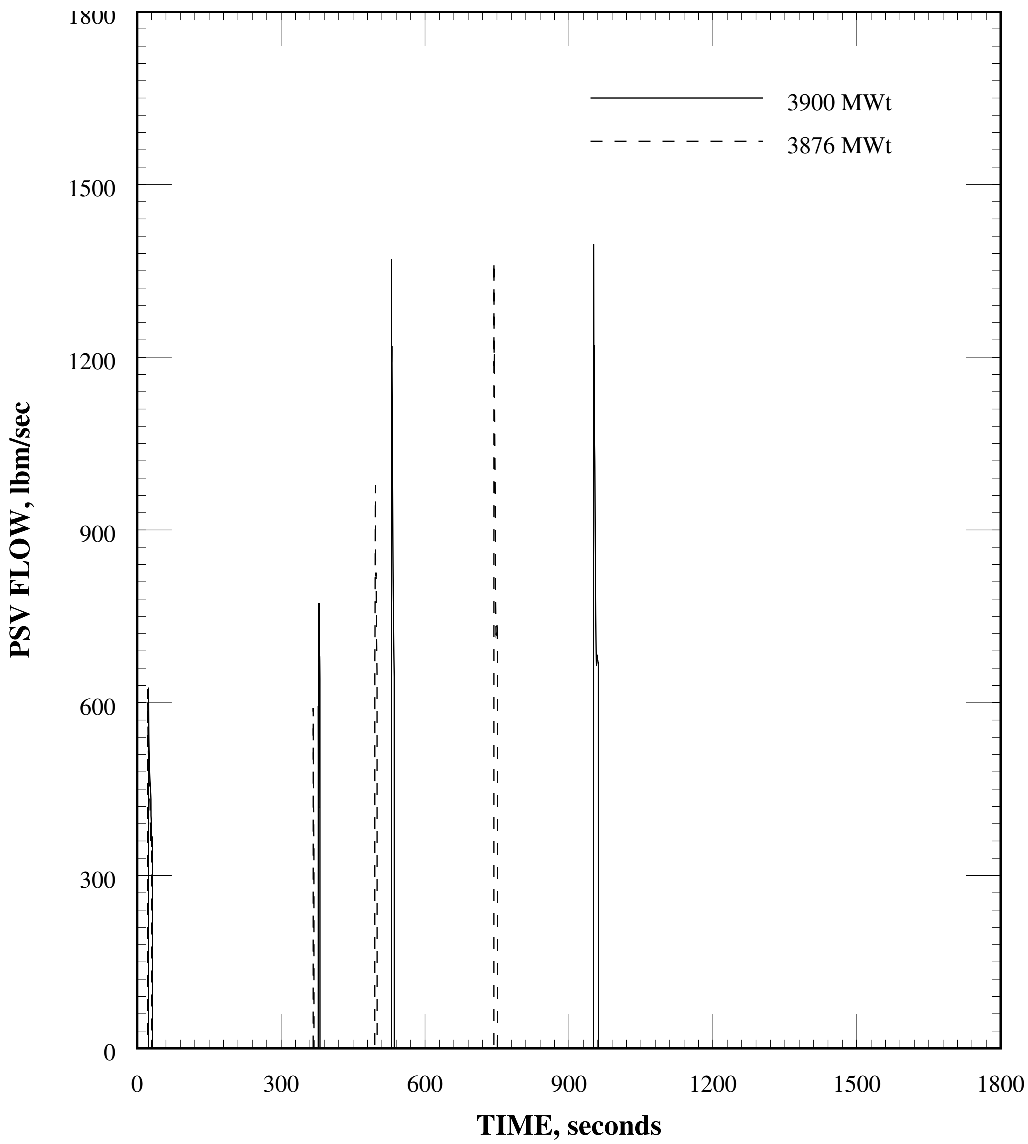


Figure 6.3-122
FWLB with LOP Long-Term Cooling Case - PSV Flow vs. Time



Section 6.3.2.8.2 Small Feedwater Line Break Event

Section 6.3.2.8.2.1 Identification of Event and Causes

As described in UFSAR Section 15.2.8, the small Feedwater Line Break (SFWLB) event is initiated by a break that is smaller than 0.2 ft^2. Assuming a break downstream of the check valves, inoperability of the FW system, and the low enthalpy liquid break discharge, the event can be described as follows:

The termination of the main FW or loss of subcooled FW flow to both SGs causes increasing SG temperatures and decreasing levels, thus decreasing heat removal by the secondary system. This results in 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 (if the break is inside containment) may occur much earlier than the HPPT making the consequences less adverse. RCS heatup may continue after the trip due to gradual decrease of heat transfer in the SGs. During the transient, opening of the MSSVs after turbine trip provides additional cooling by the secondary system, and eventually, decreasing core power following reactor trip reduces the heat load to the SGs. Following the turbine trip, fast transfer to offsite power provides power to the station. AFAS is actuated by low SG level in the affected SG, and AFW that is supplied to both SGs increasing cooldown of the RCS. Reduction in secondary system pressure causes a MSIS. The pressure difference between the SGs increases and the AFW is diverted to the unaffected SG due to AFW lockout on pressure difference, restoring the SG liquid level and long-term cooling of RCS.

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

Section 6.3.2.8.2.2 Acceptance Criteria

SFWLB with offsite power available is a “limiting fault” event that results in an unplanned decrease in secondary system heat removal. As defined in 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 6.3.2.8.2.3 Description of Analysis

The NSSS response to a SFWLB (break size less than 0.2 ft^2) with offsite power available is simulated using the CENTS code.

Maximum primary pressure case is analyzed for both 3990 MW_t and 3876 MW_t rated power units. The input parameters and initial conditions are selected to maximize the RCS pressure, and demonstrate that the peak RCS pressure remains within 110% of the design pressure.

Long-term cooling case is not analyzed for SFWLB. The criterion for long-term cooling capability is the same as for FWLB with LOP resulting from turbine trip and a single failure. This event is analyzed in Section 6.3.2.8.1.3 is more limiting in terms of long-term cooling capability due to more conservative assumptions on heat transfer.

Section 6.3.2.8.2.3.1 Transient Simulation

The system is initialized at 102% power and 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 to the SFWLB (affected SG), 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 relative to the fluid conditions within the affected SG, it is assumed that saturated liquid is discharged until no liquid remains, at which time saturated steam is discharged.

A total LOFW is simulated by a rapid ramp down of main FW flow to zero (in 0.1 seconds). The total LOFW and discharge from the break yields to a reduction of the SG water inventory, pressurization of the secondary side, and a resulting heatup and pressurization of the primary side.

A trip condition on a low water level in the affected SG is assumed when the liquid inventory decreases below a certain mass. In this event, low-level trip is anticipated to occur at a liquid inventory in excess of 90,000 lb_m. In this analysis, a trip setpoint corresponding to an inventory that is less than half of that value is used for conservatism. This conservative delay results in delayed 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 resulting from decreasing heat transfer area and fluid conditions. This reduction of heat transfer is modeled by a steeper degradation than that expected, linearly decreasing to total loss of heat transfer when the affected SG empties of liquid.

Reactor trip occurs on HPPT or LSGLT. The most adverse condition is when HPPT occurs at the same time as low-level trip. Therefore, the initial conditions are selected to result in coinciding HPPT with the low level trip on liquid level in the affected SG. AFAS is generated on low SG level in the affected SG. An AFAS is assumed to occur at 10% WR level in the unaffected SG.

The most limiting single failure is failure of one of the Fast Transfer Busses (FFBTs). Depending on the initial plant configuration, FFBT following a turbine trip may result in either a two RCP coastdown or a four RCP coastdown. Both flow coastdown cases were evaluated for their impact on primary pressure. The limiting scenario is FFBT following turbine trip resulting in a two RCP coastdown.

The maximum primary pressure transient is simulated until the primary and secondary pressures and temperatures are decreasing and stabilizing.

Section 6.3.2.8.2.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-31 contains the initial conditions used for the peak primary event. 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 was used, i.e., discharge of saturated liquid until SG is dry.
3. A reactor trip from a low water level signal within the affected SG is credited and conservatively delayed.
4. A conservative degradation of primary-to-secondary heat transfer is assumed.
5. Key parameters are initialized such that a reactor trip occurs from a high pressurizer pressure signal occurring simultaneously with a trip on low SG level.
6. There is no operator action for the first 30 minutes of the event.

Table 6.3-31
Parameters Used for SFWLB RCS Peak Pressure Event

Parameter	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	562	566
Initial pressurizer pressure (psia)	2212	2286
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	11.4	11.4
Initial SG level (ft)	31.2	32.8
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)	16	10
PSV tolerance	+3%	+3%
PSV blowdown	5%	5%
MSSV tolerance	+3%	+3%
MSSV blowdown	5%	5%
Single failure	FFBT	FFBT
LOP	no	no
FWLB area (ft^2)	0.20	0.20

Section 6.3.2.8.2.5 Results

Table 6.3-32 presents a sequence of events that occur following the SFWLB event until operator action is initiated for the primary peak pressure case. This sequence of events is representative of the reference cycles for 3990 MW_t, and 3876 MW_t units.

The representative behaviors of NSSS parameters are presented in Figure 6.3-123 to Figure 6.3-138 showing the response for the SFWLB event under most adverse initial and transient conditions that maximize the RCS peak pressure.

The reduction of primary-to-secondary heat transfer caused by decrease in 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 low-level trip in the affected SG. A turbine trip followed by FFBT failure causes further increases in pressure and temperature in both primary and secondary systems and degradation of RCS flow. The PSVs and MSSVs open providing cooldown and maintaining primary pressure well below 110% of the design value. MSSVs provide adequate pressure relief so the main steam system pressure is limited to opening pressures of the MSSVs. Thus, the secondary system pressure remains well below 110% of the design pressure.

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.

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

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	FWLB and complete LOFW to both SGs, break size (ft^2)	0.20	0.20
18.54	18.14	Pressurizer pressure reaches trip setpoint (psia)	2450	2450
18.54	18.14	HPPT signal generated	---	---
19.04	18.64	Reactor trip breakers open	---	---
19.04	18.64	Turbine trip occurs	---	---
19.04	18.64	FFBT occurs	---	---
19.64	19.24	Scram CEAs begin falling	---	---
21.25	20.81	PSVs open (psia)	2550	2550
21.75	21.38	Maximum RCS pressure (psia)	2681	2702
22.55	21.04	MSSVs bank 1 open on unaffected SG (psia)	1303	1303
22.57	21.04	MSSVs bank 1 open on affected SG (psia)	1303	1303
23.22	23.06	PSVs close (psia)	2422	2422
24.63	22.70	MSSVs bank 2 open on unaffected SG (psia)	1344	1344
24.84	22.80	MSSVs bank 2 open on affected SG (psia)	1344	1344
25.91	25.81	Peak secondary pressure occurs	1354	1368
27.35	26.46	Dryout of affected SG; AFAS generated in affected SG	<5000	<5000
27.93	---	PSVs open (psia)	2550	2550
28.84	---	PSVs close (psia)	2422	2422
29.70	29.24	Maximum liquid volume of pressurizer (ft^3)	770	703
31.06	32.24	MSSVs bank 2 close on affected SG (psia)	1277	1277
31.48	32.73	MSSVs bank 2 close on unaffected SG (psia)	1277	1277
38.25	40.20	MSSVs bank 1 close on affected SG (psia)	1238	1238
40.69	42.87	MSSVs bank 1 close on unaffected SG (psia)	1238	1238
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the long-term cooling FWLB event	---	---
1800	1800	Operator initiates the cooldown (min)	30	30

Section 6.3.2.8.2.6 Conclusions

For the SFWLB event, all acceptance criteria are met for a spectrum of break sizes less than 0.2 ft^2. The peak primary pressure remains below 110% of design at all times, thus ensuring the integrity of the RCS. The secondary system pressure remains below 110% of design pressure thus ensuring integrity of the main steam system. The minimum DNBR remains above the safety analysis fuel design limit thus ensuring fuel cladding integrity. AFW capacity is adequate to provide removal of the core decay heat until an operator action is taken 30 minutes after the initiating event.

Figure 6.3-123
SFWLB Primary Peak Pressure Case - Core Power vs. Time

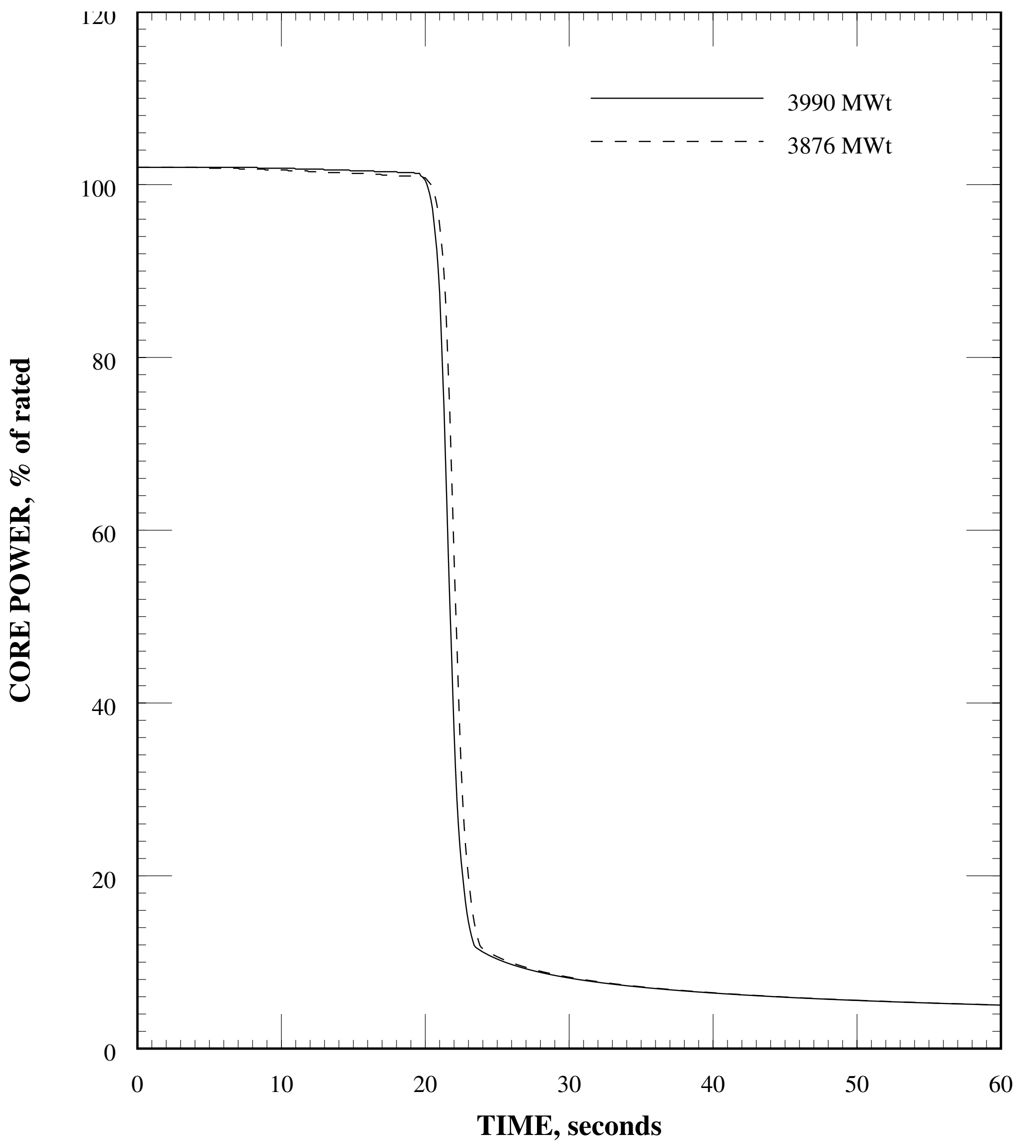


Figure 6.3-124
SFWLB Primary Peak Pressure Case - Core Heat Flux vs. Time

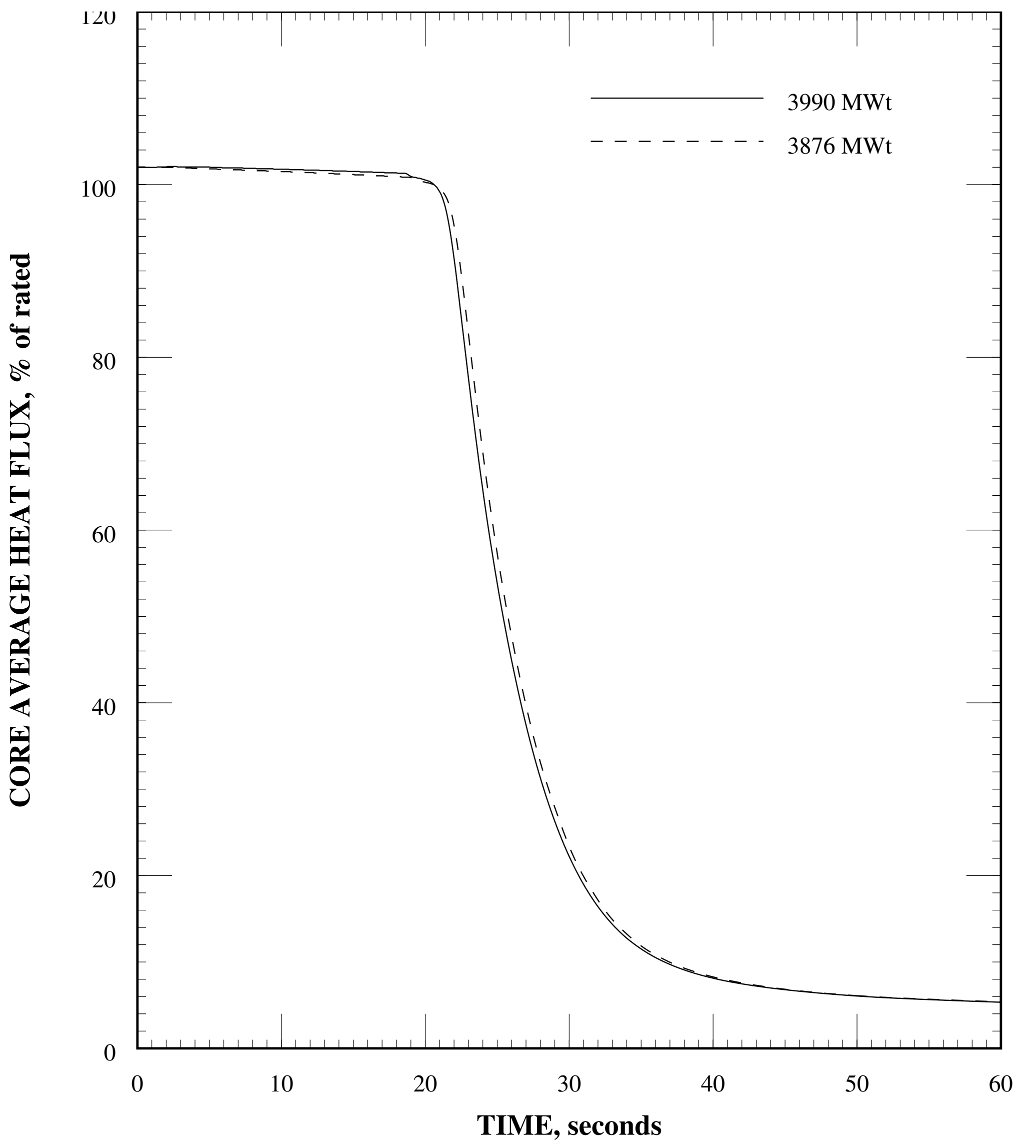


Figure 6.3-125
SFWLB Primary Peak Pressure Case - Reactivity vs. Time

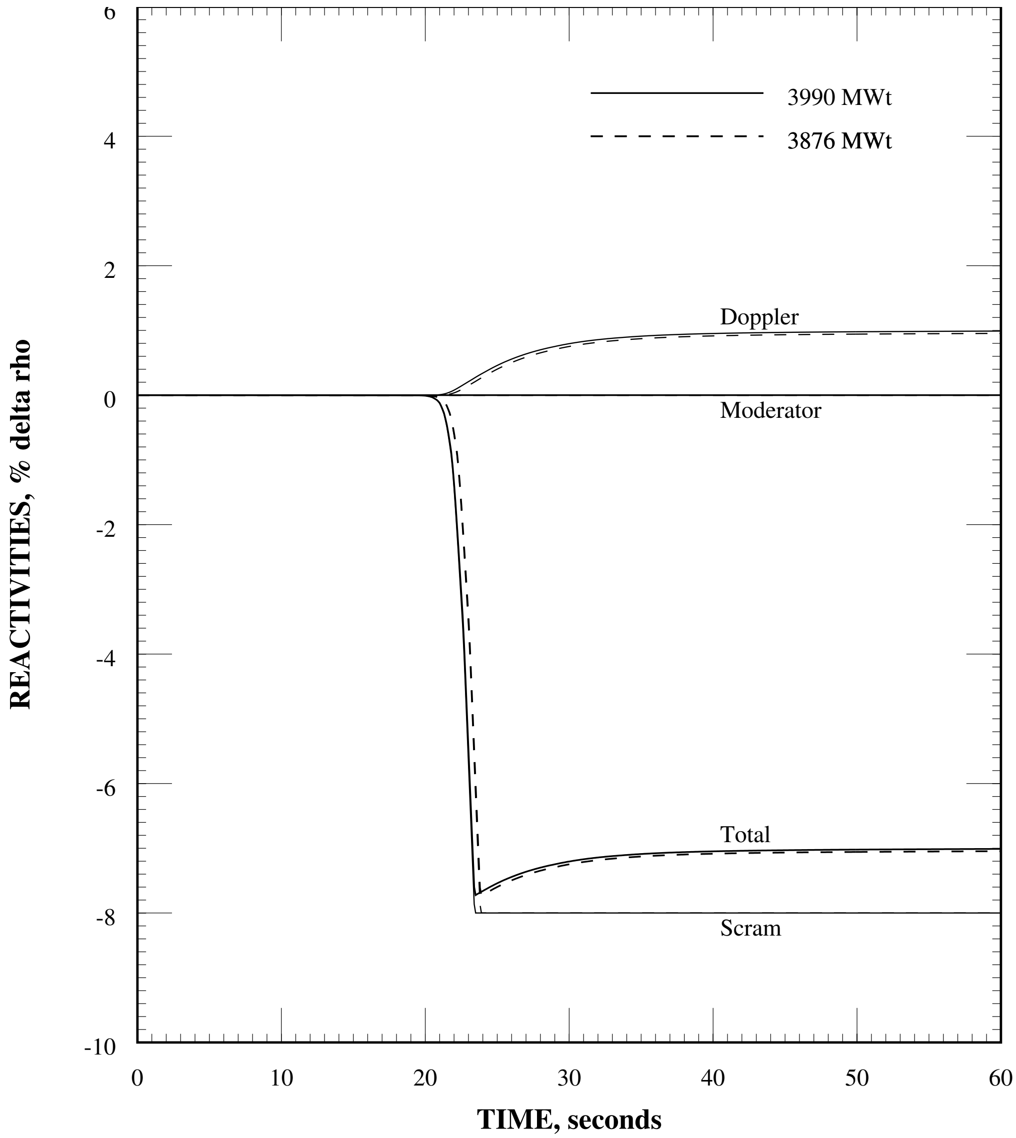


Figure 6.3-126
SFWLB Primary Peak Pressure Case – Affected Loop RCS Temperatures vs. Time

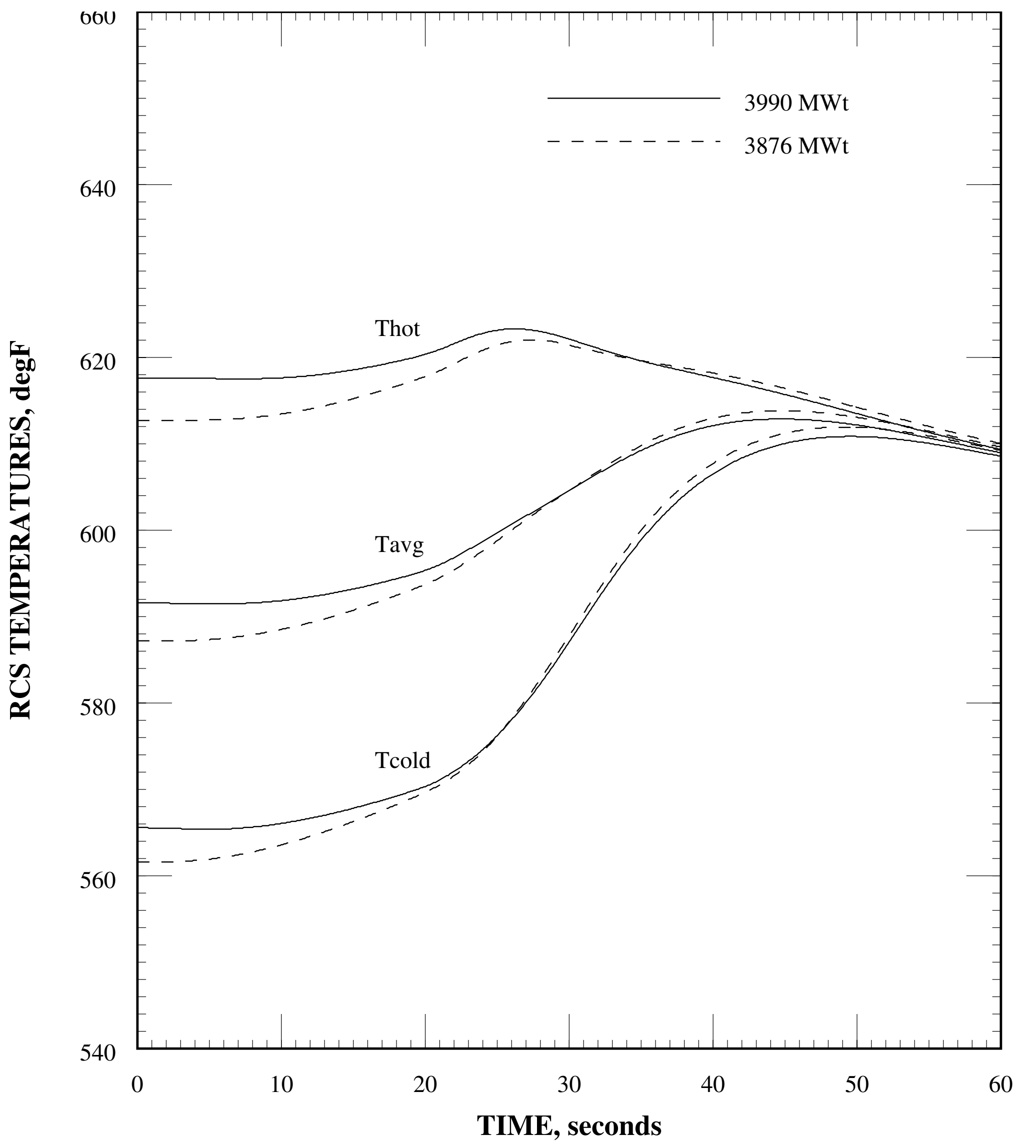


Figure 6.3-127
SFWLB Primary Peak Pressure Case - RCS Pressure vs. Time

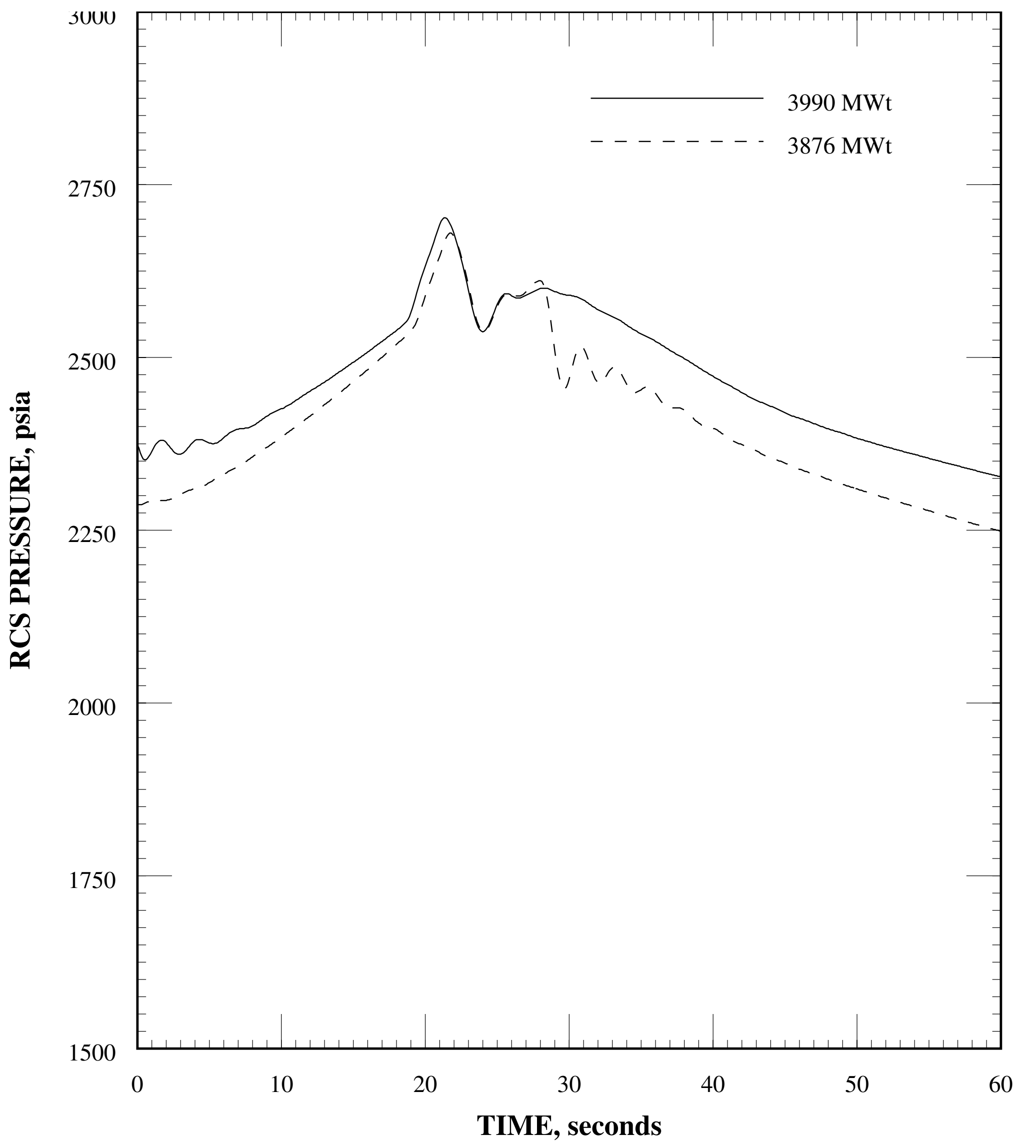


Figure 6.3-128
SFWLB Primary Peak Pressure Case - Pressurizer Pressure vs. Time

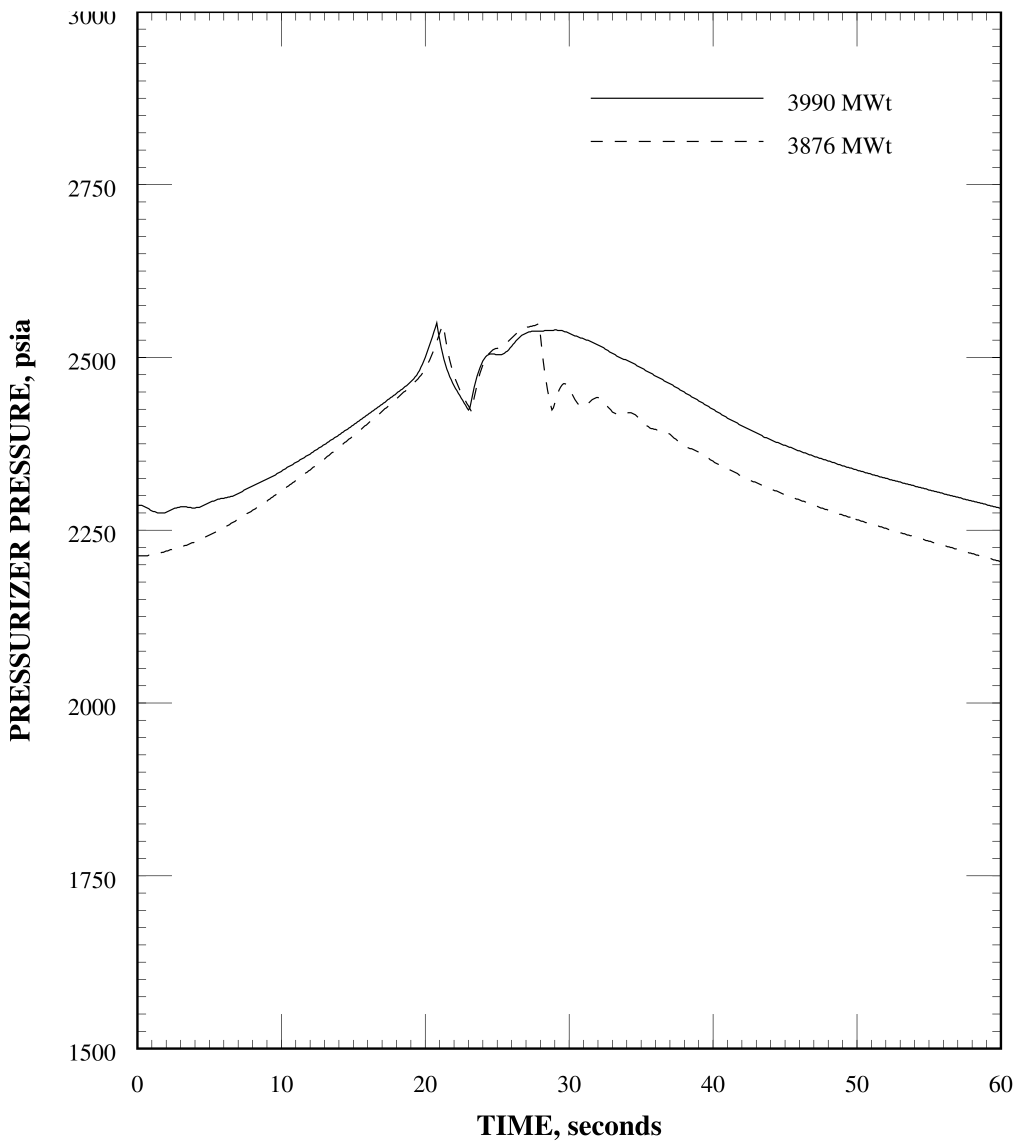


Figure 6.3-129
SFWLB Primary Peak Pressure Case - Pressurizer Water Volume vs. Time

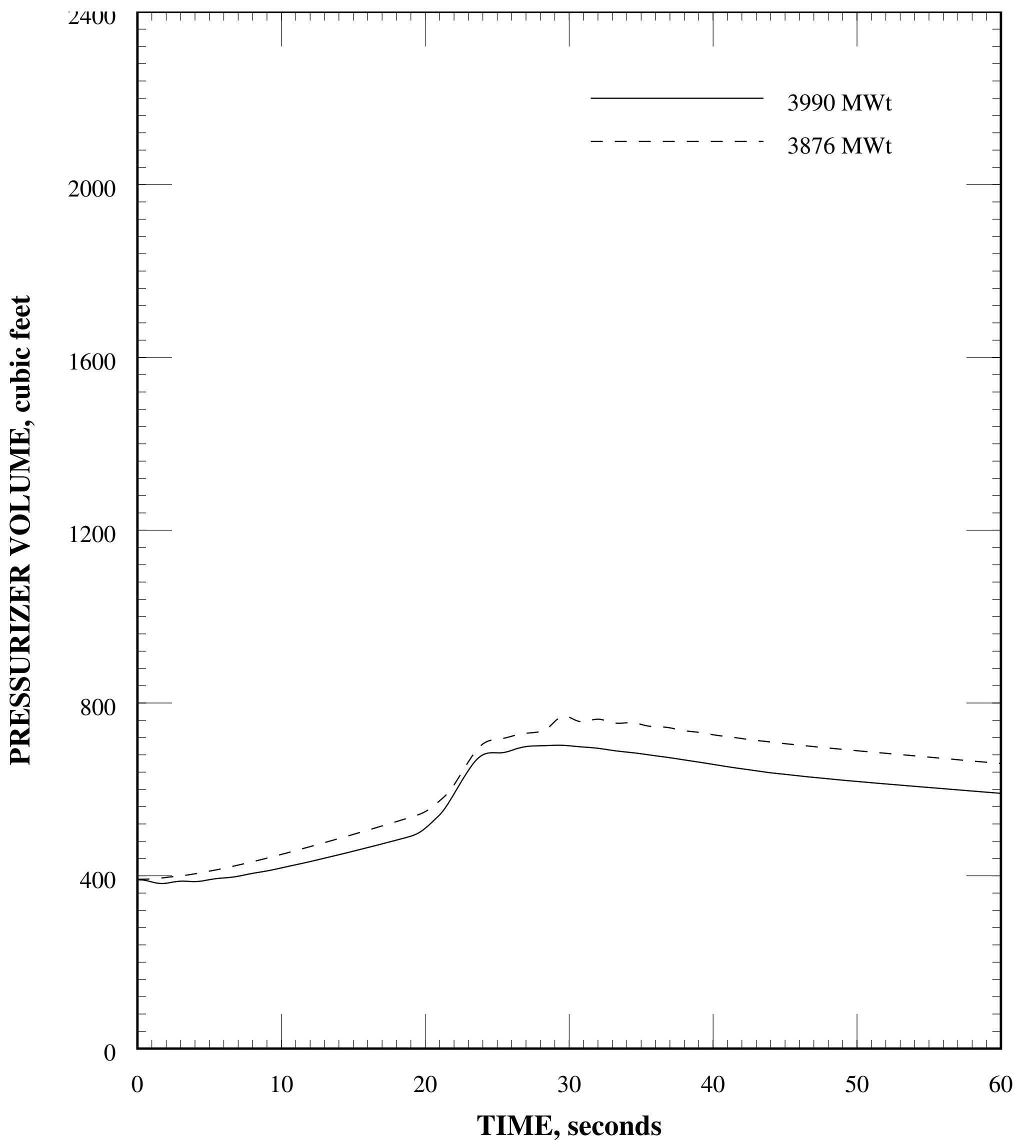


Figure 6.3-130
SFWLB Primary Peak Pressure Case - SG Pressures vs. Time

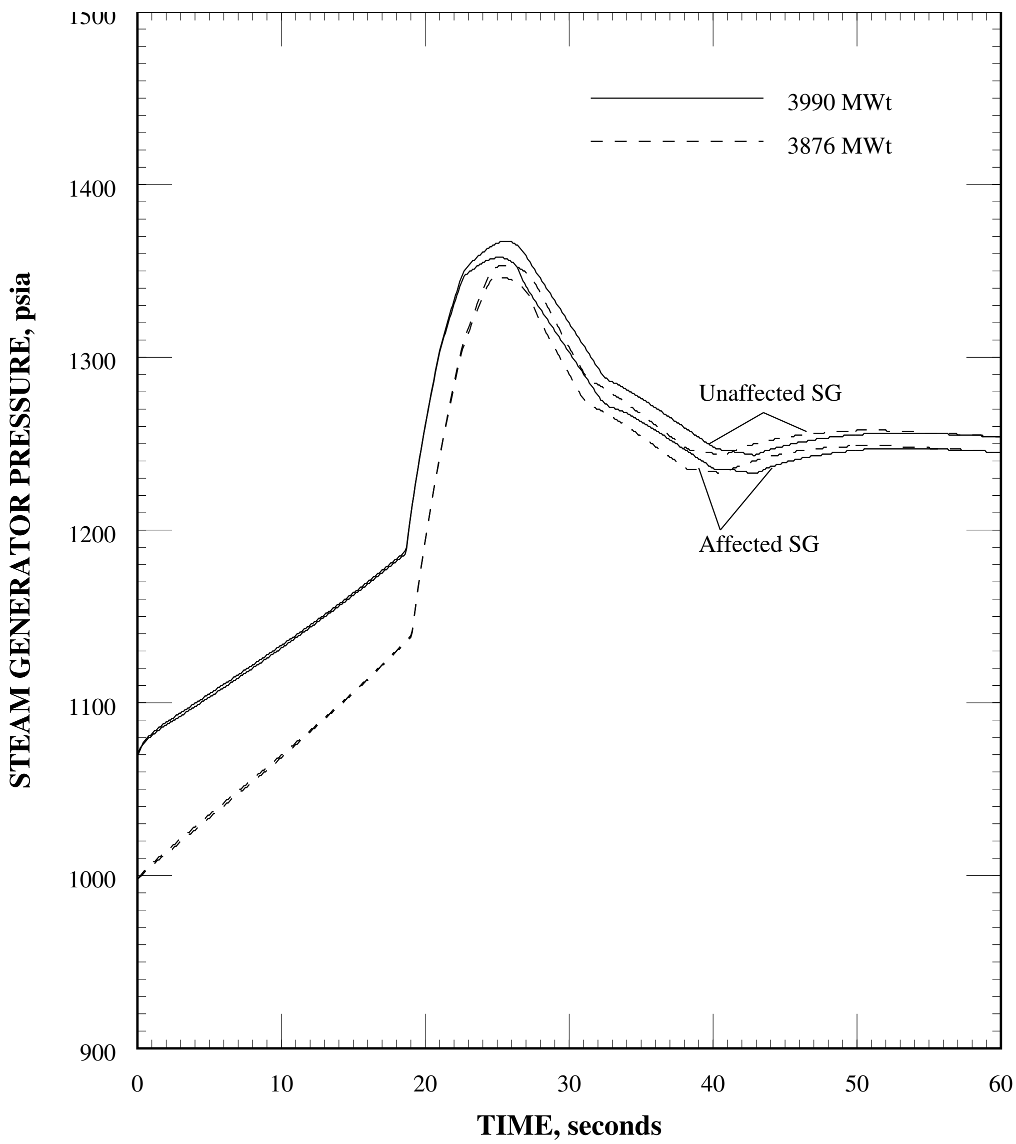


Figure 6.3-131
SFWLB Primary Peak Pressure Case - SG Levels vs. Time

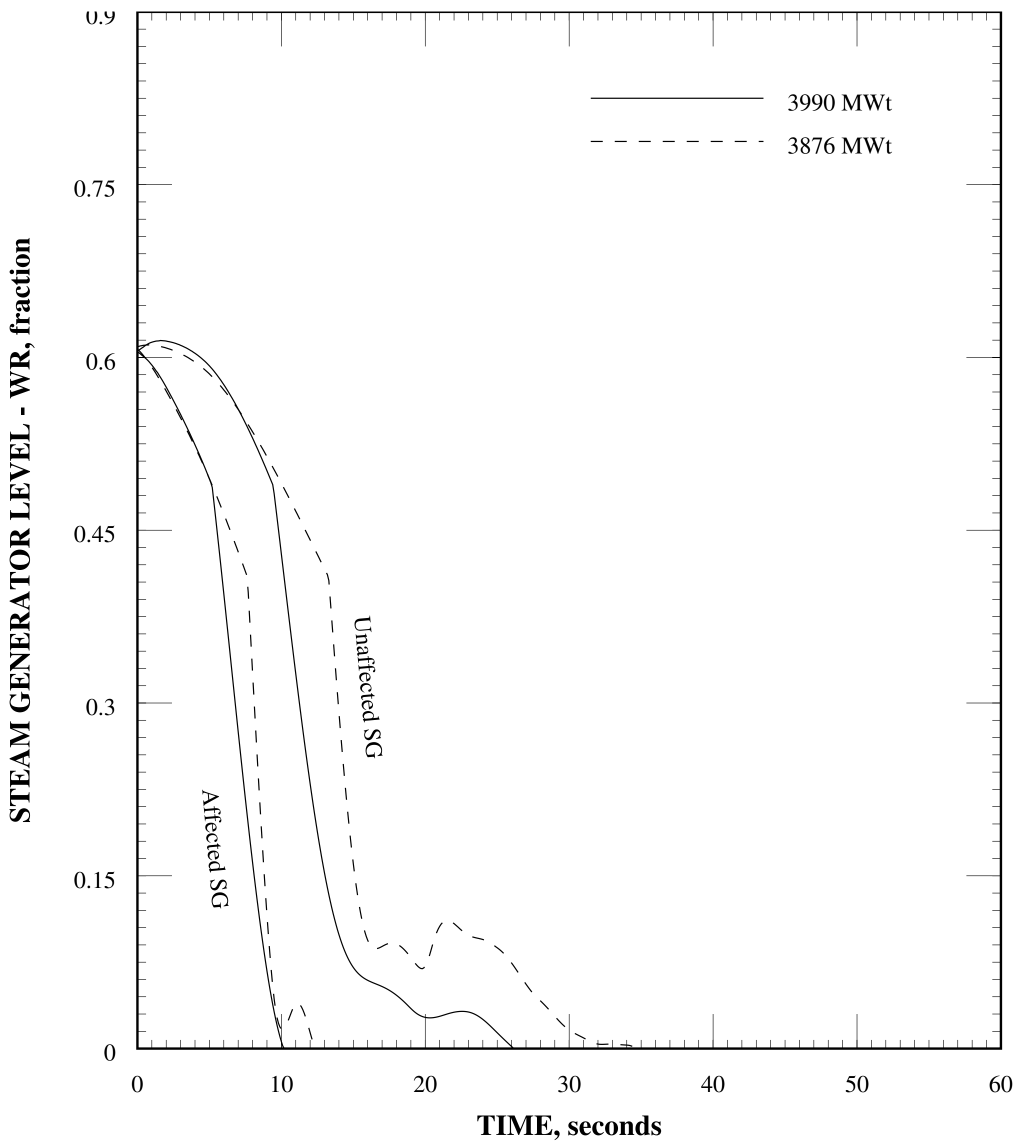


Figure 6.3-132
SFWLB Primary Peak Pressure Case - SG Liquid Inventories vs. Time

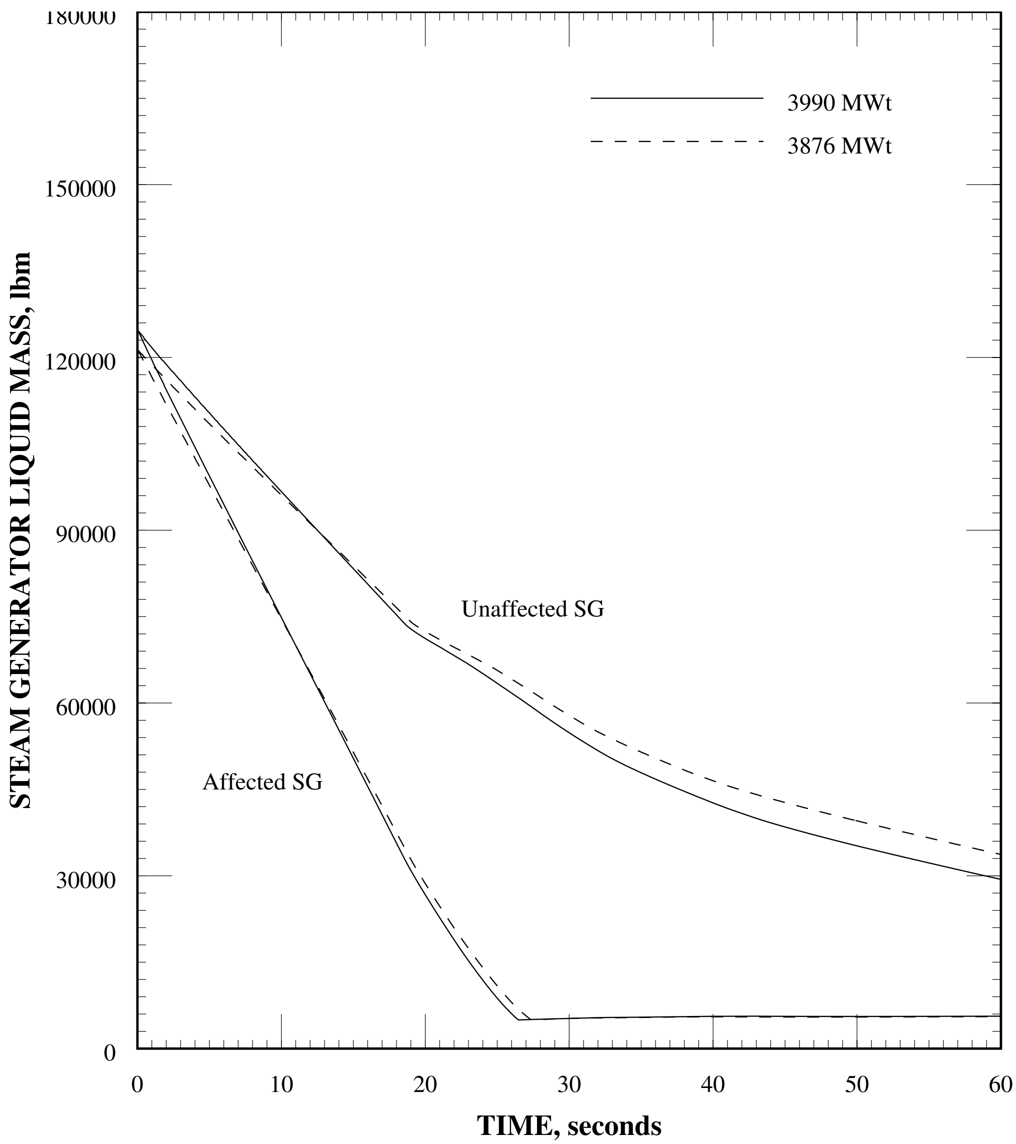


Figure 6.3-133
SFWLB Primary Peak Pressure Case - RCS Loop Flows vs. Time

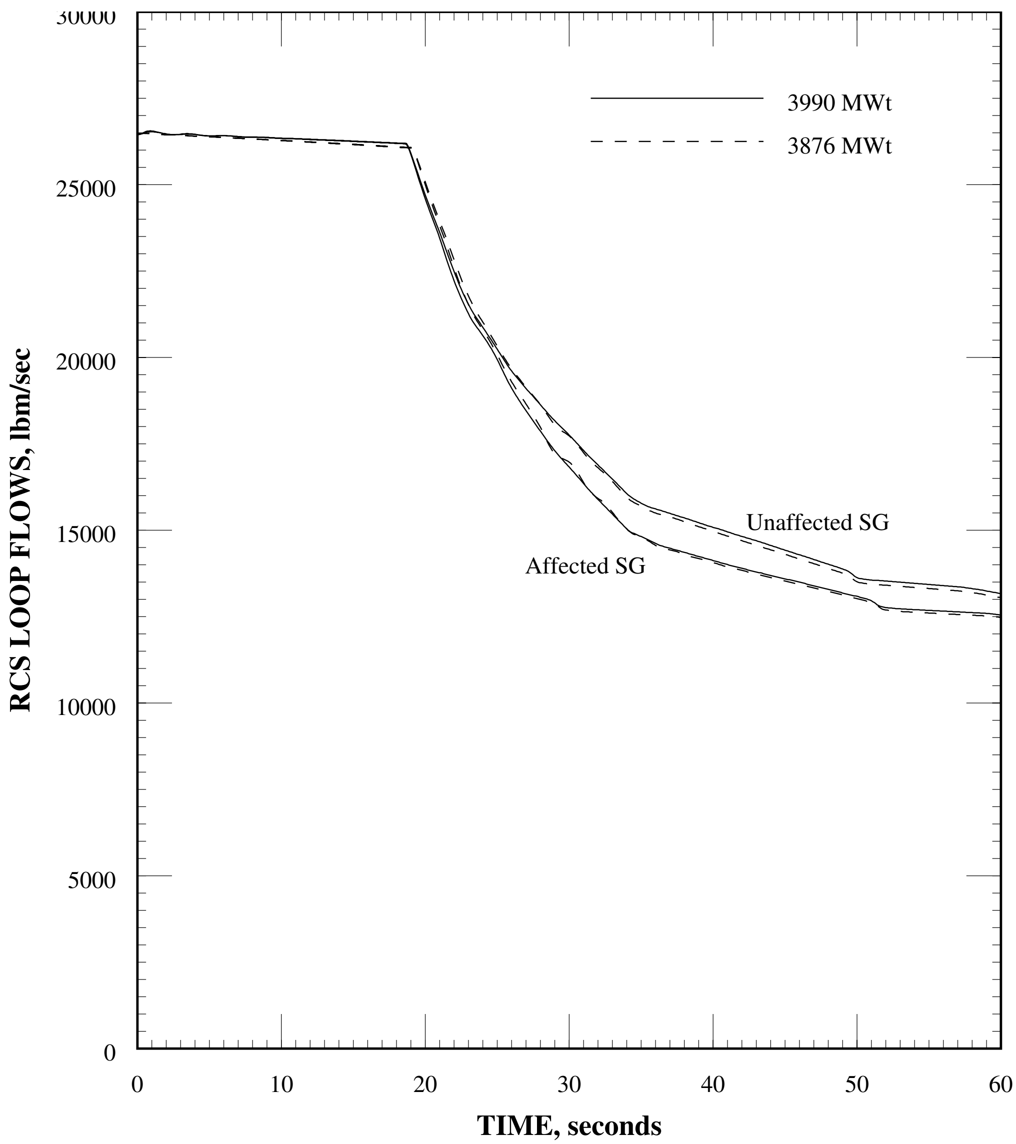


Figure 6.3-134
SFWLB Primary Peak Pressure Case - SG Steam Flows vs. Time

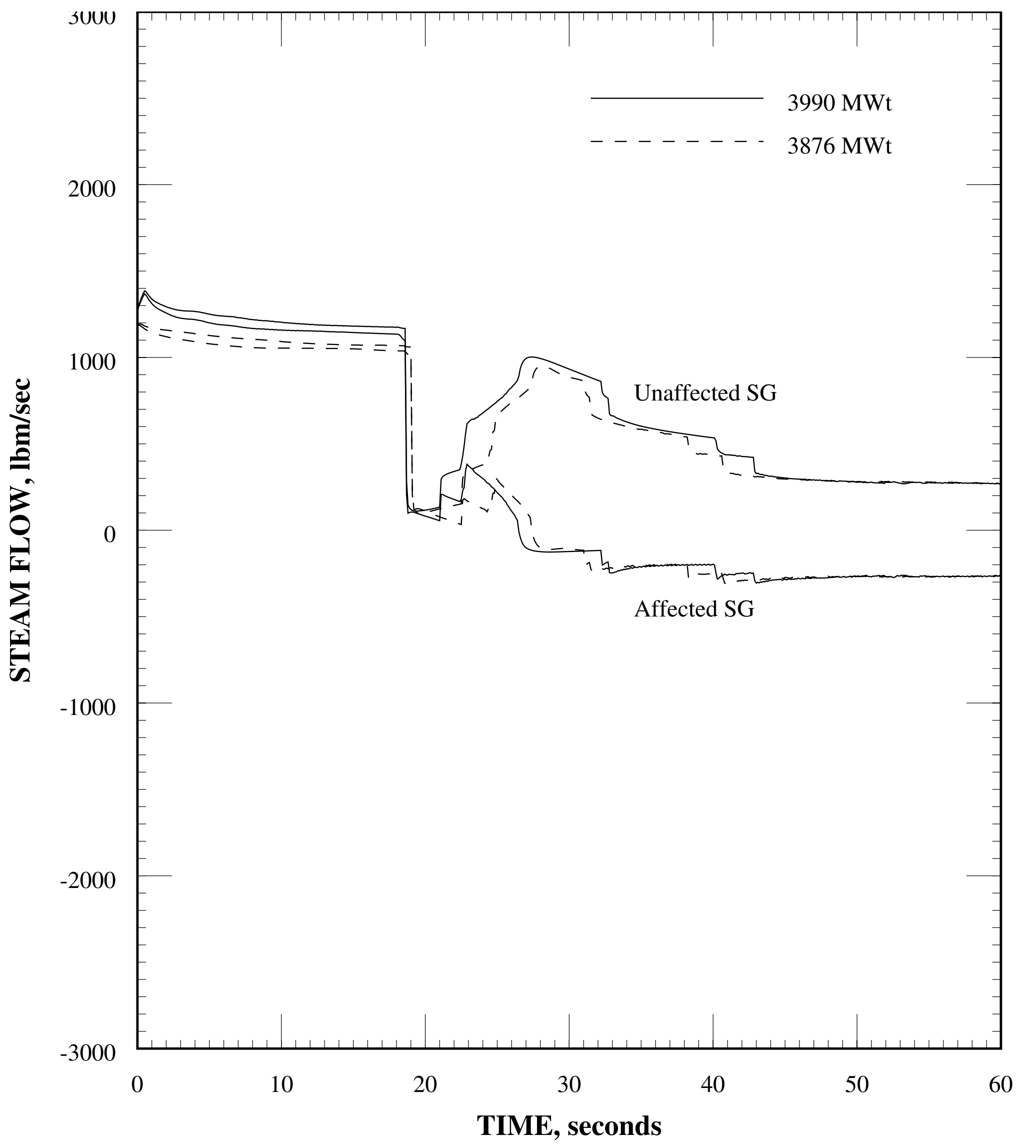


Figure 6.3-135
SFWLB Primary Peak Pressure Case - Break Flow vs. Time

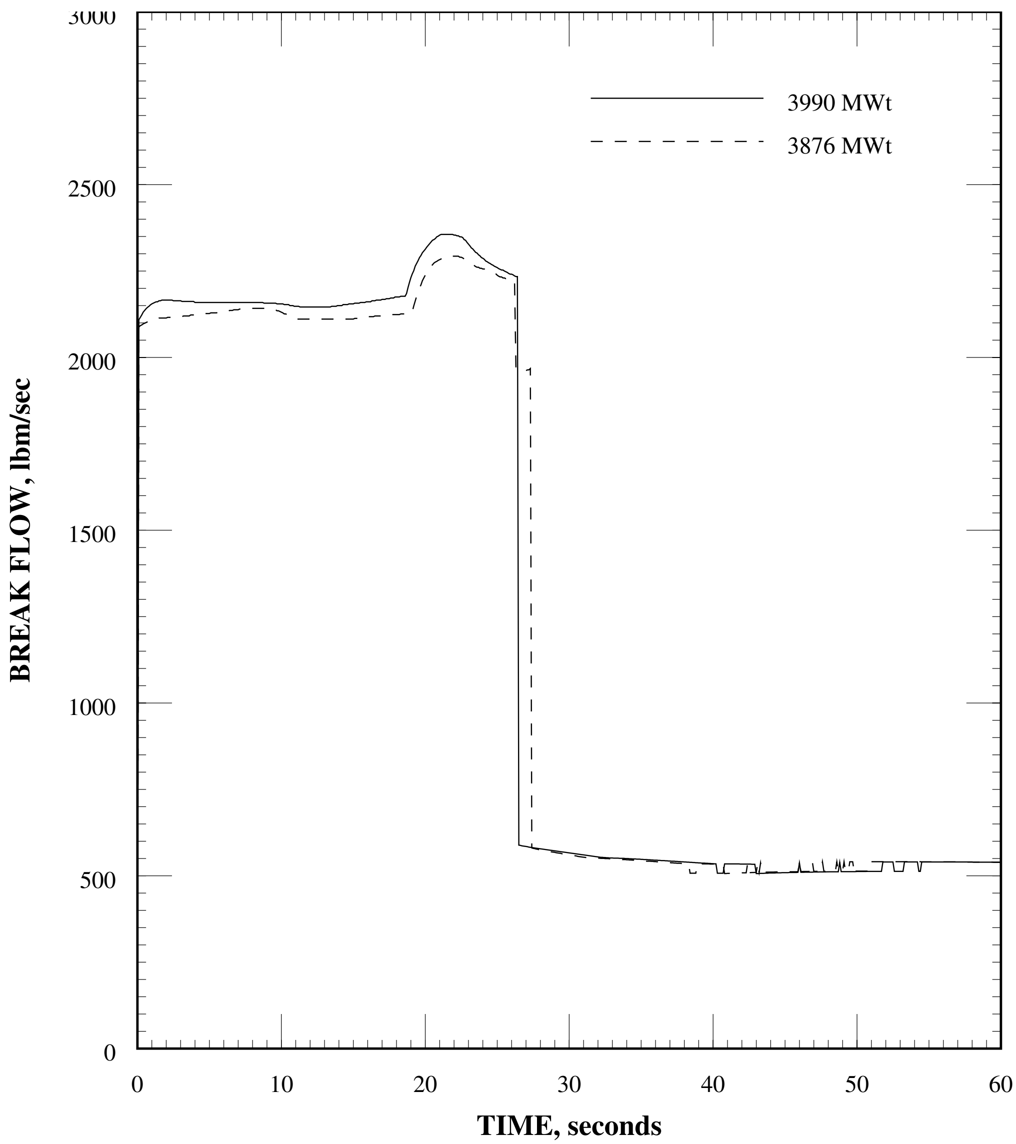


Figure 6.3-136
SFWLB Primary Peak Pressure Case - Break Enthalpy vs. Time

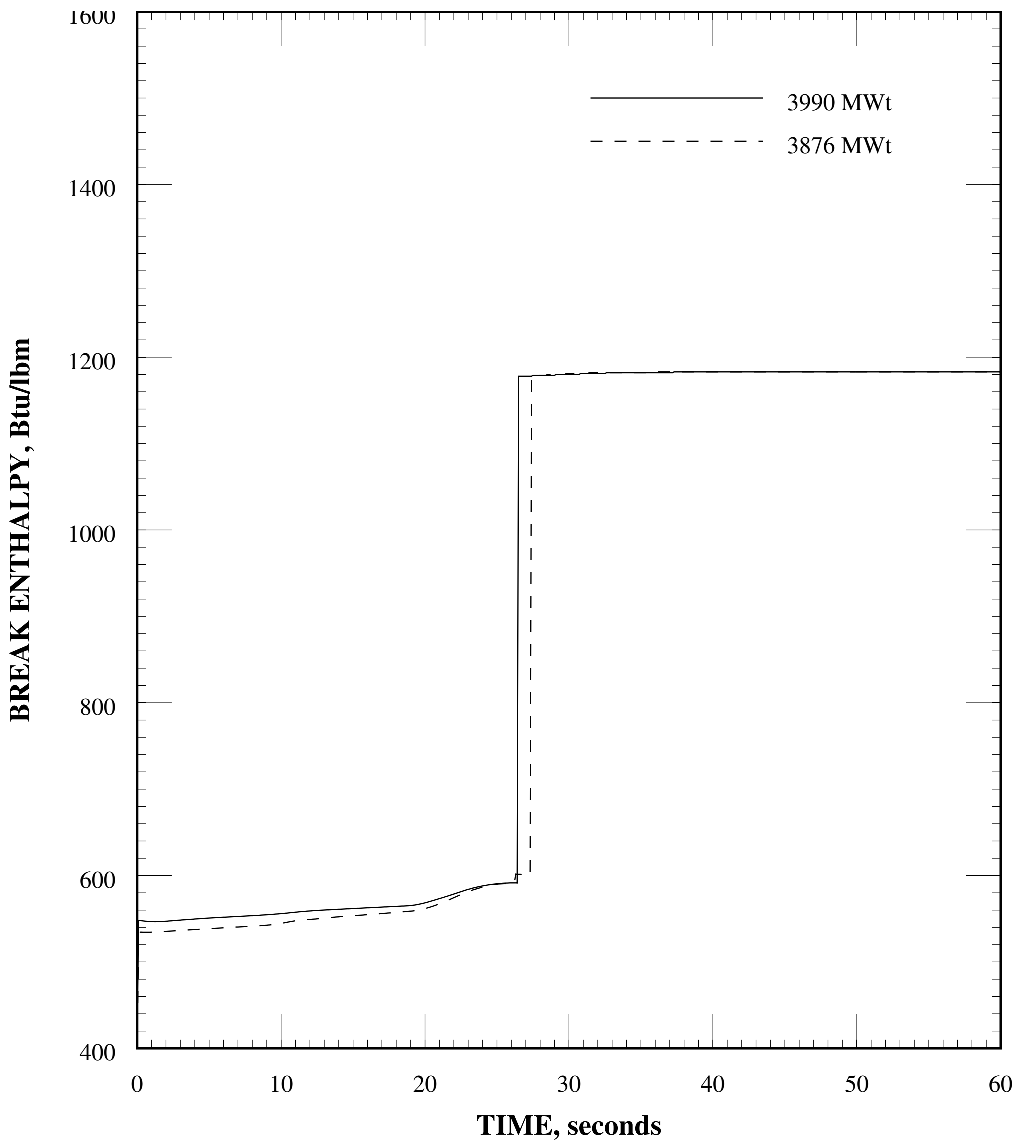


Figure 6.3-137
SFWLB Primary Peak Pressure Case - PSV Flow vs. Time

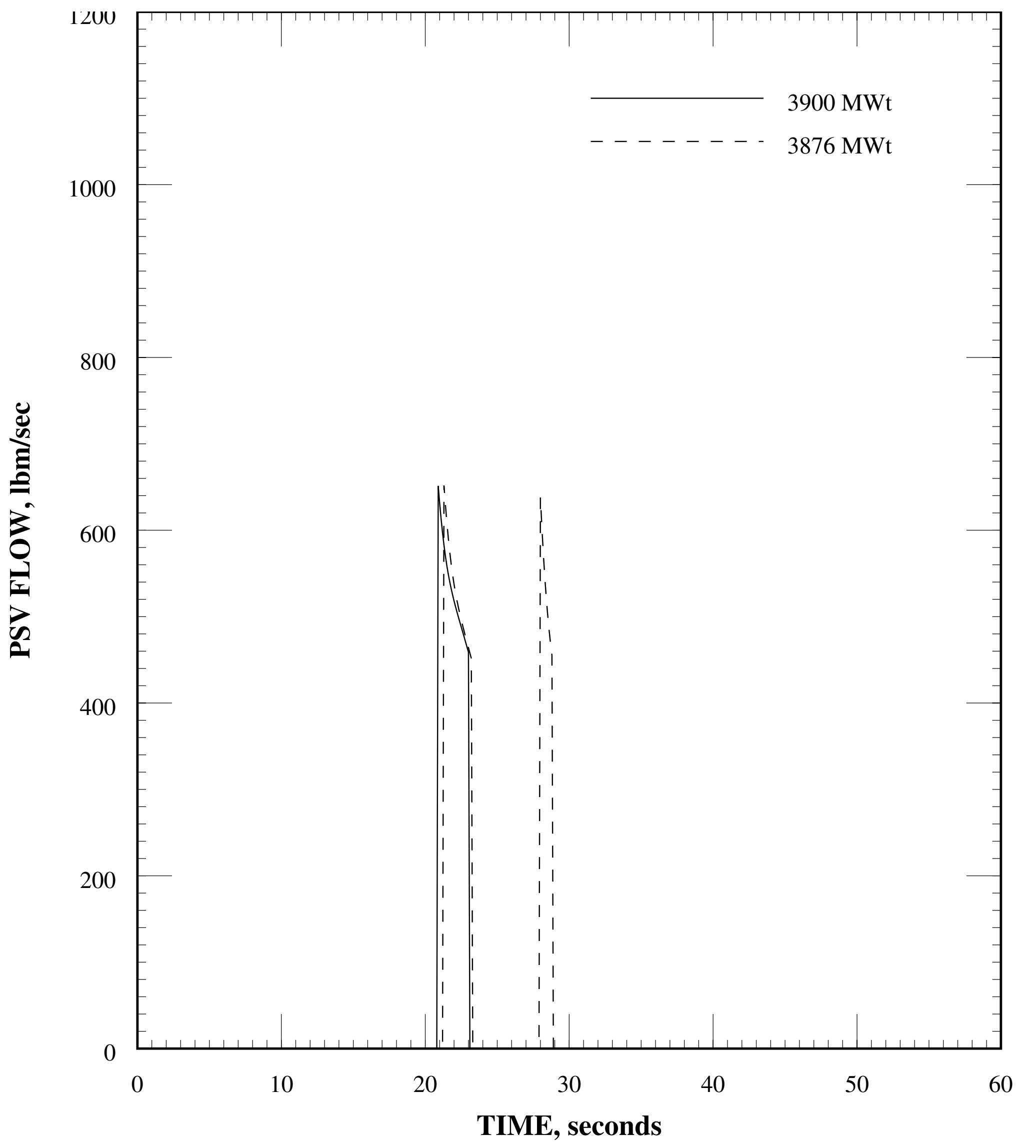
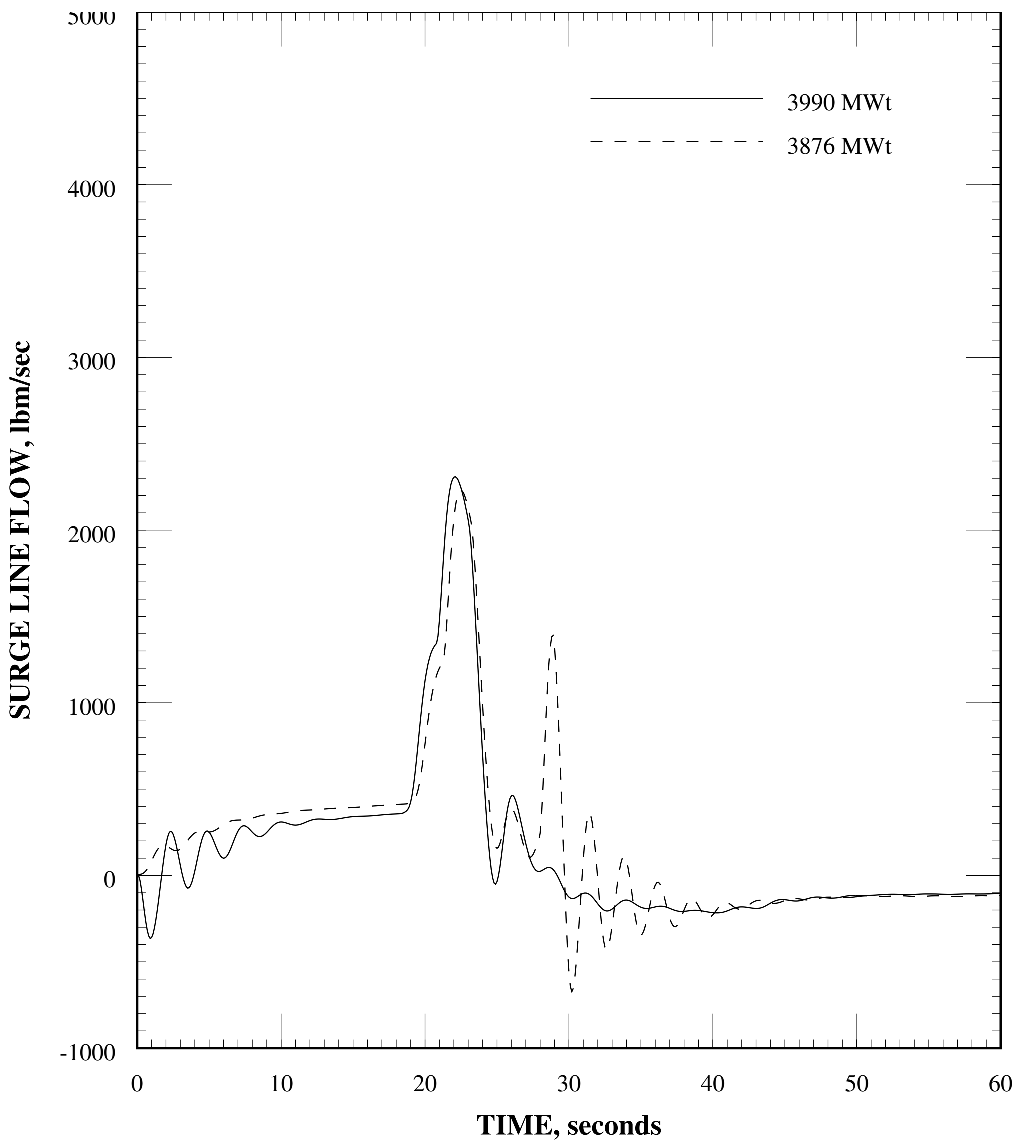


Figure 6.3-138
SFWLB Primary Peak Pressure Case - Surge Flow vs. Time



Section 6.3.3 Decrease in Reactor Coolant Flowrate

Section 6.3.3.1 Total Loss of Reactor Coolant Flow

Section 6.3.3.1.1 Identification of Event and Causes

As described in UFSAR Section 15.3.1, a complete loss of forced RCS flow will result from a simultaneous loss of electrical power to all RCPs. The only credible failure that can result in a simultaneous loss of power, is a complete LOP. In addition, since a LOP is assumed to result in a turbine trip and the steam dump and bypass systems become unavailable, the plant cooldown is performed utilizing the MSSVs and ADVs.

A total loss of forced RCS flow will produce a minimum DNBR more adverse than any partial loss of forced RCS flow event.

The LOF event plus a single failure will not result in a lower DNBR than that calculated for the LOF event alone. For decreasing RCS flow events, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel damage might be anticipated.

For the LOF event, the minimum DNBR occurs during the first few seconds of the transient and the reactor is tripped by the CPCS. Therefore, any single failure that would result in a lower DNBR during the transient would have to occur during the first few seconds of the event. None of the single failures listed in UFSAR Table 15.0-0 will have any effect on the transient minimum DNBR during this period of time.

Additionally, none of these single failures will have any effect on the peak primary system pressure. The LOP will make unavailable any systems whose failure could affect the calculated peak pressure. For example, a failure of the SBCS TBVs to modulate or quick open and a failure of the PPCS control valve to open involve systems which are assumed to be in the manual mode as a result of the LOP and, hence, unavailable for at least 30 minutes. Another example involving the PPCS would be the failure of the backup heaters to turn off. For this event, the backup heaters will not be called upon to operate due to the increase in RCS pressure.

The LOP event with a single failure is no more adverse than the LOP event in terms of the minimum DNBR and peak primary system pressure.

Section 6.3.3.1.2 Acceptance Criteria

As defined in the SRP Section 15.3.1, the specific acceptance criteria for limiting moderate frequency events are:

a. Pressure in the RCS and main steam system should be maintained below 110% of the design.
b. Fuel cladding integrity should be maintained by ensuring that Acceptance Criterion 1 of SRP Section 4.4 is satisfied throughout the transient.

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Section 6.3.3.1.3 Description of Analysis

The overall NSSS response to a total loss of RCS flow was simulated using the CENTS code. The detailed neutronic behavior of the core and hot channel to the LOF was modeled with the HERMITE code. The minimum DNBR was calculated using the TORC code which uses the CE-1 CHF correlation.

Both the existing configuration and PUR were evaluated in order to compare the NSSS response to the LOF event. Input parameters and initial conditions were selected to maximize the DNBR degradation and demonstrate that fuel cladding integrity is maintained throughout the event.

Section 6.3.3.1.3.1 Transient Simulation

The system is initialized at 102% power, using the most limiting initial parameters. At time equals zero, the LOF is simulated by a LOP which results in a turbine trip, main FW to ramp to zero and coastdown of all four RCPs. Reactor trip occurs on the CPC low RCP shaft speed trip.

Although it does not affect the DNBR, an AFAS occurs as the SG levels decrease due to the blowdown caused by the MSSVs.

Section 6.3.3.1.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-33 contains the initial conditions used for the LOF event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. There is no operator action for the first 30 minutes of the event.

Table 6.3-33
Parameters Used for the LOF Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	548	548
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
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)	16	10
Single failure	none	none
LOP	yes	yes

Section 6.3.3.1.5 Results

Table 6.3-34 presents a sequence of events that occur following the LOF event until operator action is initiated at 30 minutes. Figure 6.3-139 through Figure 6.3-146 presents the behavior of NSSS parameters following the LOF event.

Sufficient initial thermal margin has been preserved in COLSS to account for the DNBR degradation experienced during the LOF event. The minimum hot channel DNBR remains above the SAFDL for the duration of the event.

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 RCP coastdown (due to LOP) is not significantly different between the two plant configurations. Therefore, the DNBR degradation for the LOF event is equivalent.

Table 6.3-34
Sequence of Events for the LOF Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	LOP occurs	---	---
0.00	0.00	Turbine trip occurs	---	---
0.00	0.00	EDG starting signal and main FW is lost	---	---
0.60	0.60	Low RCP shaft speed trip condition	0.95	0.95
0.90	0.90	Reactor trip occurs	---	---
1.50	1.50	Scram CEAs begin falling	---	---
2.75	2.75	Minimum DNBR	>SAFDL	>SAFDL
4.40	4.40	PSVs begin to cycle open/closed (psia)	2553	2553
4.70	4.60	Maximum RCS pressure (psia)	2642	2651
30.2	28.8	MSSVs begin to cycle open/closed (psia)	1303	1303
30.2	28.8	Maximum SG pressure (psia)	1303	1303
141.0	140.8	PSVs close (psia)	2469	2475
1065.4	1057.6	MSSVs close (psia)	1279	1279
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.3.1.6 Conclusions

For the RCS LOF event, all acceptance criteria are met. The peak primary and secondary pressures remain below 110% of design at all times thus ensuring the integrity of the RCS or the main steam system. The minimum DNBR remains above the safety analysis limit, thus ensuring fuel cladding integrity.

Figure 6.3-139
Total Loss of RCS Flow - Core Power vs. Time

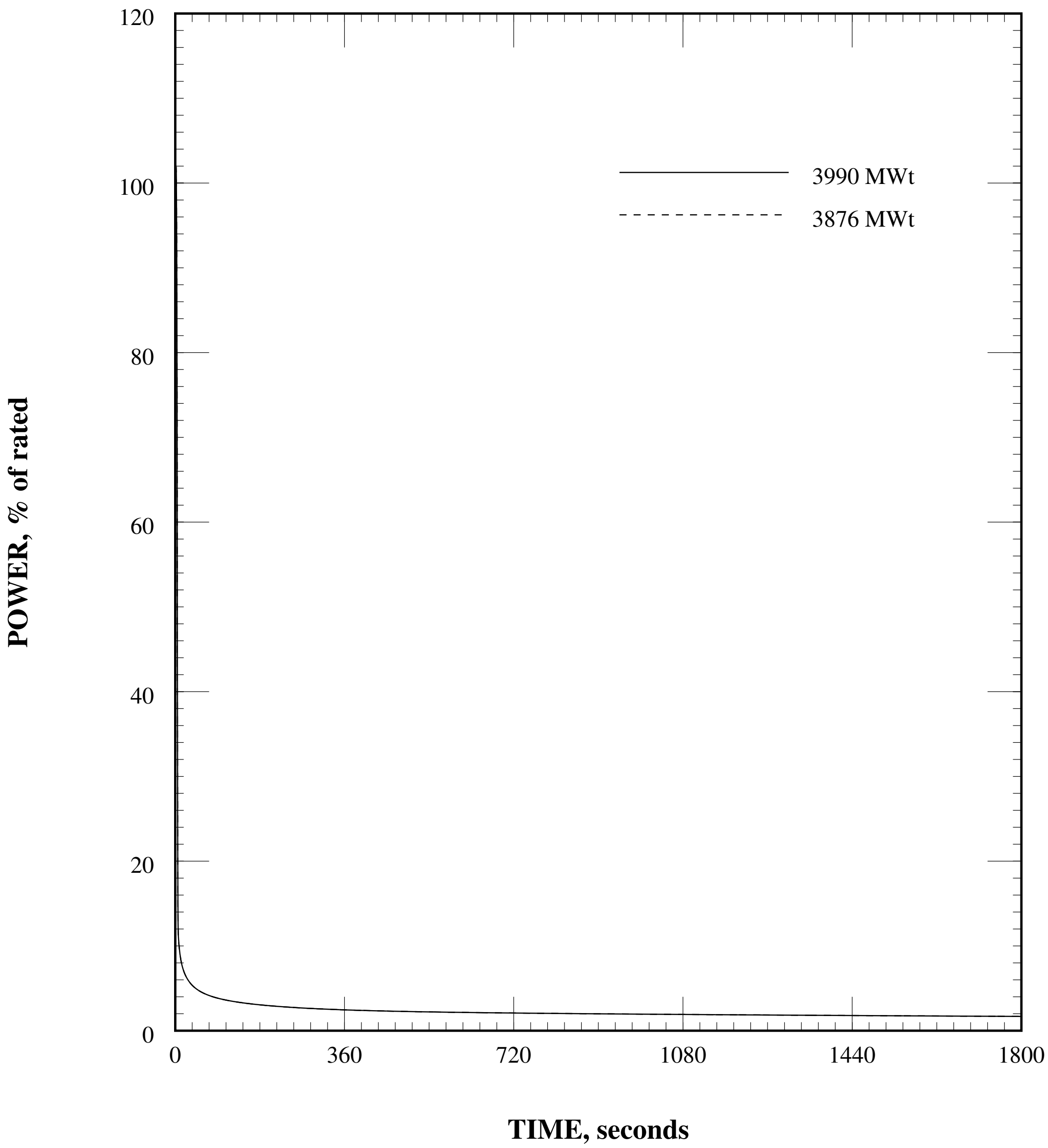


Figure 6.3-140
Total Loss of RCS Flow - Core Average Heat Flux vs. Time

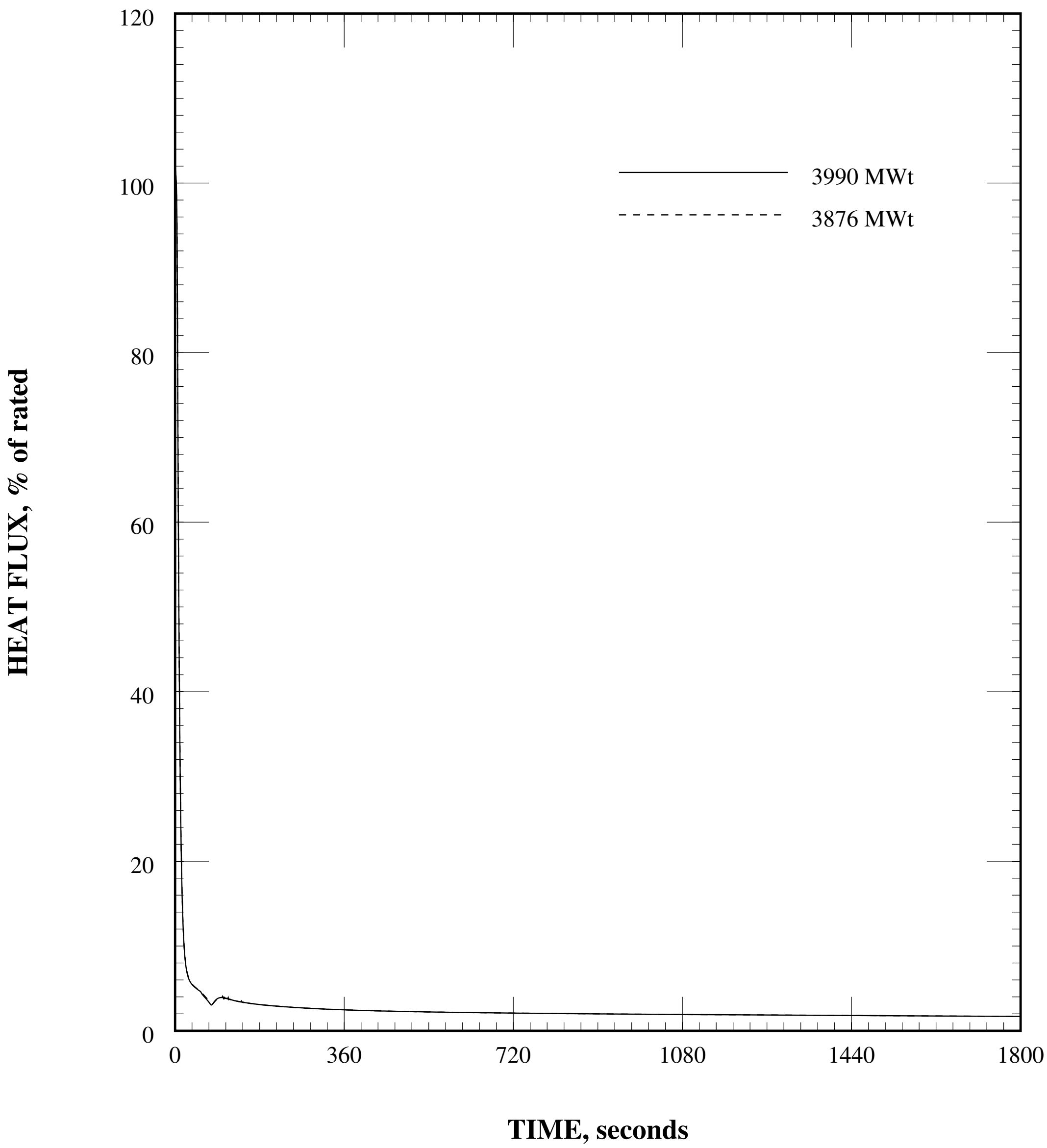


Figure 6.3-141
Total Loss of RCS Flow - RCS Pressure vs. Time

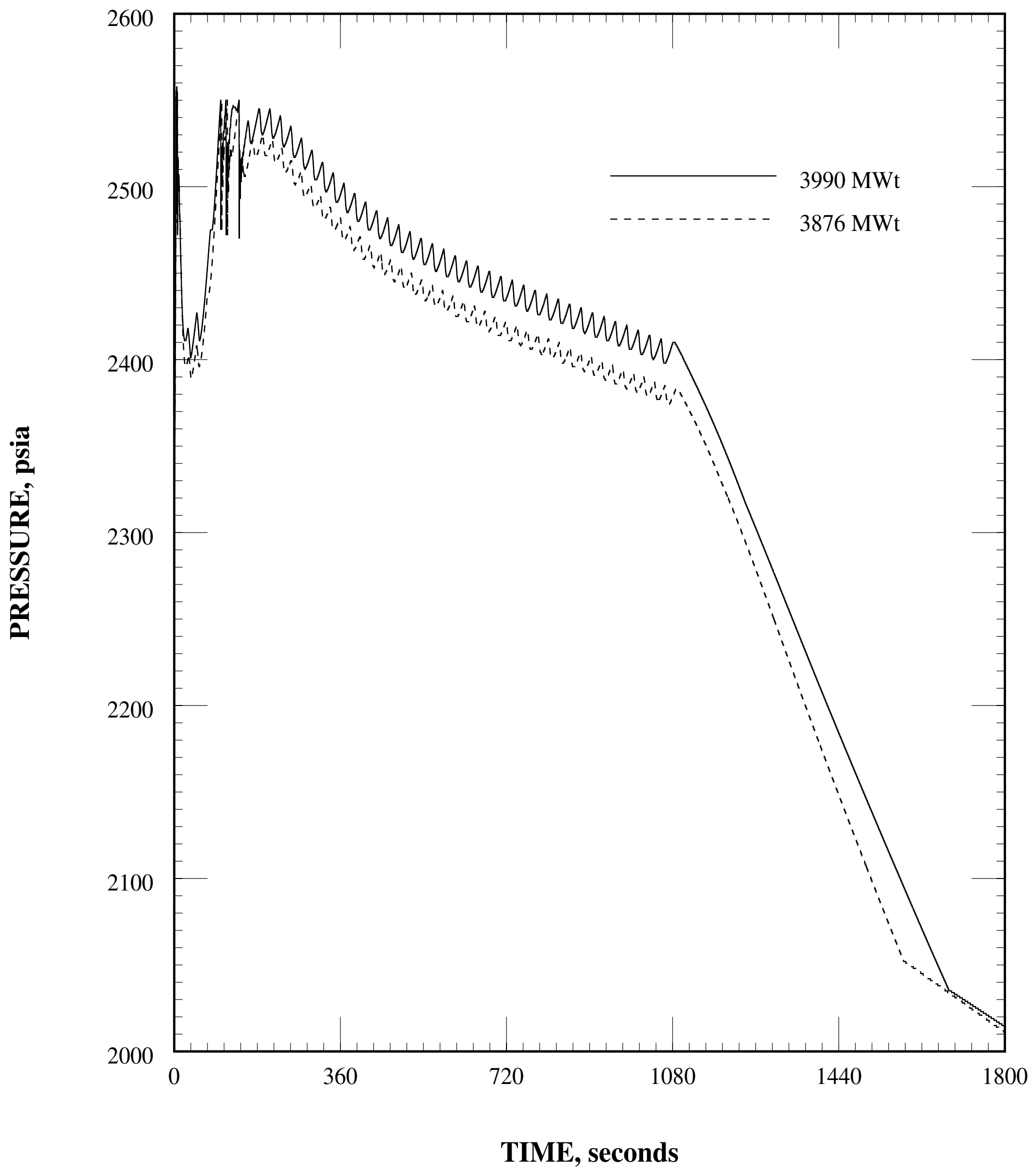


Figure 6.3-142
Total Loss of RCS Flow - RCS Temperature vs. Time

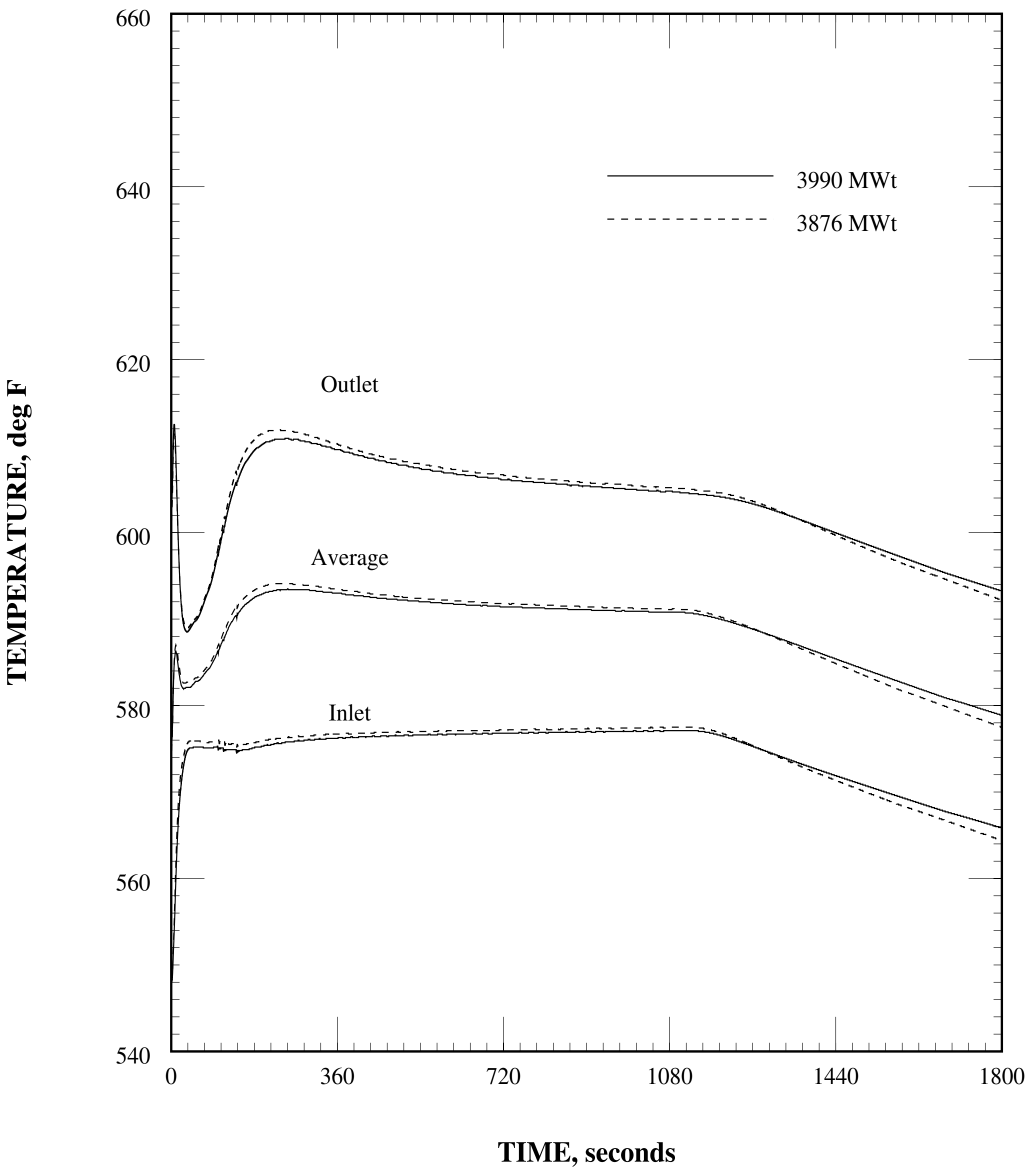


Figure 6.3-143
Total Loss of RCS Flow - Reactivity vs. Time

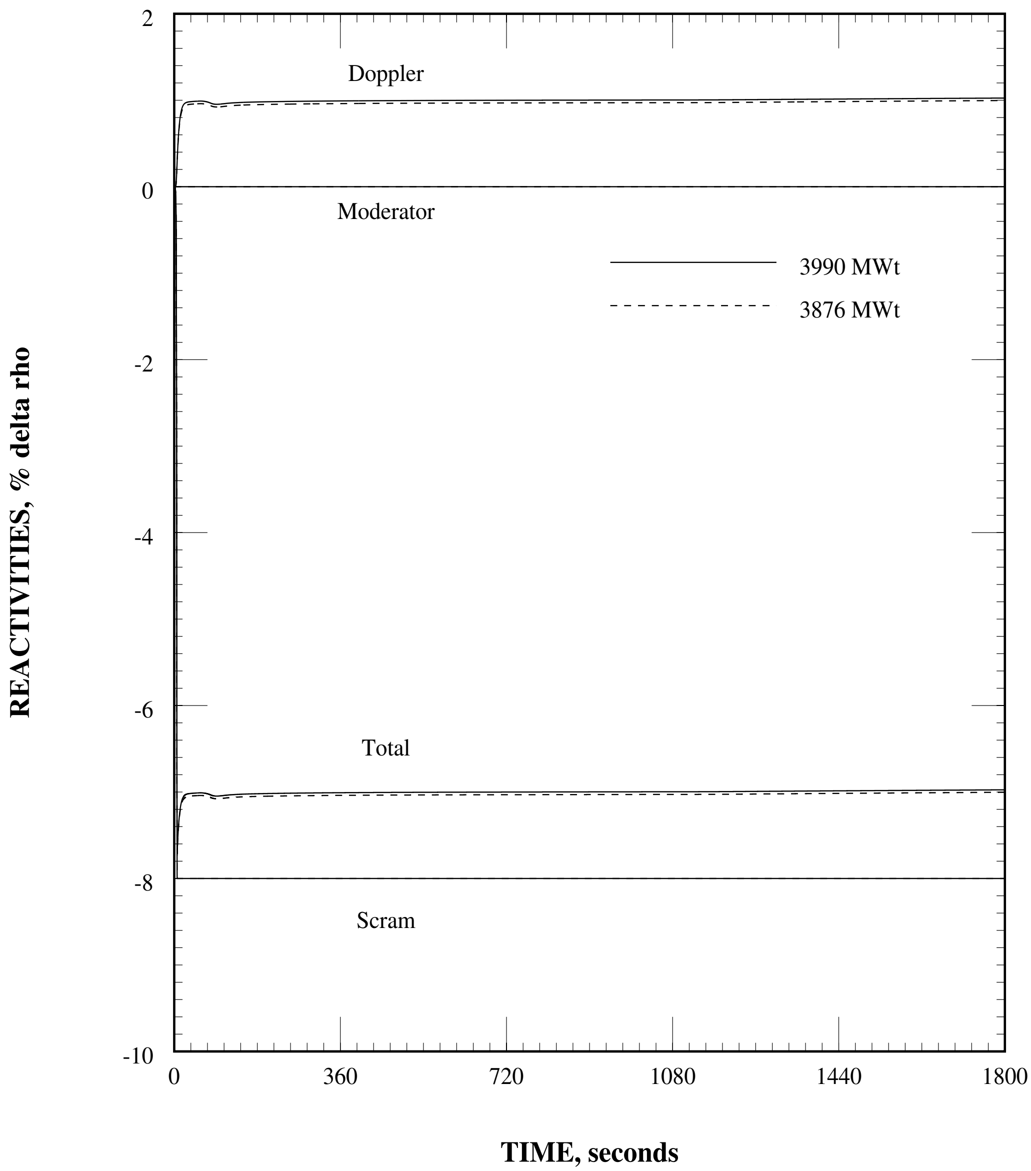


Figure 6.3-144
Total Loss of RCS Flow - Core Flow Fraction vs. Time

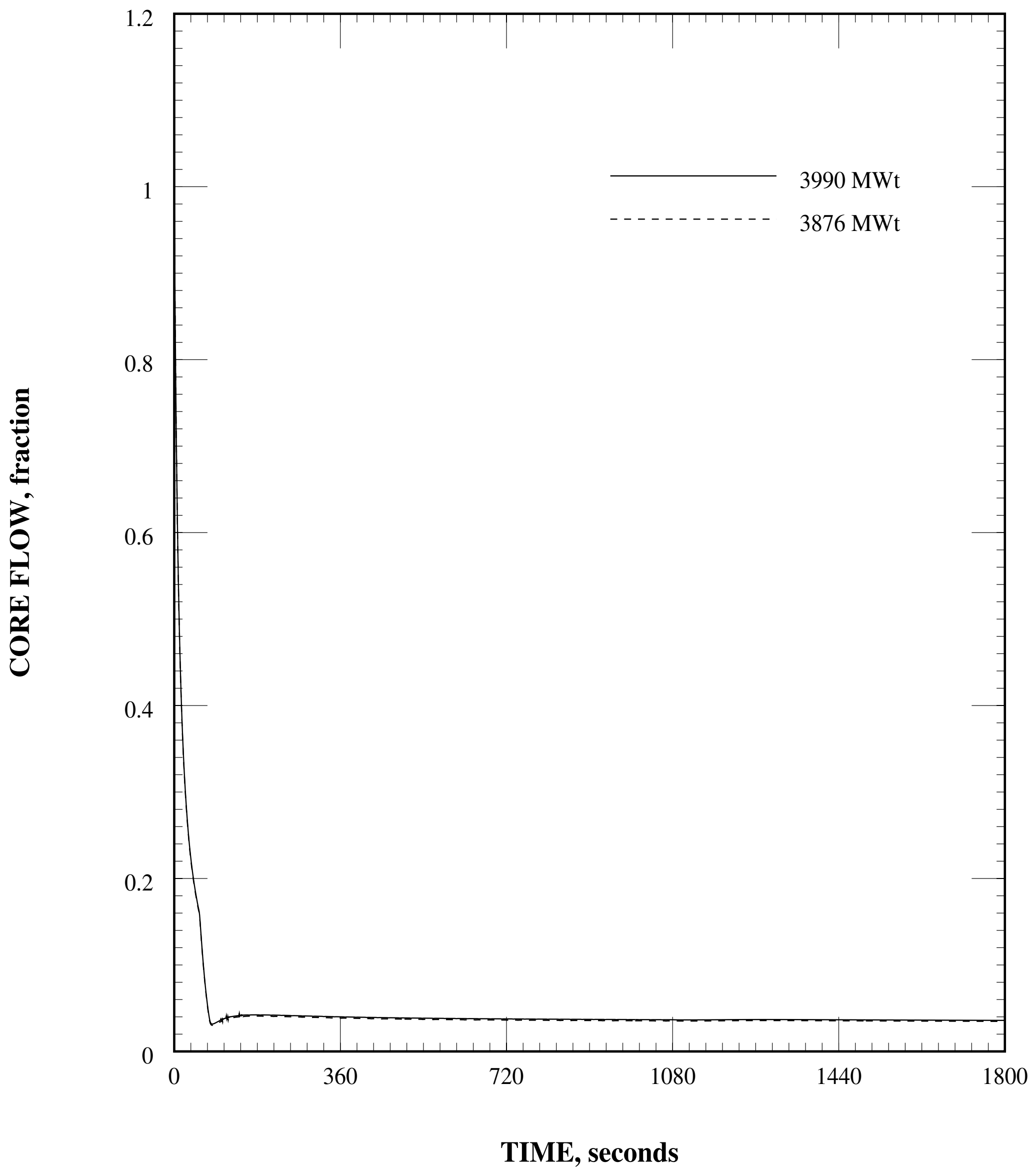


Figure 6.3-145
Total Loss of RCS Flow - SG Pressure vs. Time

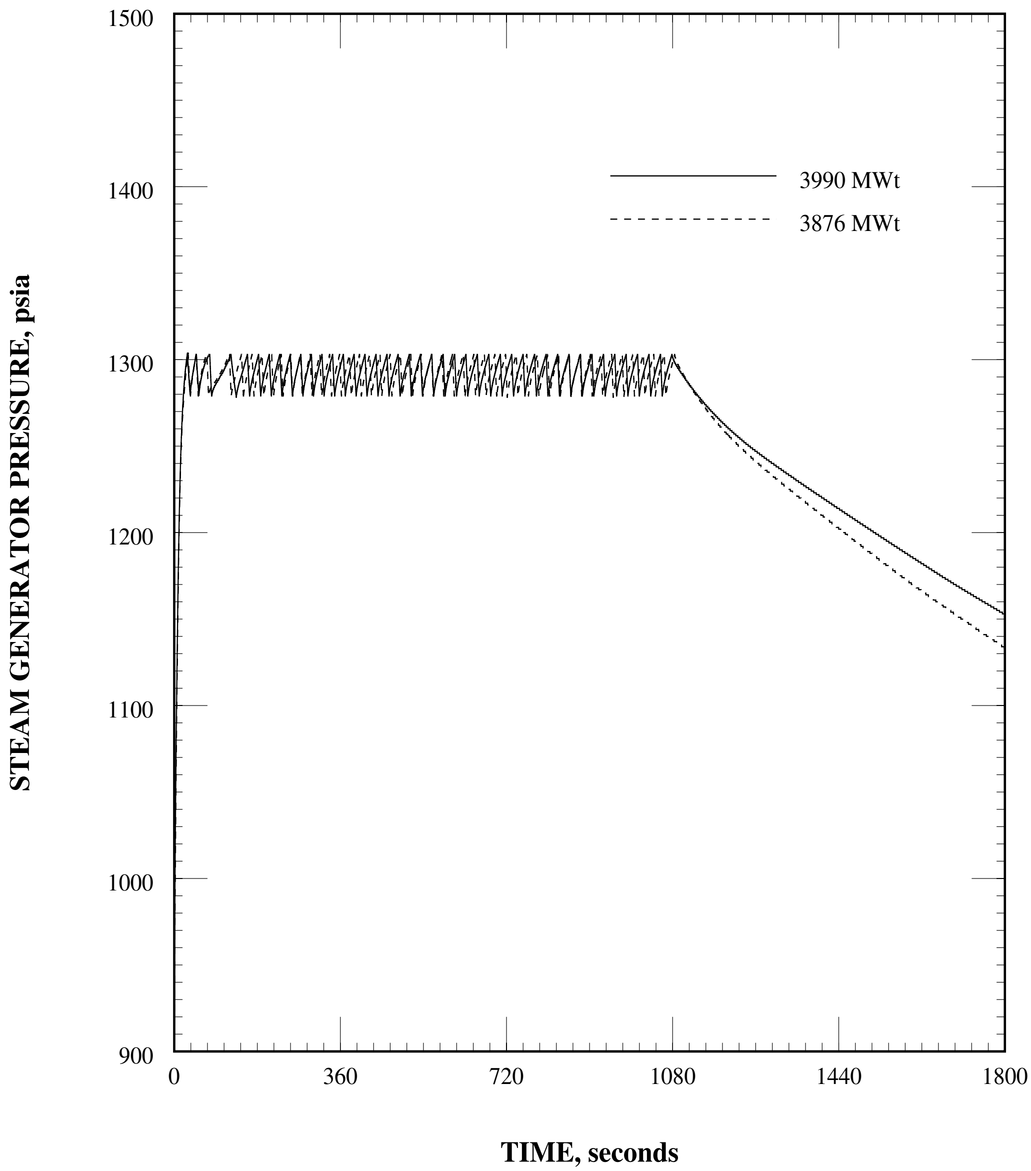
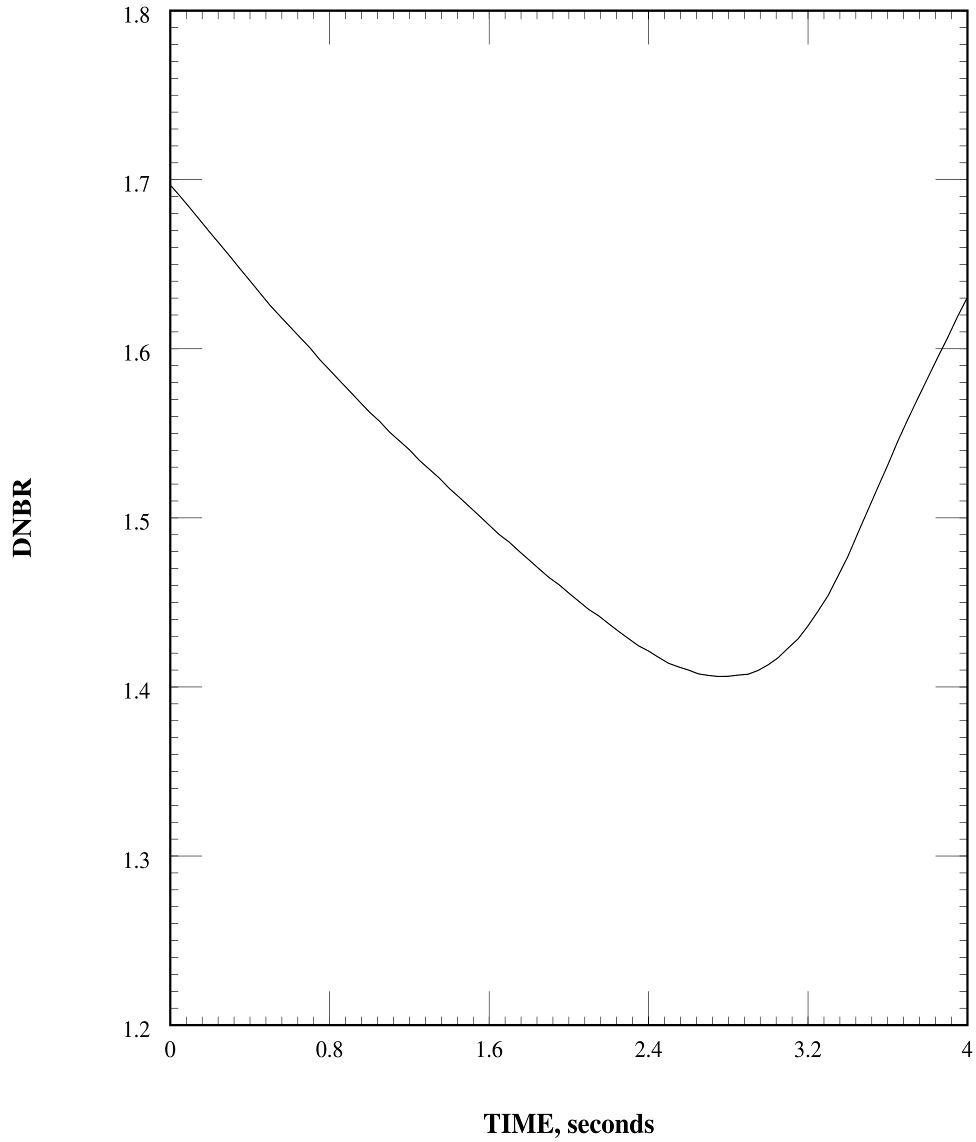


Figure 6.3-146
Total Loss of RCS Flow - Minimum DNBR vs. Time



Section 6.3.3.2 Flow Controller Malfunction Causing a Flow Coastdown

As described in UFSAR Section 15.3.2, the SRP classifies the flow controller malfunction event as pertaining to Boiling Water Reactors (BWRs). This event is not applicable and is not analyzed.

Section 6.3.3.3 Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power

As described in UFSAR Section 15.3.3, the rotor seizure event would be no more adverse than the RCP shaft break event as discussed in Section 6.3.3.4.

Section 6.3.3.4 Reactor Coolant Pump Shaft Break with Loss of Offsite Power

Section 6.3.3.4.1 Identification of Event and Causes

As described in UFSAR Section 15.3.4, a single RCP rotor seizure can be caused by seizure of the upper or lower thrust-joint bearings. A single RCP sheared shaft could be caused by mechanical failure of the pump shaft.

The sequence of events for the RCP rotor seizure is similar to that of a RCP shaft break. The difference is that for the rotor seizure event, the reactor is tripped by the CPCS on a low RCP speed condition, whereas for the shaft break event, the reactor is tripped by the PPS on a low RCS flow condition (SG ΔP).

The seized rotor, having the greater resistance to the RCS flow, has a slightly faster coastdown. The sudden stopping of the RCP rotor and motor assembly results in a CPC RCP speed-based trip.

The RCP shaft break allows a freewheeling coastdown of the impeller with the RCP motor continuing to rotate. The RCS flow coastdown is slightly slower, but the continued motion of the RCP motor does not generate the CPCS trip. Protection for this event is delayed until the SG differential pressure low flow RPS trip is generated.

Approximately 3 seconds following the turbine trip, a LOP causes a loss of power to the onsite loads. This results in a simultaneous LOFW, condenser inoperability, and a coastdown of the remaining three RCPs.

For decreasing reactor flow events, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated, and thus whether fuel damage could be anticipated.

Minimum DNBR occurs within the first 4 seconds of the event. Any single failure would have to occur within this time span to affect minimum DNBR. None of the postulated single failures that could occur in the first four seconds would result in a more adverse DNBR than that caused by the event itself.

The assumed LOP renders the SBCS inoperable. This results in the secondary system steam being released to the atmosphere by the MSSVs (before operator action) and by the ADVs after operator action is assumed. The dose calculation assumes a single failure of an ADV to close. The stuck-open ADV was opened by the operators following the reactor trip. The affected SG eventually empties due to the continuous steaming through the stuck-open ADV. After 30 minutes, the operator begins refilling the affected SG, covering the top of the tubes after 90 minutes. This is a change in previous methodology. This failure in combination with the LOP maximizes the radiological consequences of the single RCP shaft break event. None of the other single failures in combination with a LOP will yield more severe radiological consequences. The dose calculation is presented in Section 6.4.3.1.

<u>Section 6.3.3.4.2</u> <u>Acceptance Criteria</u>

As defined in the SRP Section 15.3.4, the specific acceptance criteria for this event are:

a. For events such as the rotor seizure or shaft break in a RCP, the plant should be designed to limit the release of radioactive material to assure that doses to persons offsite are kept to values which are a small fraction of 10 CFR Part 100 guidelines.
b. The potential for core damage should be evaluated based on the acceptance criterion for DNBR in SRP Section 4.4. If DNBR falls below the limits of this criterion, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model that no fuel failure results. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and unaffected with no loss of core cooling capability.
c. Pressure in the RCS and main steam system should be maintained below 110% of the design pressure.
d. A rotor seizure or shaft break in a RCP should not, by itself, generate a more serious condition or result in a loss of function of the RCS or containment barriers.
e. Only safety-grade equipment should be used to mitigate the consequences of the accident. Safety functions should be accomplished assuming the worst single failure of a safety system active component.
f. The ability to achieve long-term coolability of the core should be verified.
g. This event should be analyzed assuming turbine trip and coincident LOP and coastdown of undamaged pumps.

Also, with the inclusion of a LOP and a fully stuck-open ADV, the 10 CFR Part 100 guidelines for offsite doses are met.

<u>Section 6.3.3.4.3</u> <u>Description of Analysis</u>

Transient core response was simulated using the CENTS and HERMITE codes to generate core conditions at the time of minimum DNBR. Output from the HERMITE code at the time of minimum DNBR was then used as input to the TORC code using a 3-pump model to generate radial peak versus DNBR values.

Section 6.3.3.4.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-35 contains the initial conditions used for the single RCP sheared shaft event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. For the radiological consequences (see Section 6.4.3.1), it is assumed that an operator opens an ADV on each SG two minutes after the trip for secondary system pressure control. SG1 ADV instantaneously opens fully and sticks open. After closure of SG2 ADV shortly thereafter, no further operator action is credited until 30 minutes.

Table 6.3-35
Parameters Used for the Sheared Shaft Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	100	100
Initial core inlet temperature (°F)	548	548
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-0.18E-04	-0.18E-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)	16	10
Single failure	stuck-open ADV	stuck-open ADV
LOP	yes	yes

Section 6.3.3.4.5 Results

Table 6.3-36 presents a sequence of events that occur following the sheared shaft event until operator action is initiated at 30 minutes. Figure 6.3-147 through Figure 6.3-154 presents the behavior of NSSS parameters following the event.

Based on fuel failure results, the RCP sheared shaft is the more limiting of the two scenarios. Figures and sequence of events reflect the sheared shaft event. The initiating event causes a flow coastdown in the affected loop and a consequent reduction of flow in the core. This results in an increase in the average core coolant temperature and a corresponding reduction in the margin to DNB. Primary system pressure also increases. A reactor trip is generated on a SG differential pressure low RCS flow trip and the CEAs drop into the core. The reactor trip causes a turbine trip. Three seconds later, a LOP occurs. The LOP also causes a LOFW and condenser operability. The turbine trip, with the SBCS and the condenser unavailable, leads to a rapid buildup in secondary system pressure and temperature. The opening of primary and secondary relief valves limits this pressure increase to less than 110% of design.

The increasing temperature of the secondary system leads to a reduction of the primary to secondary heat transfer. Concurrently, the failed RCP and the three RCPs coasting down result in RCS flow that further reduces the heat transfer capability. This decrease in heat removal from the RCS leads to an increase in the core coolant temperatures. The core coolant temperatures peak shortly after the reactor trip.

The increase in RCS temperature leads to an increase in RCS pressure caused by thermal expansion of the RCS fluid. The RCS pressure remains less than 110% of design pressure throughout the transient. The RCS pressure decreases rapidly due to opening of PSVs and decreasing core heat flux. Opening of the MSSVs limits the peak temperature and pressure of the secondary system. The MSSVs cycle until the AFW begins entering the SGs, enhancing the RCS cooldown and subsequent reduction in pressure.

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 single RCP coastdown (due to sheared shaft) is not significantly different between the two plant configurations. Therefore, the DNBR degradation and amount of fuel failure experienced for this event is equivalent.

Radiological consequences for this event are presented in Section 6.4.3.1.

Table 6.3-36
Sequence of Events for the Sheared Shaft Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.0	0.0	Sheared shaft occurs	---	---
2.5	2.5	SG ΔP reaches low RCS flow variable setpoint	---	---
3.0	3.0	Reactor trip breakers open	---	---
3.0	3.0	Turbine trip occurs	---	---
3.6	3.6	Scram CEAs begin falling	---	---
5.5	5.5	LOP occurs	---	---
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the LOF event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.3.4.6 Conclusions

For the single RCP sheared shaft in combination with a LOP following a turbine trip event, all acceptance criteria are met. The peak primary and secondary pressures remained below 110% of design at all times thus ensuring the integrity of the RCS or the main steam system. Fuel pins experience DNB for a short duration thus ensuring no DNB propagation. Offsite doses remained below the acceptance criteria for this category of event. Specifically, within 10 CFR Part 100 guidelines (i.e., 300 REM thyroid, 25 REM whole body).

Figure 6.3-147
Single RCP Sheared Shaft - Core Power vs. Time

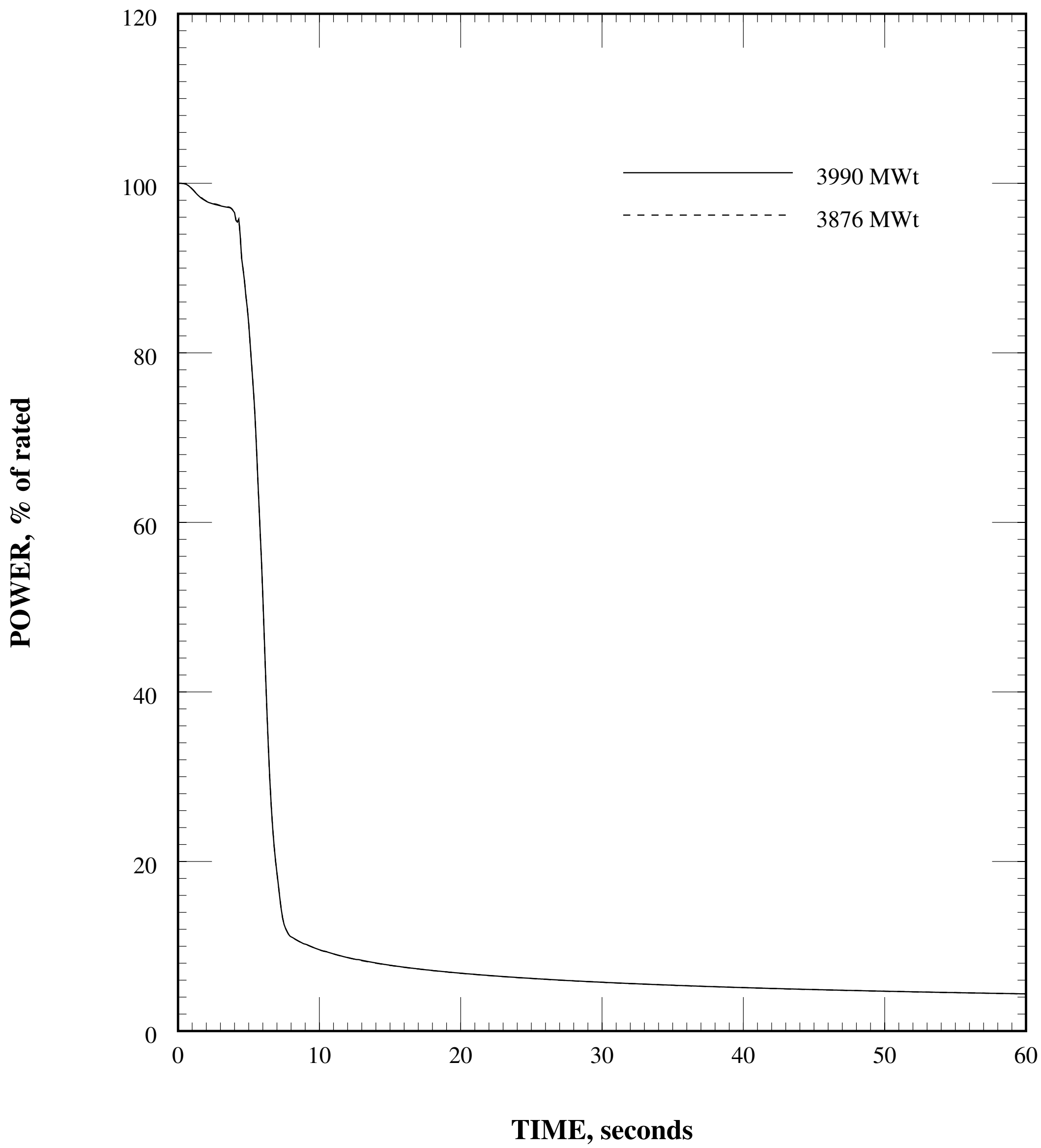


Figure 6.3-148
Single RCP Sheared Shaft - Core Average Heat Flux vs. Time

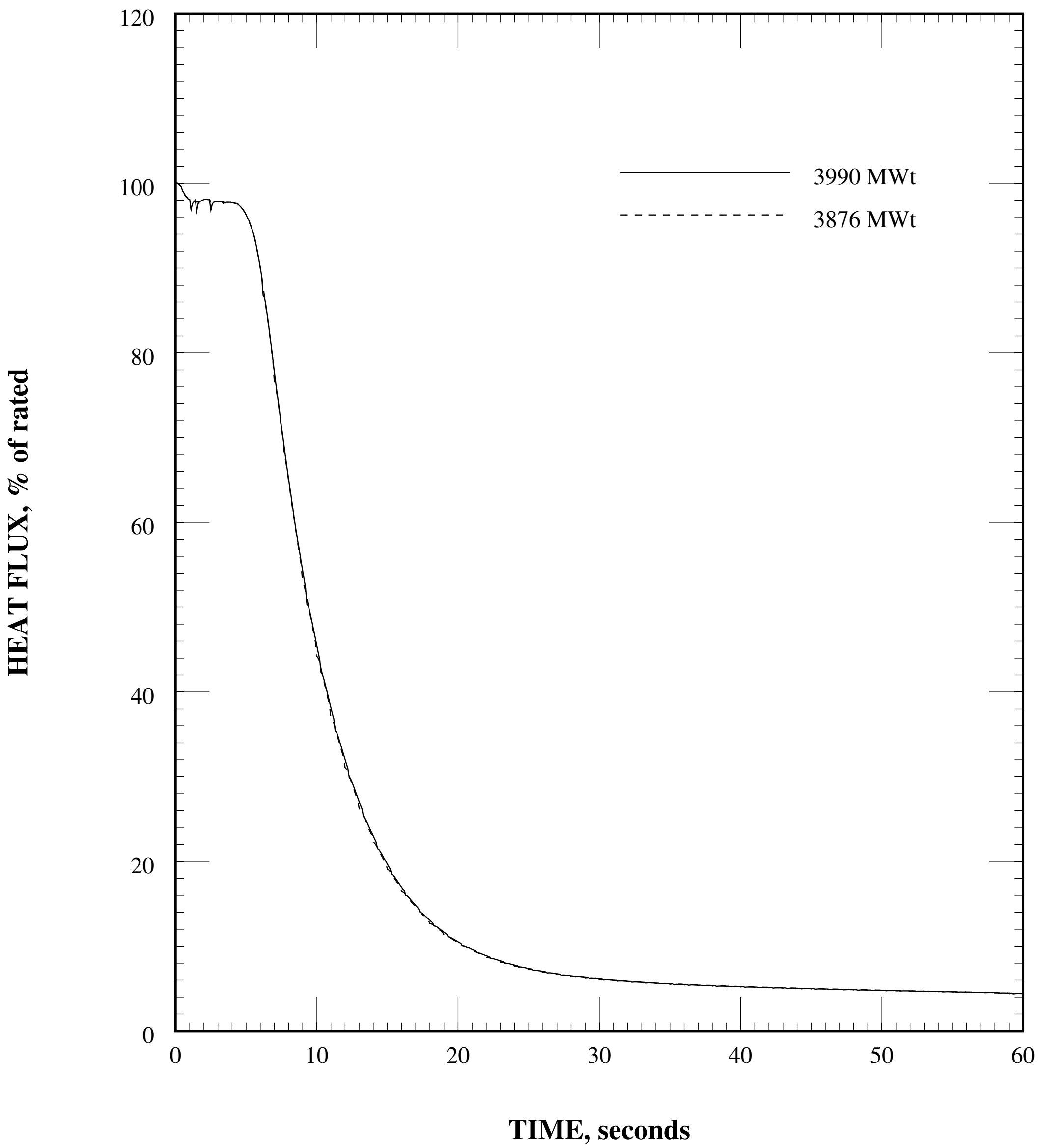


Figure 6.3-149
Single RCP Sheared Shaft - RCS Pressure vs. Time

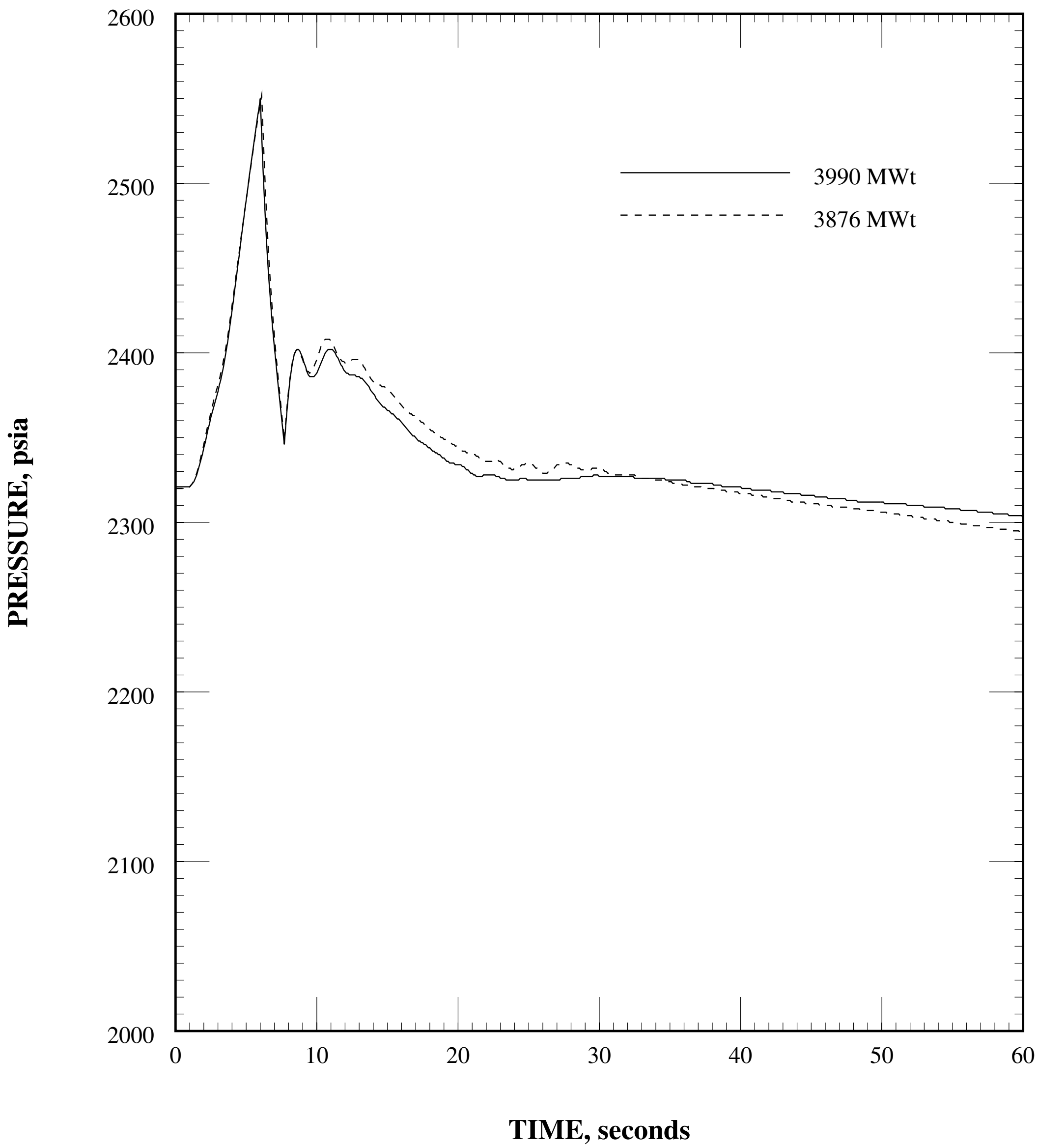


Figure 6.3-150
Single RCP Sheared Shaft - RCS Temperature vs. Time

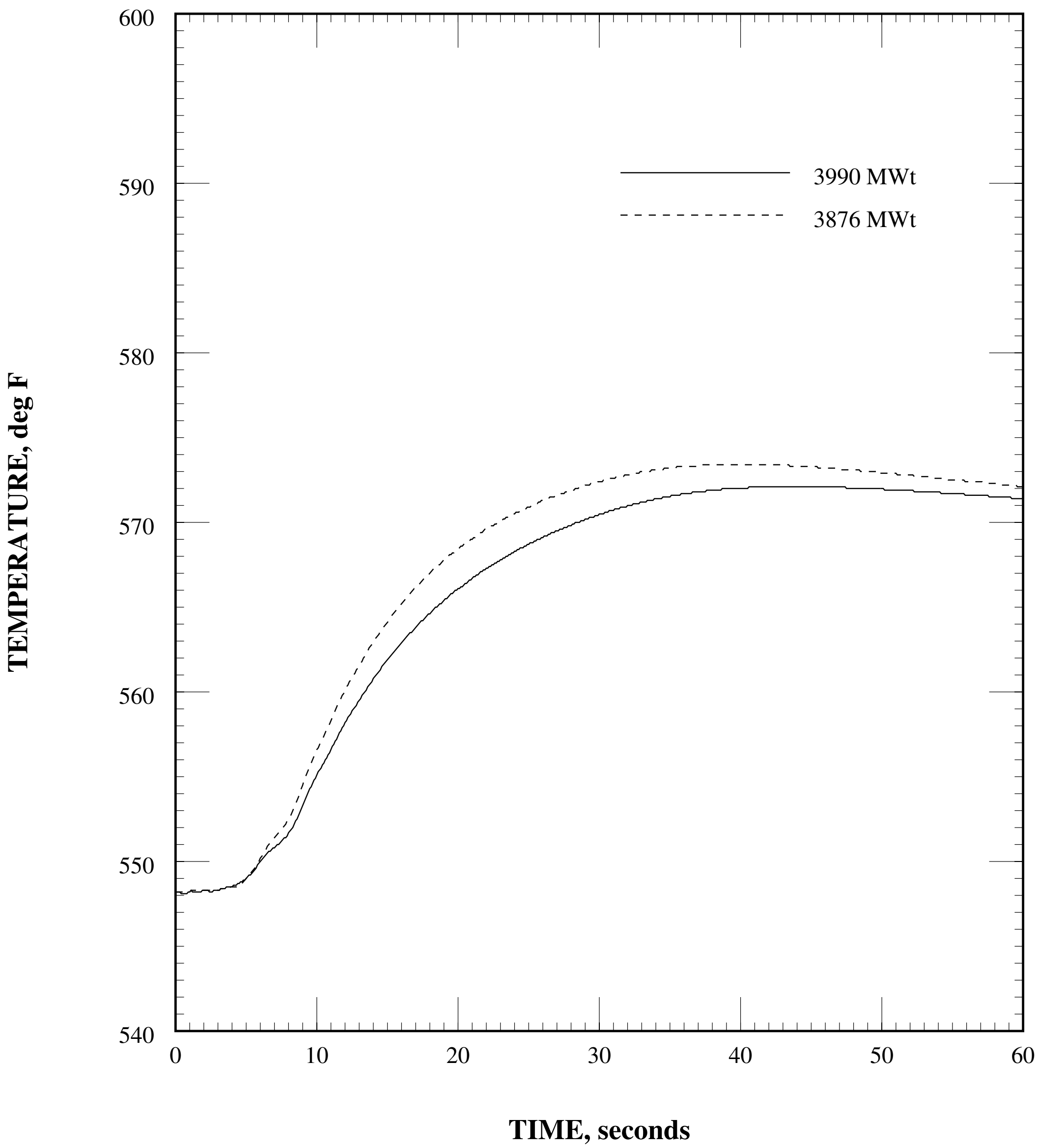


Figure 6.3-151
Single RCP Sheared Shaft - Reactivity vs. Time

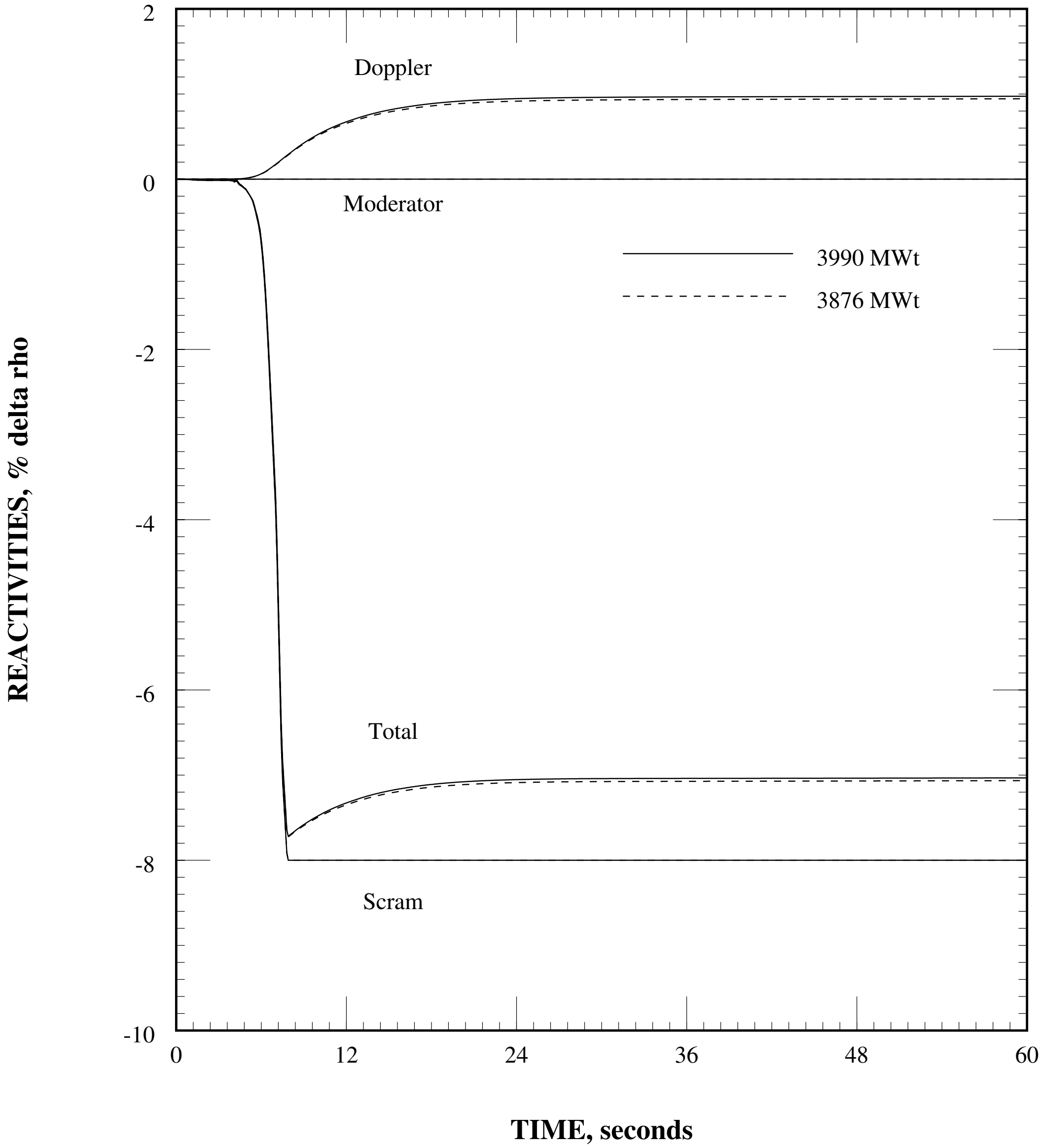


Figure 6.3-152
Single RCP Sheared Shaft - Core Flow Fraction vs. Time

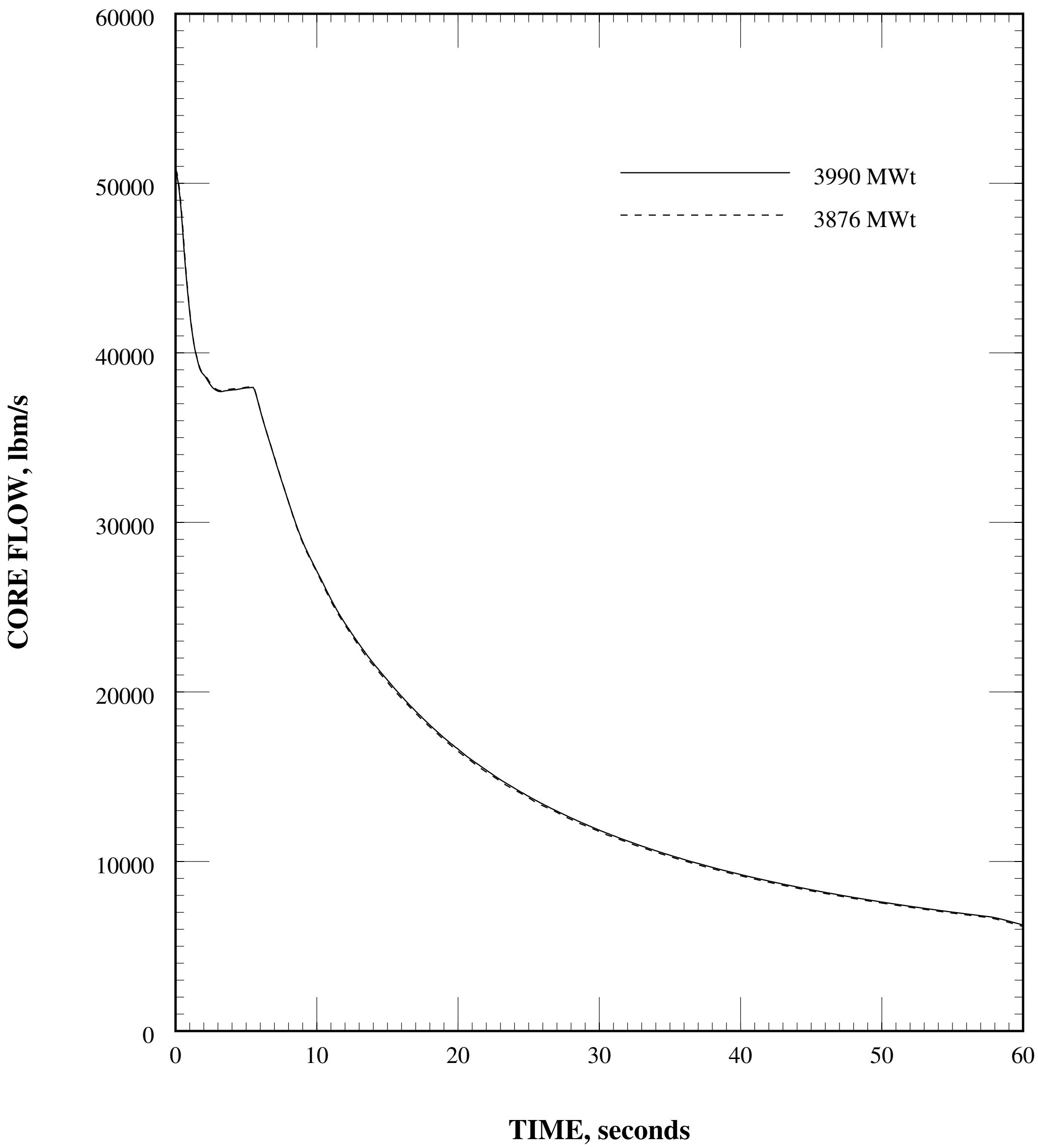


Figure 6.3-153
Single RCP Sheared Shaft - SG Pressure vs. Time

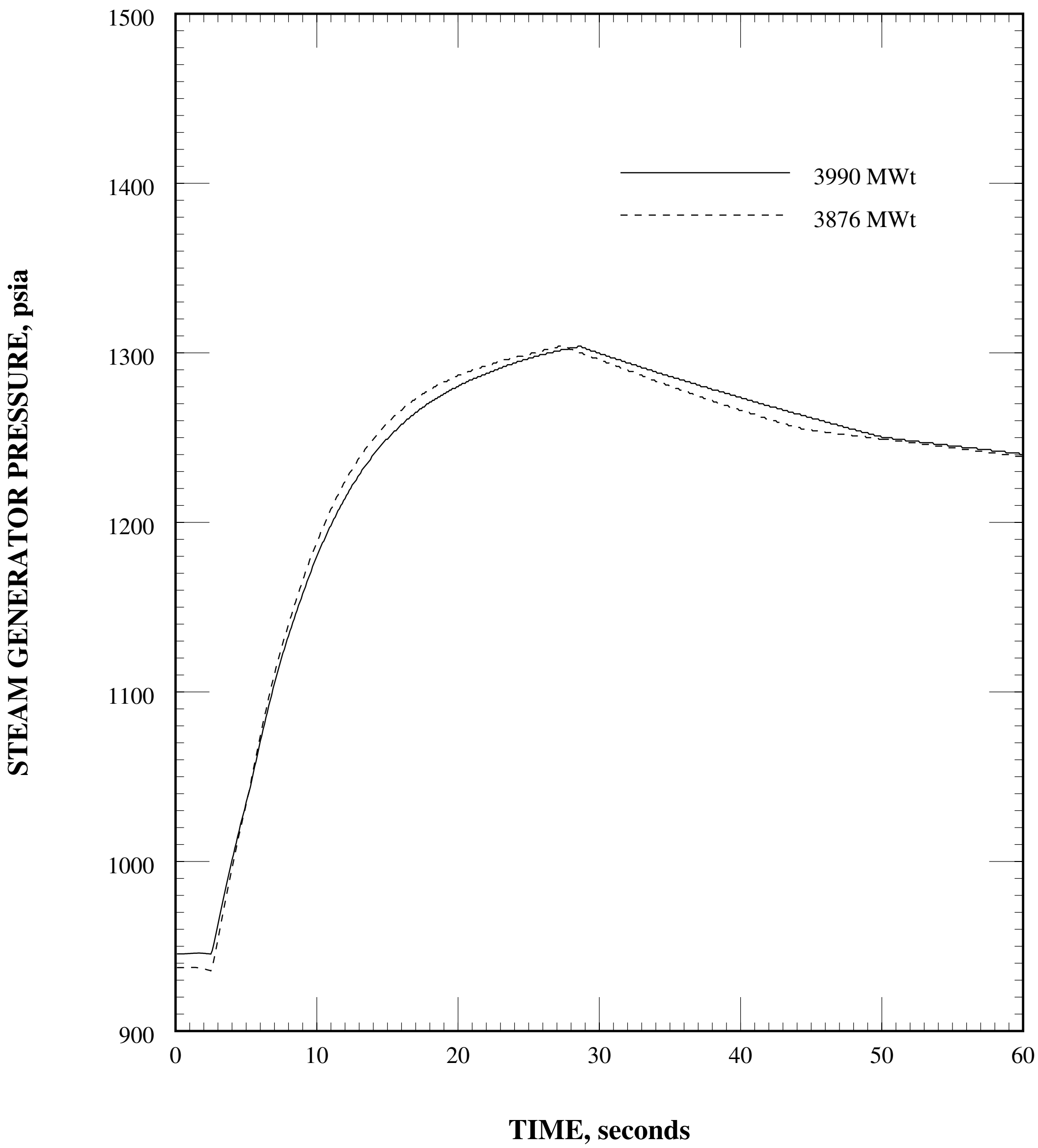
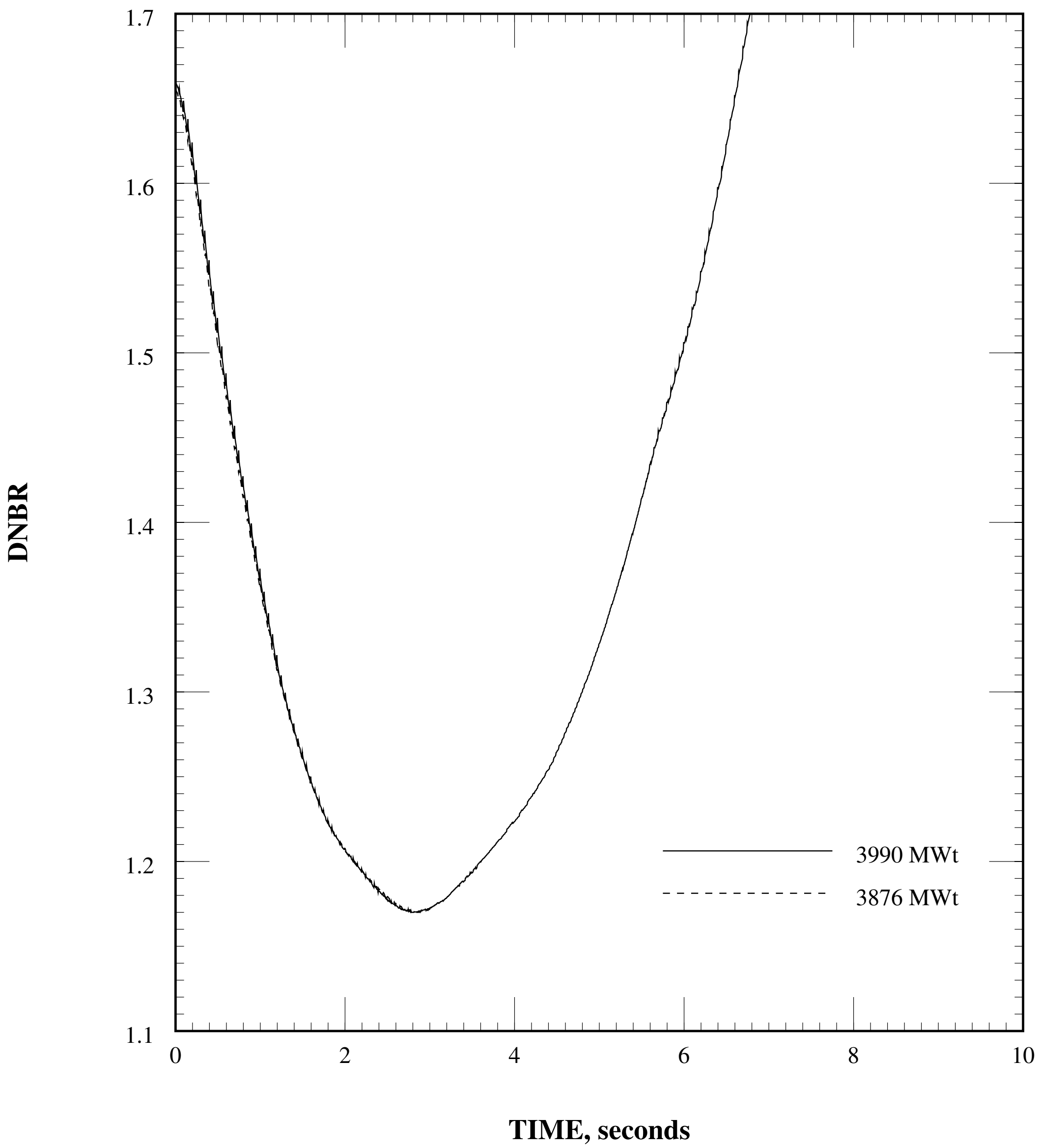


Figure 6.3-154
Single RCP Sheared Shaft - Minimum DNBR vs. Time



Section 6.3.4 Reactivity and Power Distribution Anomalies

Section 6.3.4.1 Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition

Section 6.3.4.1.1 Identification of Event and Causes

As described in UFSAR Section 15.4.1, an uncontrolled sequential CEAW is assumed to occur because of a single failure in the CEDM, Control Element Drive Mechanism Control System (CEDMCS), RRS, or because of operator error.

Section 6.3.4.1.2 Acceptance Criteria

As defined in the SRP Section 15.4.1, the specific acceptance criteria for this event are:

The following GDC apply:

a. Criterion 20 that requires that the RPS action be initiated automatically.
b. Criterion 25 that requires that the RPS be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

The following fuel design limits serve as the acceptance criteria for this event:

a. Minimum DNBR remains above SAFDL.
b. Fuel temperature and clad strain limits consistent with the acceptance criteria of SRP Section 4.2 should not be exceeded. For steady state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate. For PWRs, a steady-state linear heat generation rate of 20-22 kW/ft would result in a centerline fuel temperature equal to or less than the melting point of UO_2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

Section 6.3.4.1.3 Uncontrolled Control Element Assembly Withdrawal from a Subcritical Condition

Section 6.3.4.1.3.1 Description of Analysis

The NSSS response to a CEAW from a subcritical event was simulated using the CENTS code. The transient DNBR values were calculated using the CETOP-D code that uses the CE-1 CHF correlation. An initial subcritical power level that results from conservative neutron source strength was assumed.

Both the existing configuration and PUR were analyzed in order to compare the NSSS response to the CEAW event. Input parameters and initial conditions were selected to maximize local power and DNBR degradation and demonstrate that fuel cladding integrity is maintained throughout the event.

Section 6.3.4.1.3.1.1 Transient Simulation

At time equal zero, the CEAW was initiated as specified for the transient. The withdrawal continued until a HLPT signal was generated. Following the holding coil delay, the withdrawn bank begins to drop back into the core. The magnitude of the negative reactivity associated with the bank insertion is equal to the positive reactivity added by the CEAW before the trip. Credit was taken only for re-insertion of the CEAs withdrawn to during the event.

Section 6.3.4.1.3.2 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-37 contains the initial conditions used for the subcritical CEAW event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The reactivity worth versus position associated with the inward motion of the withdrawn CEAs during reinsertion was assumed the same worth at a given position that the CEAs had during their withdrawal. This is conservative since a higher flux will be present during the scram that would result in a higher CEA worth than during the withdrawal. The worth versus insertion time was then calculated based on a withdrawal rate of 30 in/min and the reactivity insertion rate used during withdrawal.
3. There is no operator action for the first 30 minutes of the event.

Table 6.3-37
Parameters Used for the CEAW from Subcritical Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	6.27E-10	6.27E-10
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	1785	1785
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	0.5E-04	0.5E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Reactivity insertion rate (%$\Delta\rho$/in)	0.065	0.065
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6530	6530
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.4.1.3.3 Results

Table 6.3-38 presents a sequence of events that occur following a CEAW from subcritical conditions. Figure 6.3-155 through Figure 6.3-160 presents the behavior of NSSS parameters following a CEAW from subcritical conditions.

The CEAW gradually reduces the amount by which the core was subcritical. During this time, subcritical multiplication causes core power to increase. A reactor trip on HLPT is generated before core power reaches the point of adding sensible heat. Due to the rapid rate of power increase and excess reactivity state of the core at the time of trip generation and the effect of continued CEAW until the trip breakers open, a brief power excursion occurs past the point of adding sensible heat. The CEAs begin dropping into the core terminating the power escalation. The hot channel minimum DNBR remains above the SAFDL. The peak LHGR exceeded the safety limit defined in the Technical Specifications for a short time. However, an adiabatic deposited energy calculation determined that the resulting peak fuel temperature remains below the limiting fuel centerline temperature for melting fuel. The peak RCS pressure also remained below

110% of design. The total heat generated during the transient remains low and results in only a small increase in pressure on the secondary system.

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 initial conditions and CEAs worth are not significantly different between the two plant configurations. Therefore, the power excursion resulting from the bank CEAW and the associated peak local power and minimum DNBR values are similar.

Table 6.3-38
Sequence of Events for CEAW from Subcritical Event

Time (sec)			Value	
3876 MW_t	3990 MW_t	Event	3876 MW_t	3990 MW_t
0.00	0.00	Bank CEAW begins	- -	- -
52.88	52.89	Core power reaches HLPT (% of rated)	0.06	0.06
53.38	53.38	Reactor trip breakers open	- -	- -
53.38	53.38	CEAW stops	- -	- -
53.98	53.98	Scram CEAs begin falling	- -	- -
53.98	53.98	Maximum core power (% of rated)	96.7	94.2
54.15	54.16	Maximum core average heat flux (% of full power heat flux)	17.5	17.1
54.15	54.15	Minimum DNBR	1.61	1.60
54.48	54.42	Maximum RCS pressure (psia)	1881	1888
54.69	54.51	CEAs fully inserted	- -	- -
		Maximum fuel enthalpy (cal/gm)	65.0	65.1
		Peak fuel temperature (^oF)	1730	1730
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the CEAW at power event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

<u>Section 6.3.4.1.3.4</u> <u>Conclusions</u>

The uncontrolled CEAW from a subcritical condition event meets GDC 20 and 25 as specified in SRP Section 15.4.1. These criteria require that the SAFDLs are not exceeded and that protection system action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than the DNBR SAFDL does. The LHGR was found to exceed the Technical Specification limits for a short time; however, a deposited energy calculation determined that the peak fuel centerline temperature remains well below the limiting temperature for melting fuel.

Section 6.3.4.1.4 Uncontrolled Control Element Assembly Withdrawal from a Low Power Condition

Section 6.3.4.1.4.1 Description of Analysis

The NSSS response to a CEAW event from a low power condition was simulated using the CENTS code. The transient DNBR values were calculated using the CETOP-D code that uses the CE-1 CHF correlation. The initial power level corresponds to the HLPT bypass.

Both the existing configuration and PUR were analyzed in order to compare the NSSS response to the CEAW event. Input parameters and initial conditions were selected to maximize local power and DNBR degradation and demonstrate that fuel cladding integrity is maintained throughout the event.

Section 6.3.4.1.4.1.1 Transient Simulation

At time equal zero, the bank CEAW was initiated as specified for the transient. The bank withdrawal continued until a PPS VOPT signal was generated. Following the holding coil delay, the withdrawn bank together with all withdrawn CEAs began to drop back into the core. The magnitude of the negative reactivity is equal to the total magnitude of the reactivity that was being inserted by the withdrawn bank together with the minimum scram worth of the shutdown banks.

Section 6.3.4.1.4.2 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-39 contains the initial conditions used for the HZP CEAW event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The reactivity worth versus position associated with the inward motion of the withdrawn CEAs during reinsertion is assumed the same worth at a given position that the CEAs had during their withdrawal. This is conservative since a higher flux will be present during the scram that would result in a higher CEA worth than during the withdrawal.
3. There is no operator action for the first 30 minutes of the event.

Table 6.3-39
Parameters Used for the CEAW from HZP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	0.19E-04	0.19E-04
Initial core inlet temperature (°F)	572	572
Initial pressurizer pressure (psia)	2000	2100
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	0.5E-04	0.5E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Reactivity insertion rate (%$\Delta\rho$/in)	0.040	0.040
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6530	6530
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.4.1.4.3 Results

Table 6.3-40 presents a sequence of events that occur following a CEAW from HZP conditions. Figure 6.3-161 through Figure 6.3-166 present the behavior of NSSS parameters following a CEAW from HZP conditions.

The CEAW at HZP conditions results in a gradual increase in core power until significant supercriticality occurs. A reactor trip on PPS VOPT is generated because of the rapid escalation in power when the core reaches this supercritical state. Following trip, the CEAs begin dropping into the core terminating the power escalation. The hot channel minimum DNBR remains above the safety limit. The LHGR exceeded the safety limit as defined in the Technical Specifications. However, this occurs for a short time with a resulting peak fuel temperature well below the limiting fuel centerline temperature for melting fuel. The peak RCS pressure also remained below 110% of design. The total heat generated during the transient remains low and results in only a small increase in pressure on the secondary side.

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 initial conditions and CEA worth are not significantly different between the two plant configurations. Therefore, the power excursion resulting from the bank CEAW and the associated peak local power and minimum DNBR values are similar.

Table 6.3-40
Sequence of Events for CEAW from HZP Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	CEAW begins	- -	- -
21.45	21.45	Core power reaches PPS VOPT (% of rated)	12.0	12.0
21.90	21.89	Reactor trip breakers open	- -	- -
21.90	21.89	CEAW stops	- -	- -
22.49	22.49	Scram CEAs begin falling	- -	- -
22.50	22.49	Maximum core power occurs (% of rated)	79.4	77.4
22.70	22.70	Minimum DNBR	1.43	1.45
22.70	22.70	Maximum core average heat flux to coolant (% of full power heat flux)	30.1	29.4
22.94	22.96	Maximum RCS pressure (psia)	2129	2232
26.70	26.70	CEAs fully inserted	- -	- -
		Maximum fuel enthalpy (cal/gm)	105.3	107.9
		Peak fuel temperature (^oF)	2645	2695
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the CEAW at power event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.4.1.4.4 Conclusions

The uncontrolled CEAW from HZP condition event meets GDC 20 and 25 as specified in SRP Section 15.4.1. These criteria require that the SAFDLs are not exceeded and that protection system action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than the DNBR SAFDL. The LHGR was found to exceed the Technical Specification limits for a short time; however, a deposited energy calculation determined that the peak fuel centerline temperature remains well below the limiting temperature for melting fuel.

Figure 6.3-155
Uncontrolled CEAW from Subcritical - Core Power vs. Time

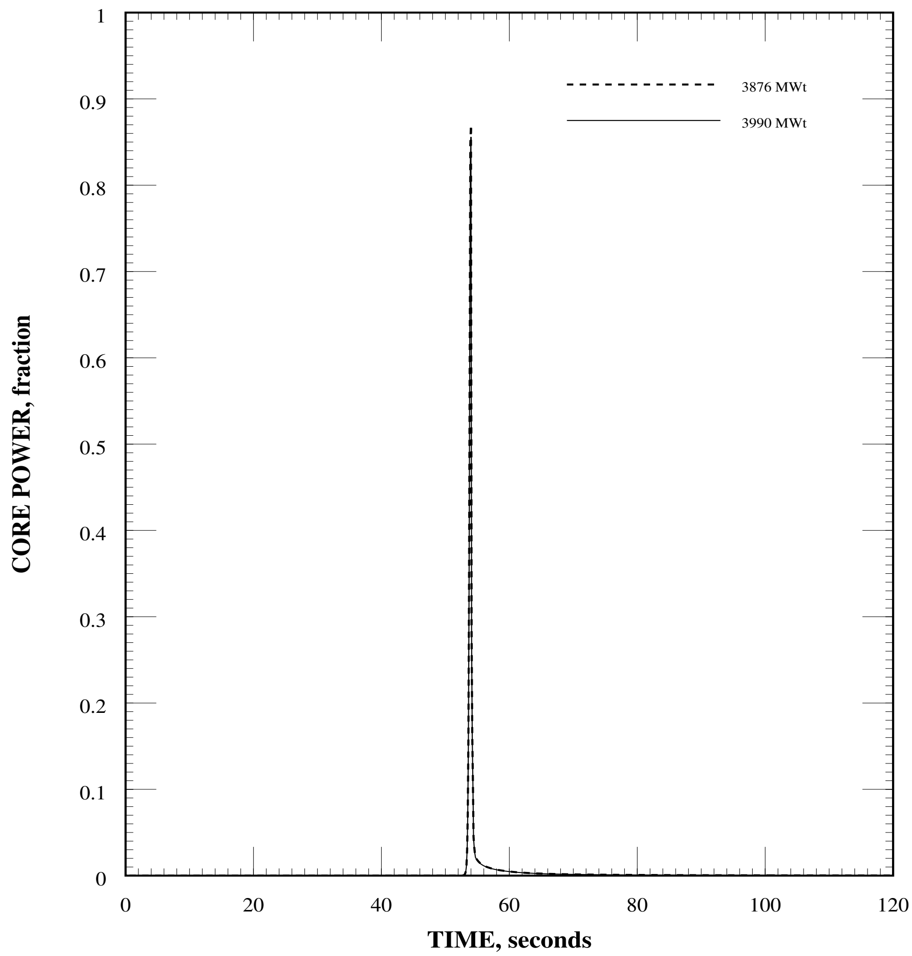


Figure 6.3-156
Uncontrolled CEAW from Subcritical - Core Heat Flux vs. Time

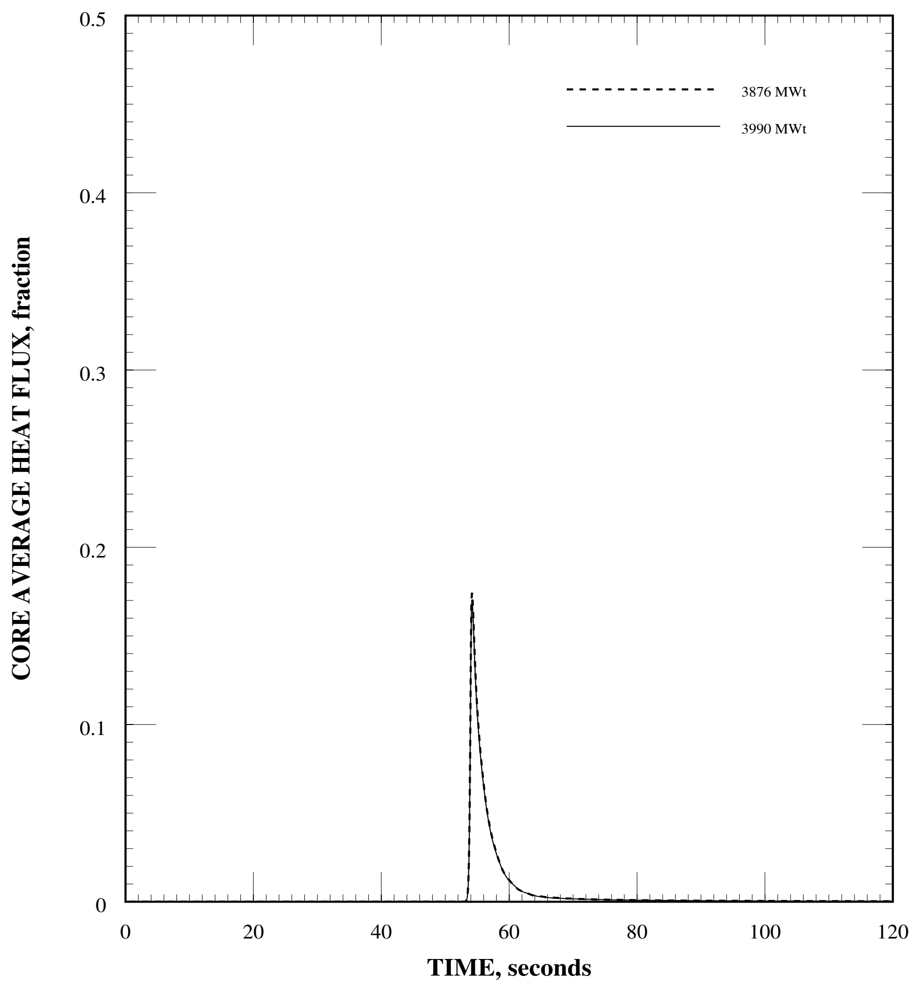


Figure 6.3-157
Uncontrolled CEAW from Subcritical - RCS Average Temperature vs. Time

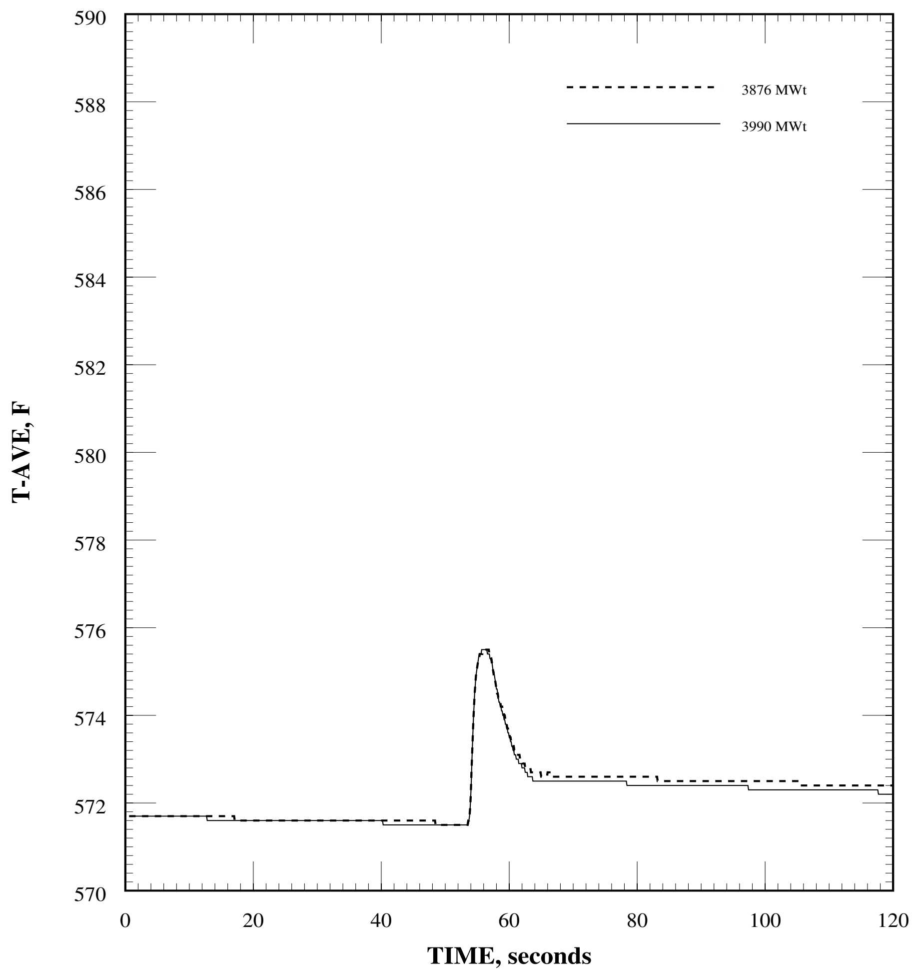


Figure 6.3-158
Uncontrolled CEAW from Subcritical - RCS Pressure vs. Time

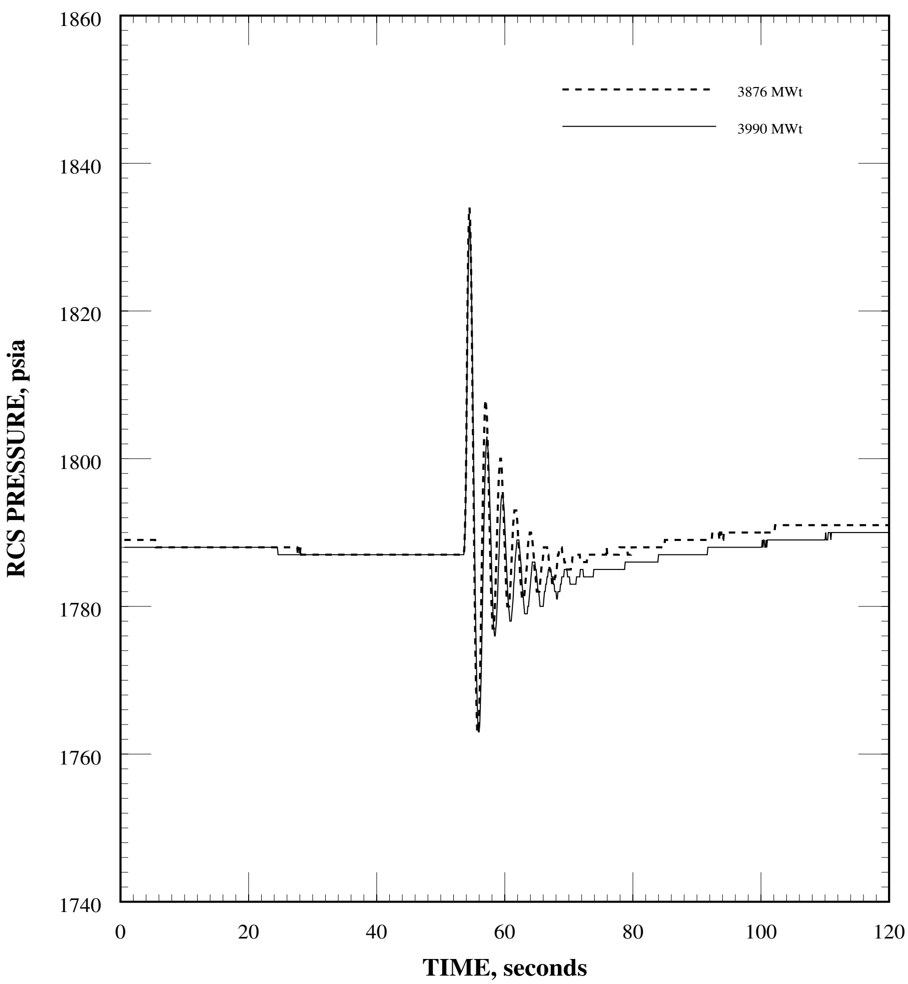


Figure 6.3-159
Uncontrolled CEAW from Subcritical - Total Reactivity vs. Time

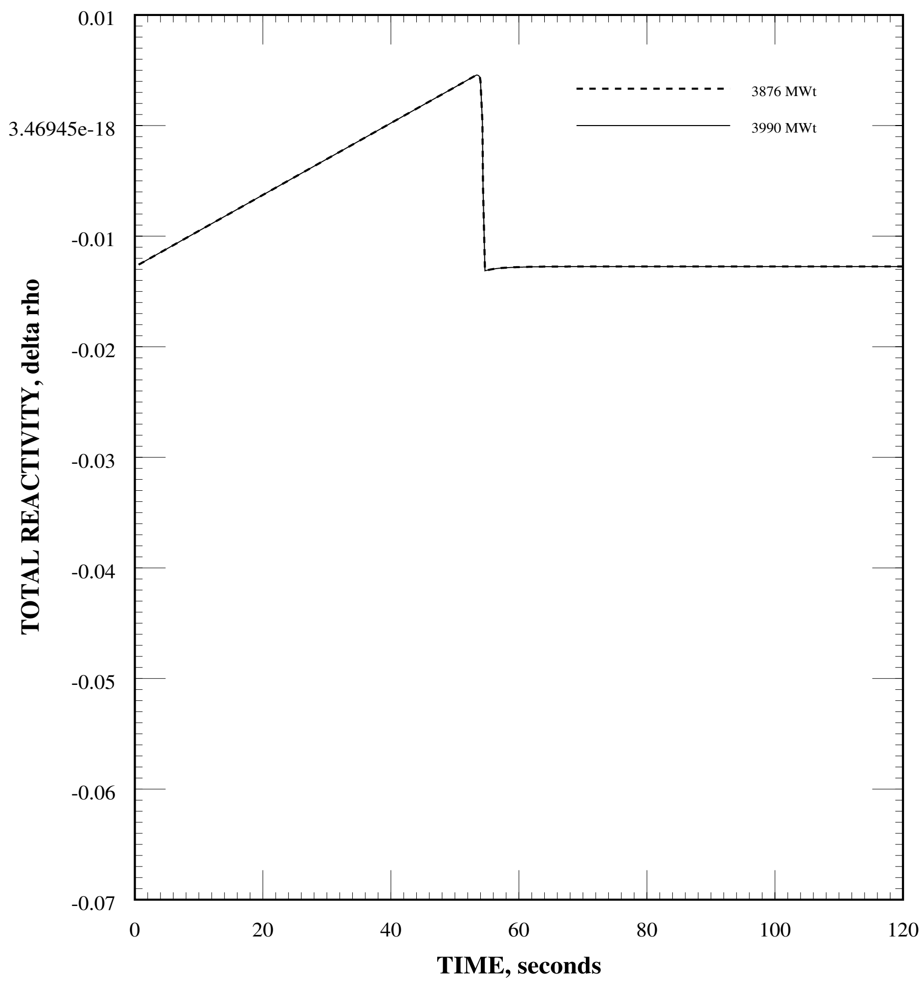


Figure 6.3-160
Uncontrolled CEAW from Subcritical - Doppler Reactivity vs. Time

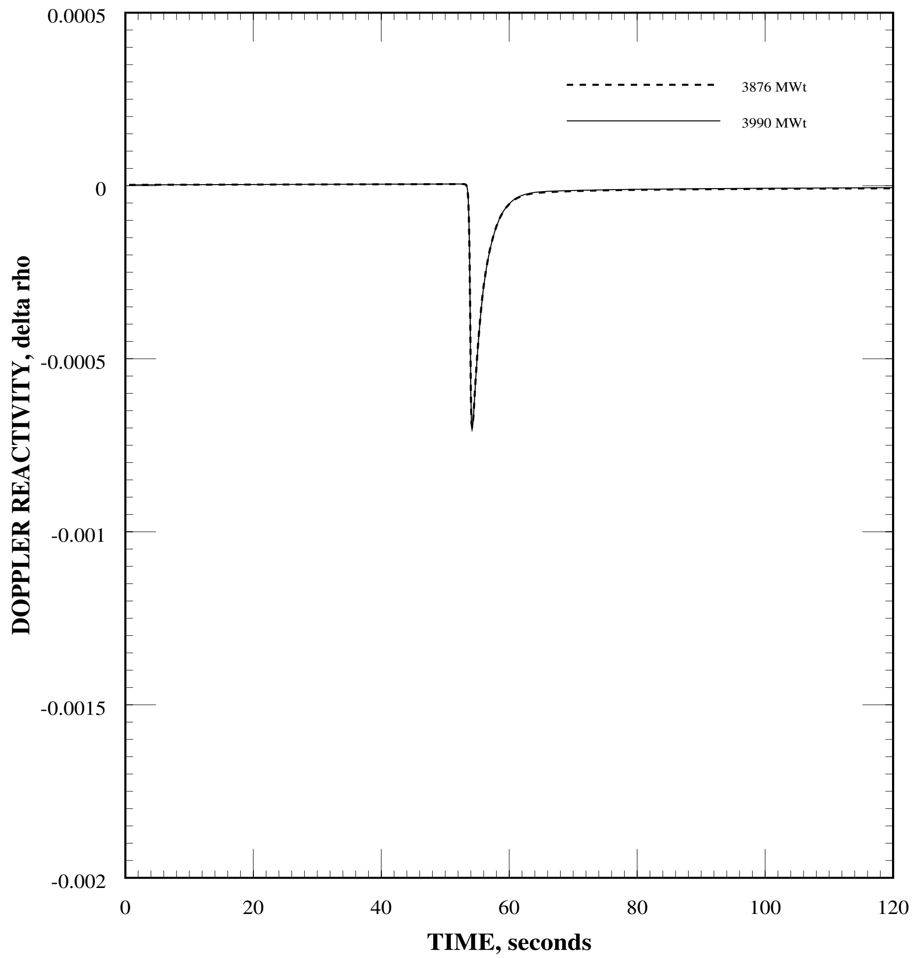


Figure 6.3-161
Uncontrolled CEAW from HZP - Core Power vs. Time

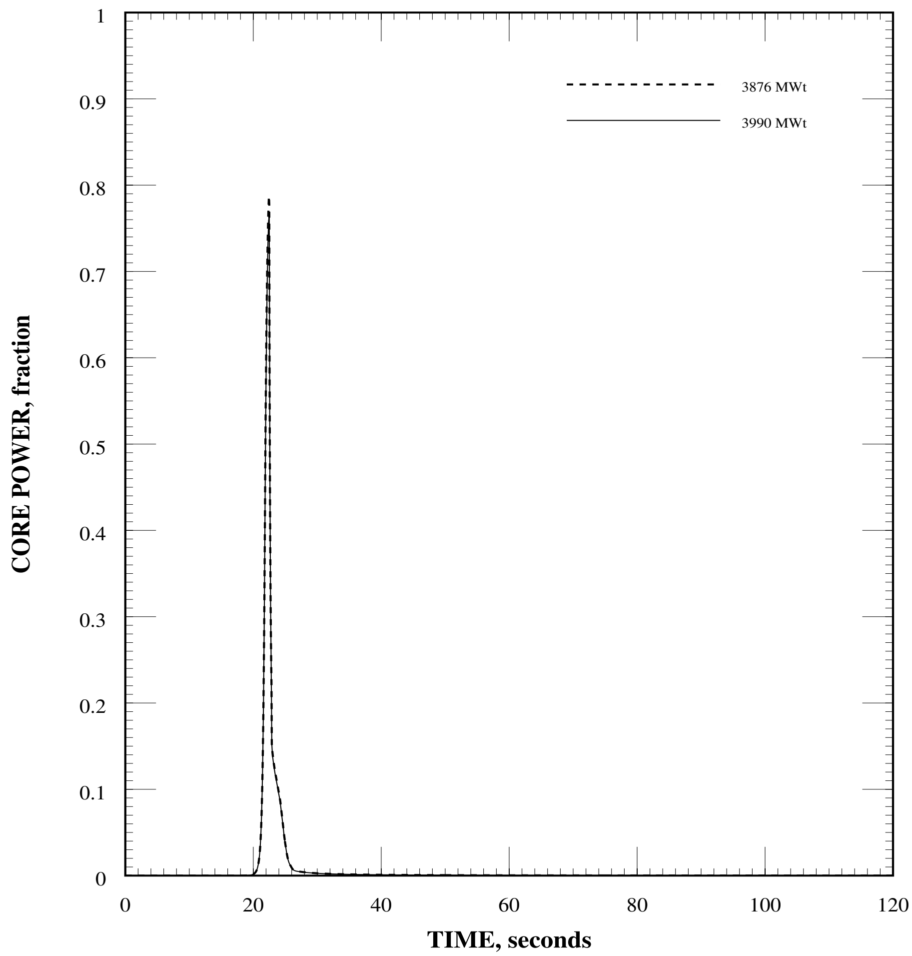


Figure 6.3-162
Uncontrolled CEAW from HZP - Core Heat Flux vs. Time

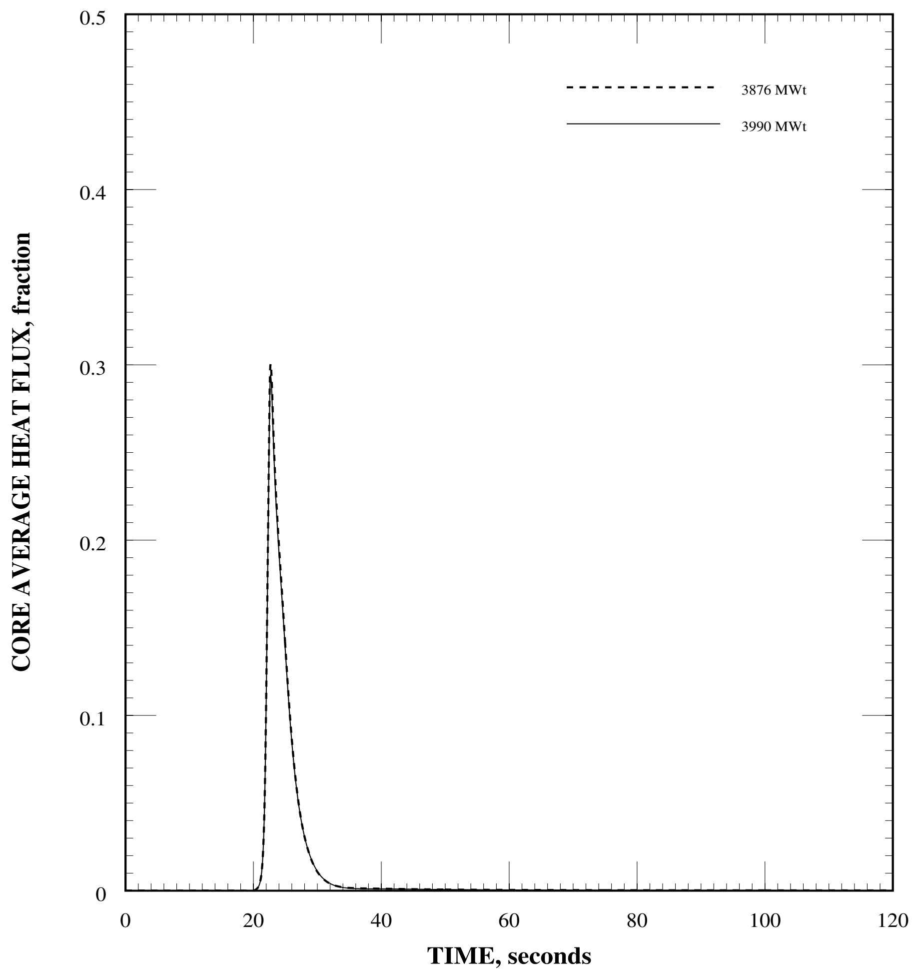


Figure 6.3-163
Uncontrolled CEAW from HZP - RCS Average Temperature vs. Time

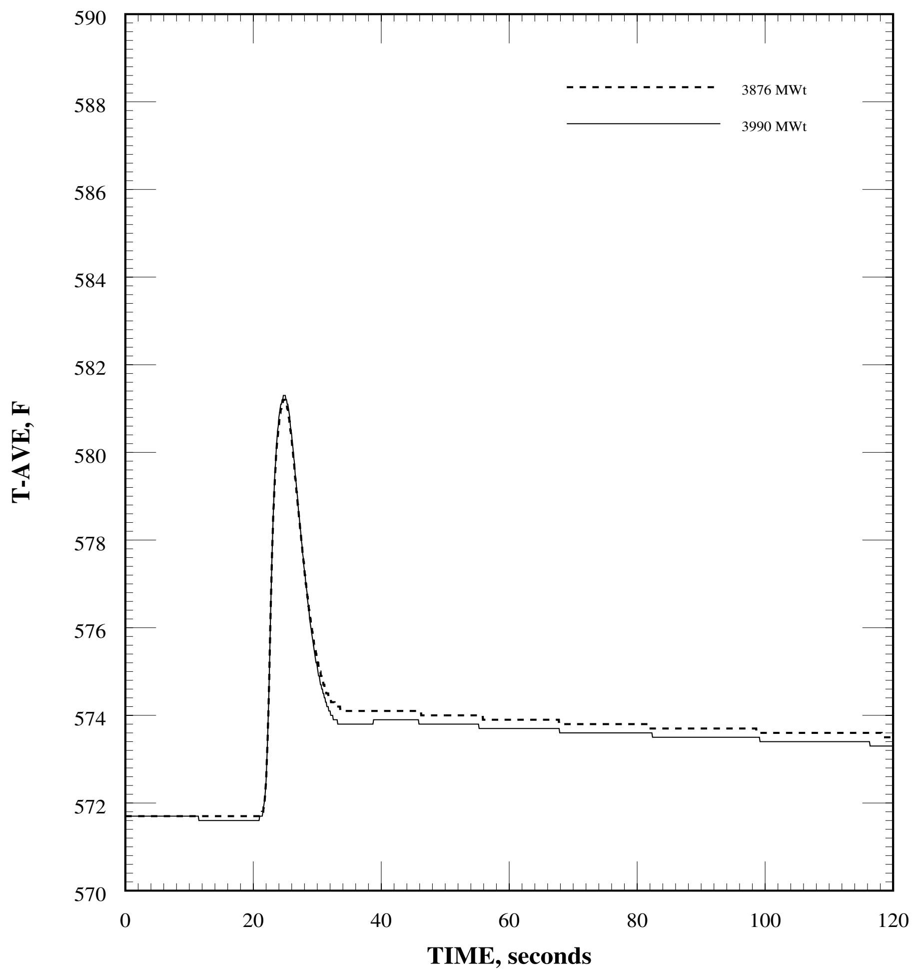


Figure 6.3-164
Uncontrolled CEAW from HZP - RCS Pressure vs. Time

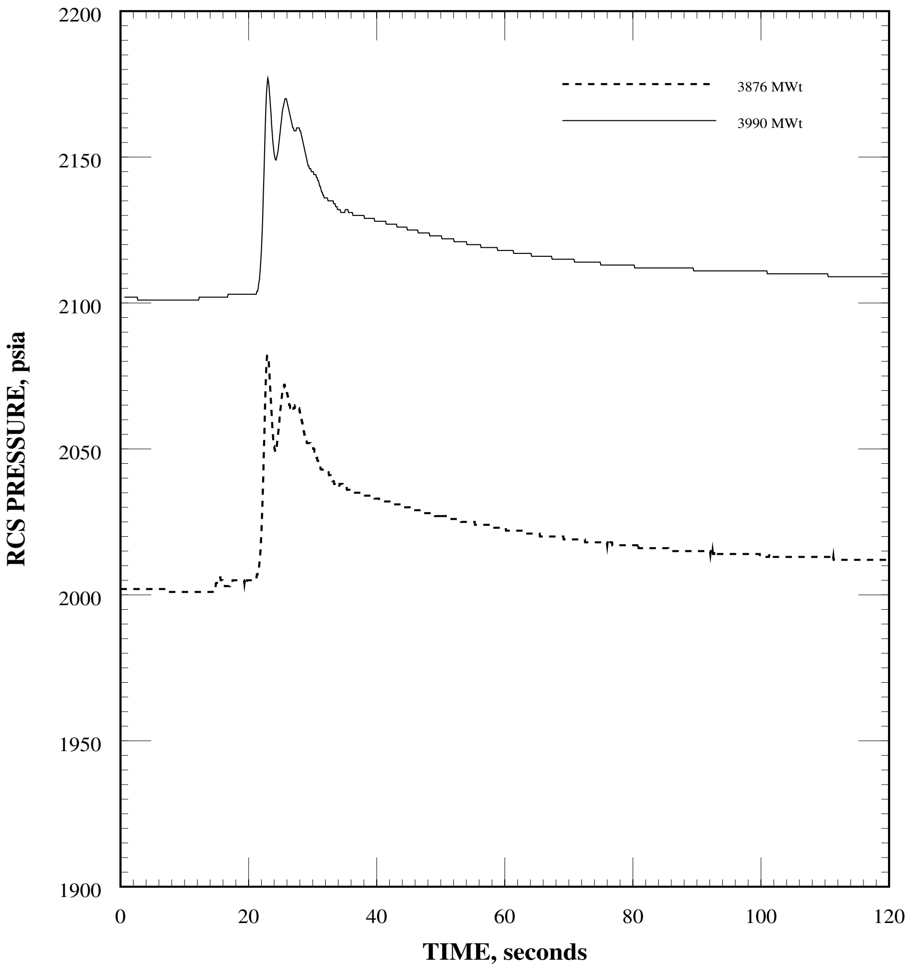


Figure 6.3-165
Uncontrolled CEAW from HZP - Total Reactivity vs. Time

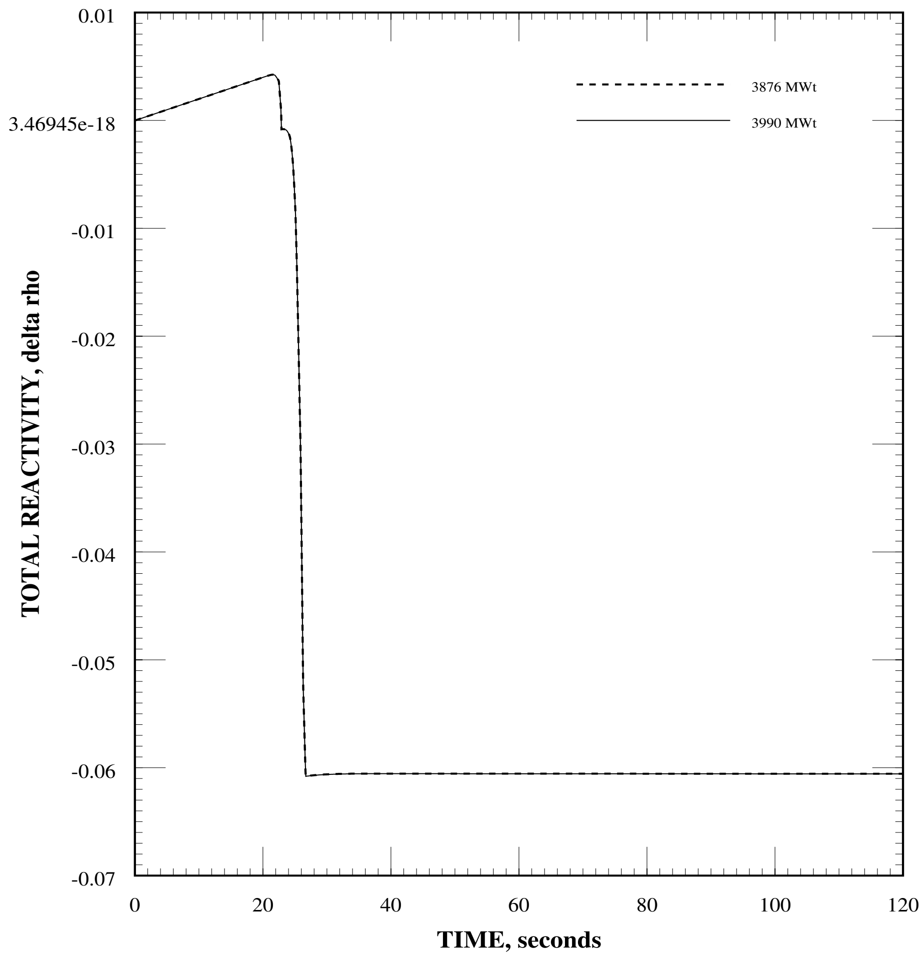
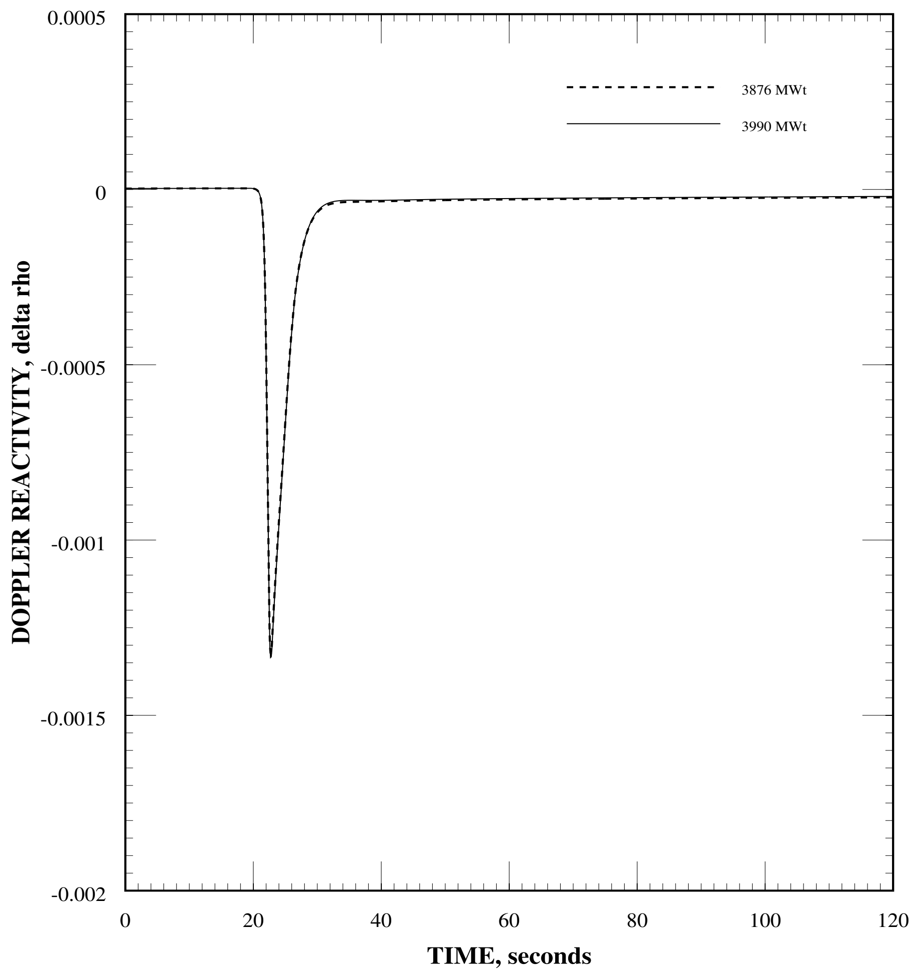


Figure 6.3-166
Uncontrolled CEAW from HZP - Doppler Reactivity vs. Time



Section 6.3.4.2 Uncontrolled Control Element Assembly Withdrawal at Power

Section 6.3.4.2.1 Identification of Event and Causes

As described in UFSAR Section 15.4.2, an uncontrolled sequential CEAW is assumed to occur because of a single failure in the CEDM, CEDMCS, RRS, or because of operator error. The transient results in an approach to a SAFDL on LPD and DNBR. A reactor trip will be generated on either CPC VOPT or CPC DNBR.

Section 6.3.4.2.2 Acceptance Criteria

As defined in the SRP Section 15.4.2, the specific acceptance criteria for this event are:

The following GDC apply:

a. Criterion 20 that requires that the RPS action be initiated automatically.
b. Criterion 25 that requires that the RPS be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

The following fuel design limits serve as the acceptance criteria for this event:

a. Minimum DNBR should remain above SAFDL.
b. Fuel temperature and clad strain limits consistent with the acceptance criteria of SRP 4.2 should not be exceeded. For steady state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate. For PWRs, a steady-state linear heat generation rate of 20-22 kW/ft would result in a centerline fuel temperature equal to or less than the melting point of UO_2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

Section 6.3.4.2.3 Description of Analysis

The NSSS response to a CEAW at power was simulated using the CENTS code. CPC's response was modeled using the CPC FORTRAN code. Acceptable results lead to the conclusion that the SAFDL for DNBR or LHR is not exceeded.

Section 6.3.4.2.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-41 contains the initial conditions used for the CEAW event.

The following assumptions were 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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences (i.e., maximum power increase).

2. there is no operator action for the first 30 minutes of the event.

Table 6.3-41
Parameters Used for the CEAW at Power Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temperature (°F)	548	548
Initial pressurizer pressure (psia)	2100	2100
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	0.0E-04	0.0E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Reactivity insertion rate (%$\Delta\rho$/in)	0.008	0.008
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6527	6527
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.4.2.5 Results

The behavior of the NSSS parameters following an uncontrolled CEAW at power is presented in Figure 6.3-167 through Figure 6.3-177. Table 6.3-42 presents the sequence of events associated with the CEAW. The tuning of the CPCS provides adequate response upon initiation of the event. Acceptable response leads to a CPC trip and fulfillment of design criteria. These criteria require that the SAFDLs are not exceeded and the protection system is initiated automatically. The CEAW from full power conditions meets the following fuel design limits that serve as the acceptance criteria for this event:

1. the transient terminates with a hot channel minimum DNBR greater than or equal to the limit and
2. peak linear heat generation rate during the transient is less than the SAFDL.

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 initial conditions and CEAs worth are not significantly different between the two plant configurations. Therefore, the power excursion resulting from the bank CEAW and the associated peak local power and minimum DNBR values are similar.

Table 6.3-42
Sequence of Events for the CEAW at Power Event

Time (sec)			Value	
3876 MW_t	3990 MW_t	Event	3876 MW_t	3990 MW_t
0.00	0.00	Withdrawal of CEAs - initiating event	- -	- -
12.82	13.02	CPC trip signal generated	- -	- -
13.57	13.77	Reactor trip breakers open	- -	- -
13.57	13.77	Turbine trip occurs	- -	- -
13.58	13.78	Maximum core power (% of rated)	110.9	110.9
14.17	14.37	Scram CEAs begin falling	- -	- -
14.33	14.49	Maximum core average heat flux (% of full power heat flux)	109.8	109.7
16.96	17.21	Maximum pressurizer pressure (psia)	2263	2268
23.96	22.82	MSSV bank 1 begins to cycle open/closed (psia)	1227	1227
24.51	23.01	Maximum secondary pressure (psia)	1228	1229
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.4.2.6 Conclusions

The uncontrolled CEAW at power event meets GDC 20 and 25 as specified in SRP Section 15.4.2. These criteria require that the SAFDLs are not exceeded and that RPS action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than the DNBR SAFDL and a peak LHGR less than the LHGR SAFDL.

Figure 6.3-167
Uncontrolled CEAW at Power - Core Power vs. Time

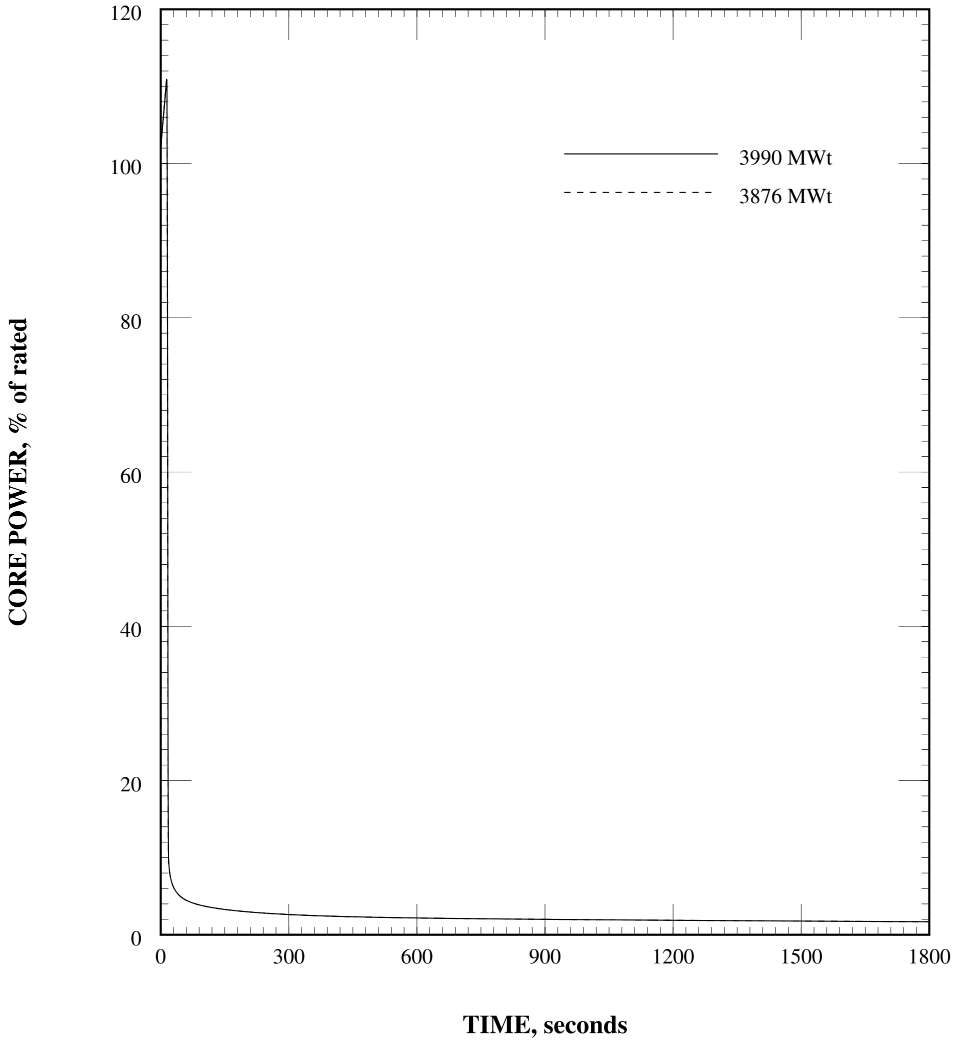


Figure 6.3-168
Uncontrolled CEAW at Power - Core Average Heat Flux vs. Time

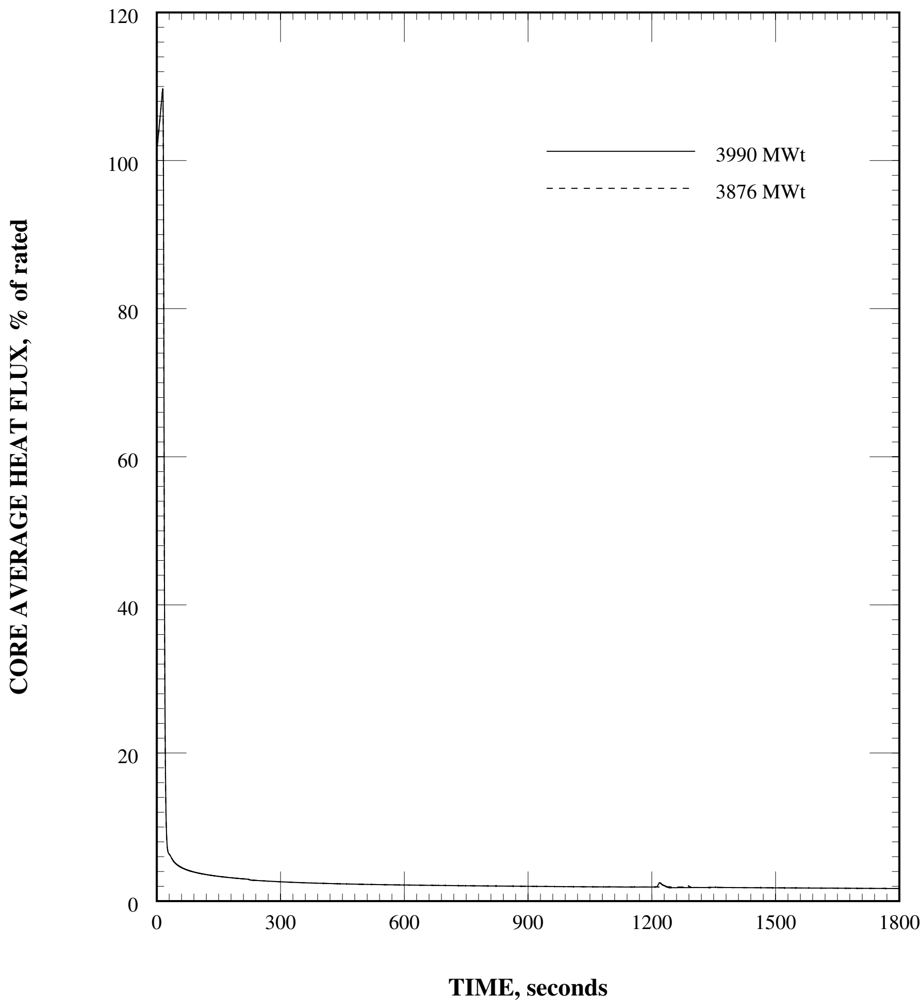


Figure 6.3-169
Uncontrolled CEAW at Power - Pressurizer Pressure vs. Time

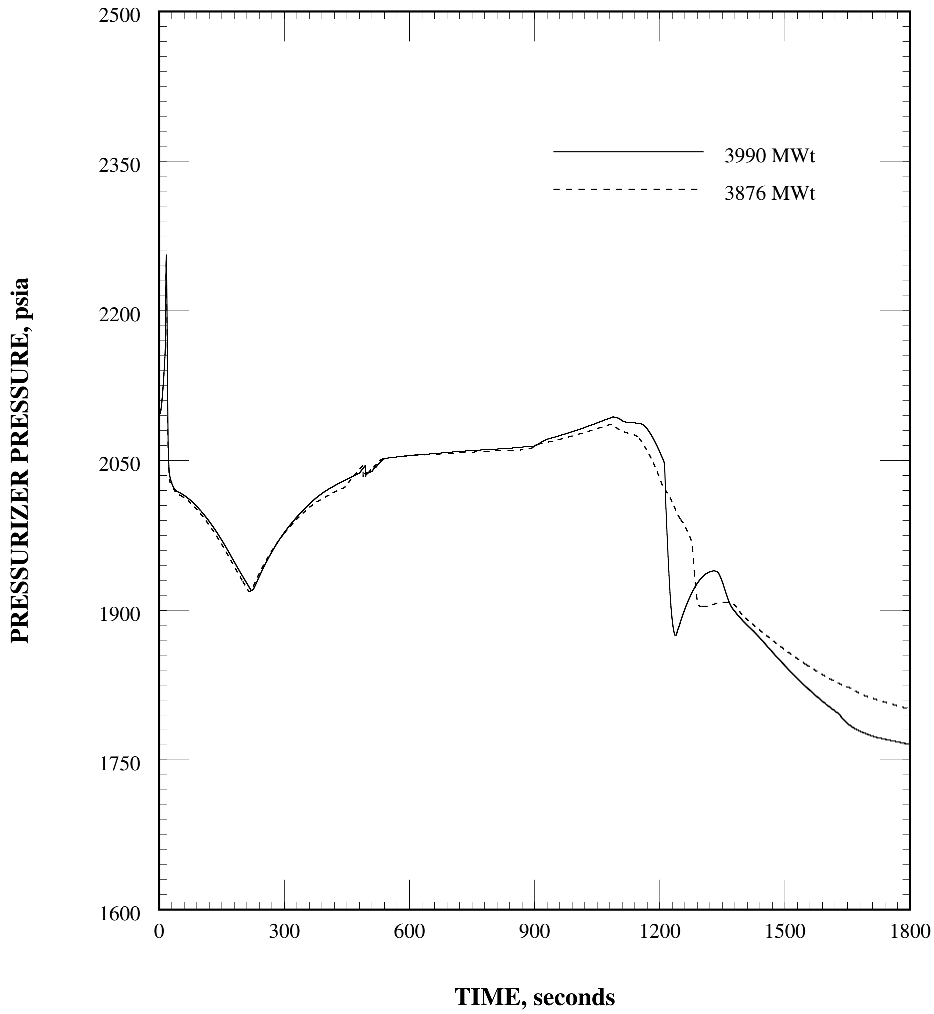


Figure 6.3-170
Uncontrolled CEAW at Power - RCS Temperature vs. Time

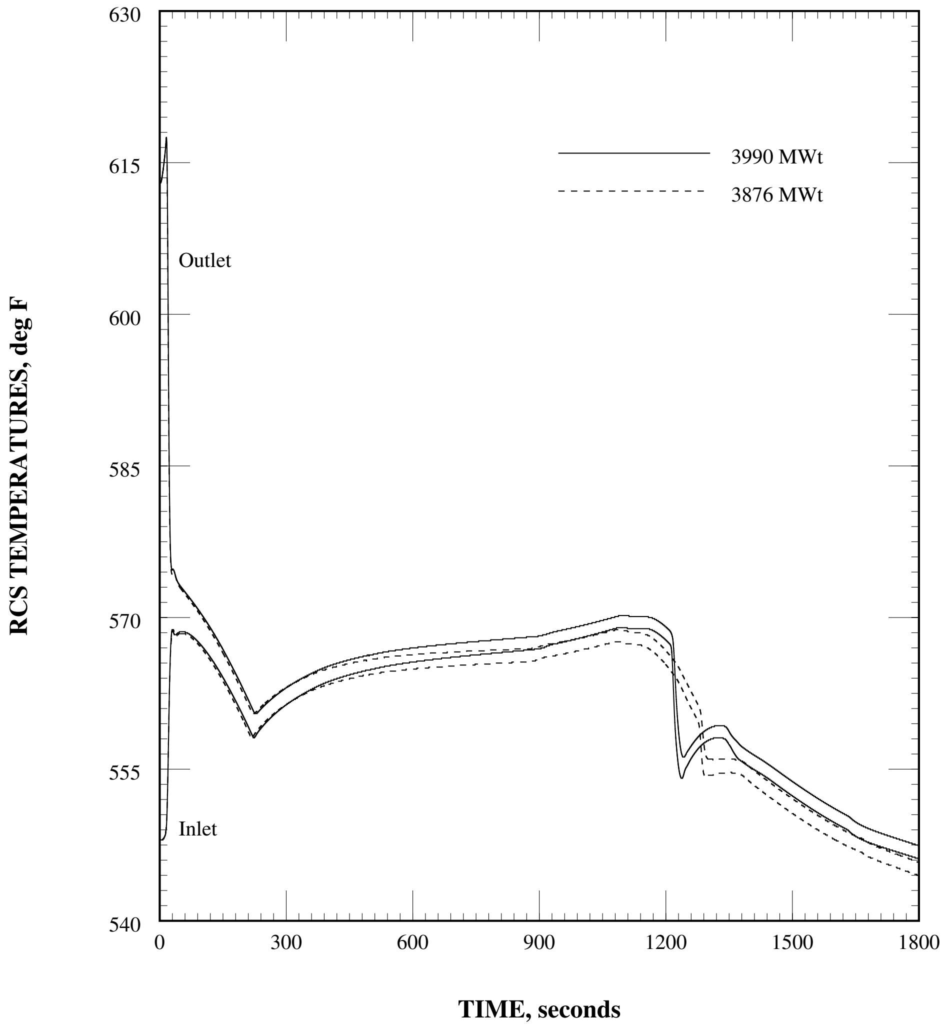


Figure 6.3-171
Uncontrolled CEAW at Power - FW Flow vs. Time

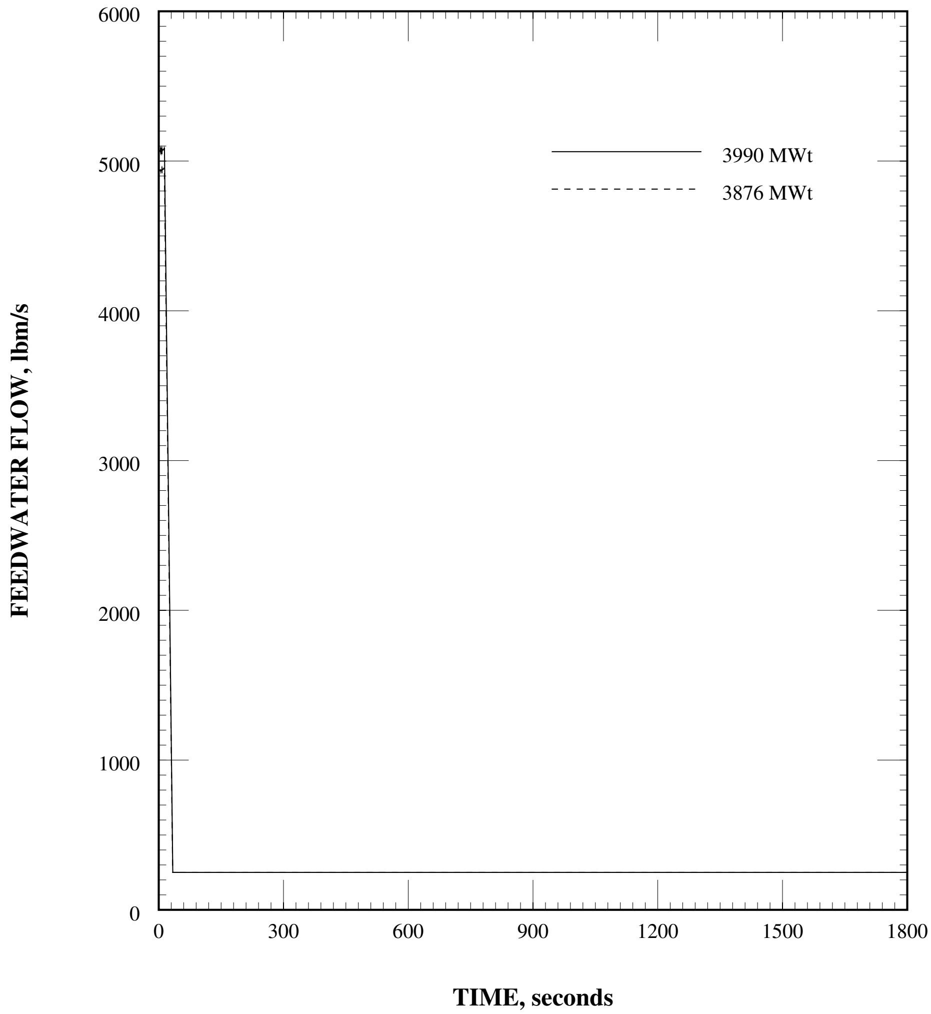


Figure 6.3-172
Uncontrolled CEAW at Power - FW Enthalpy vs. Time

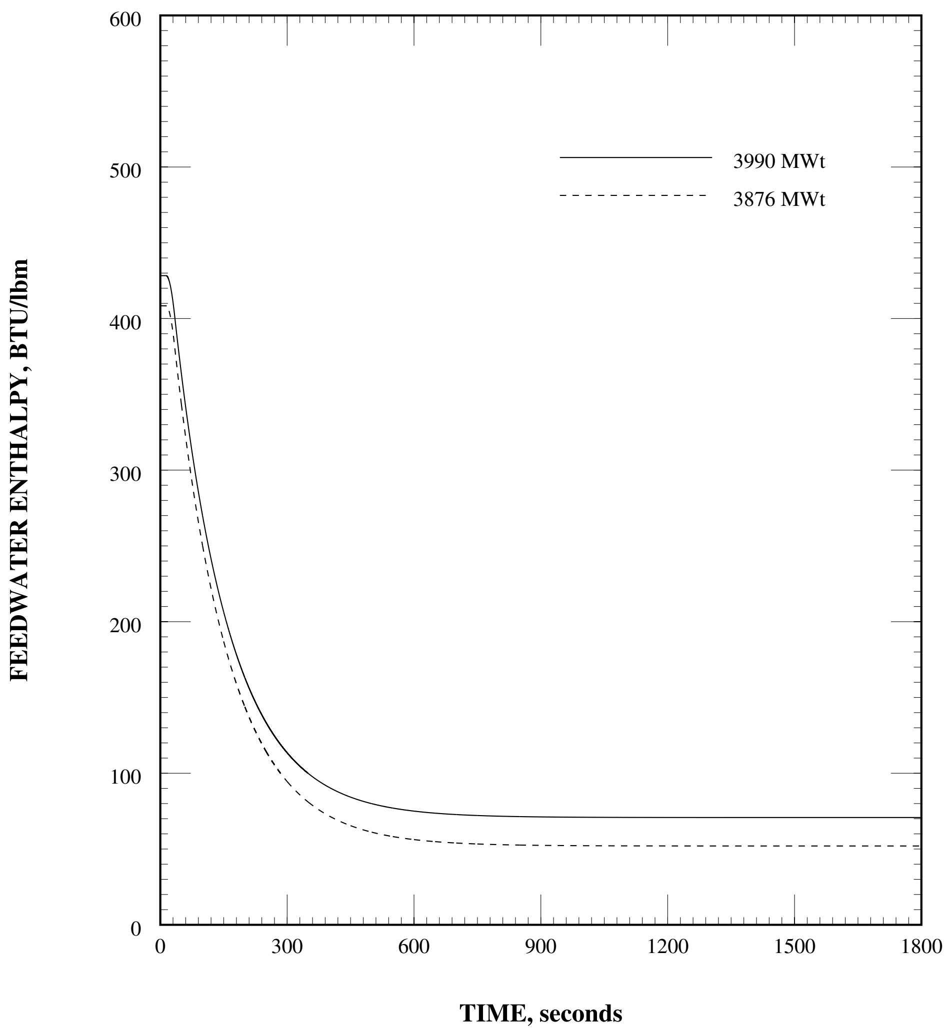


Figure 6.3-173
Uncontrolled CEAW at Power - MSSV Flow vs. Time

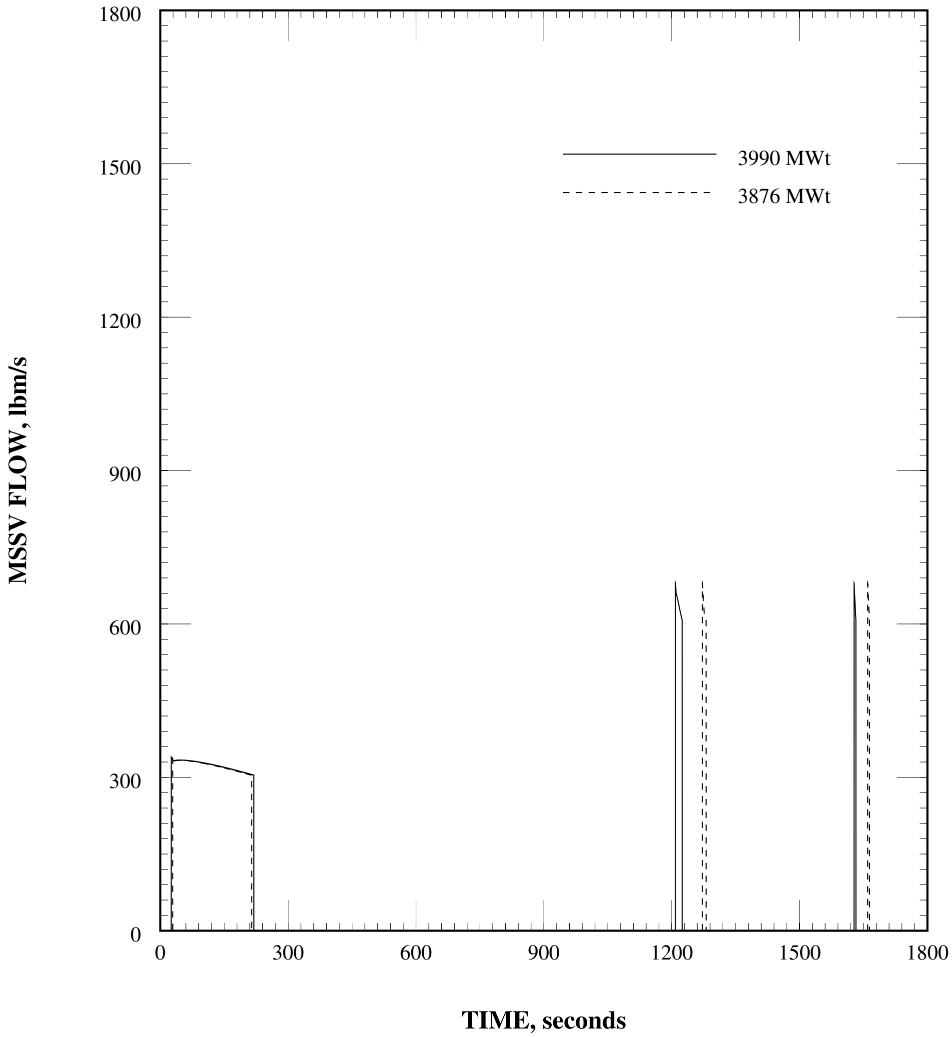


Figure 6.3-174
Uncontrolled CEAW at Power - SG Flow vs. Time

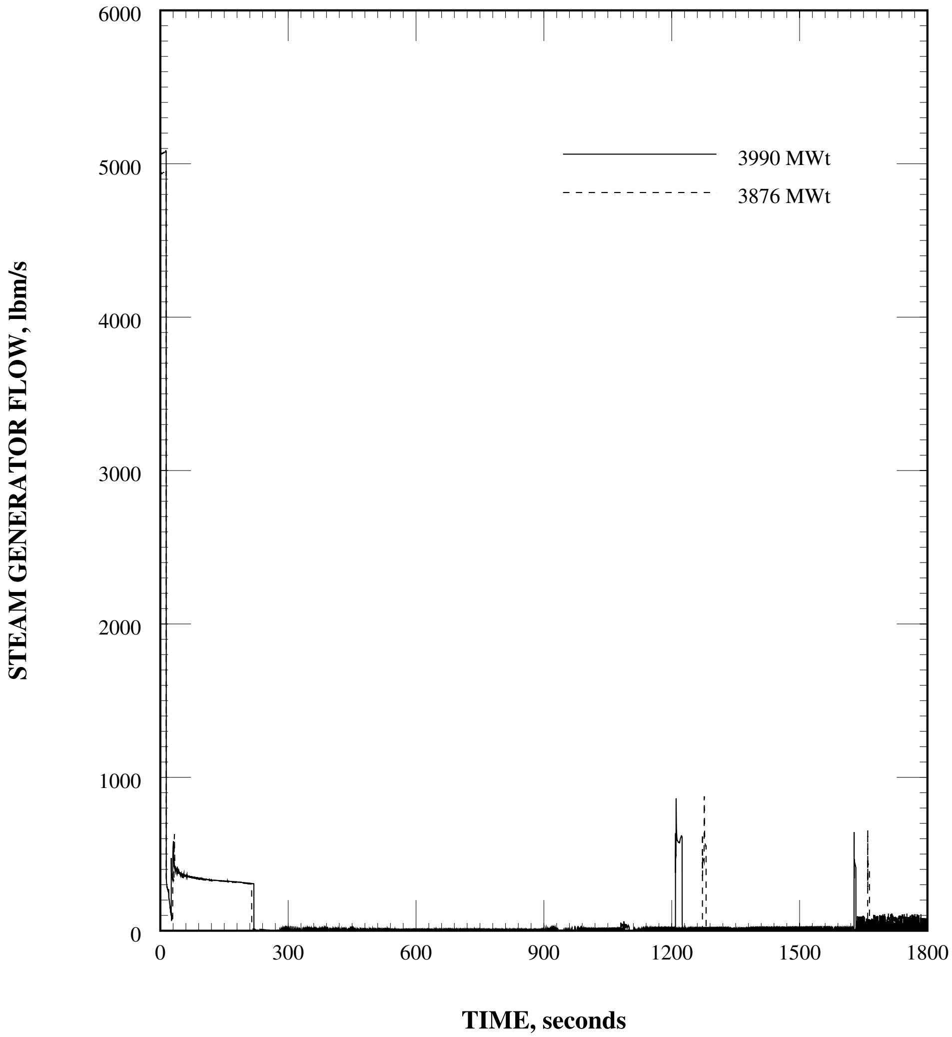


Figure 6.3-175
Uncontrolled CEAW at Power - SG Pressure vs. Time

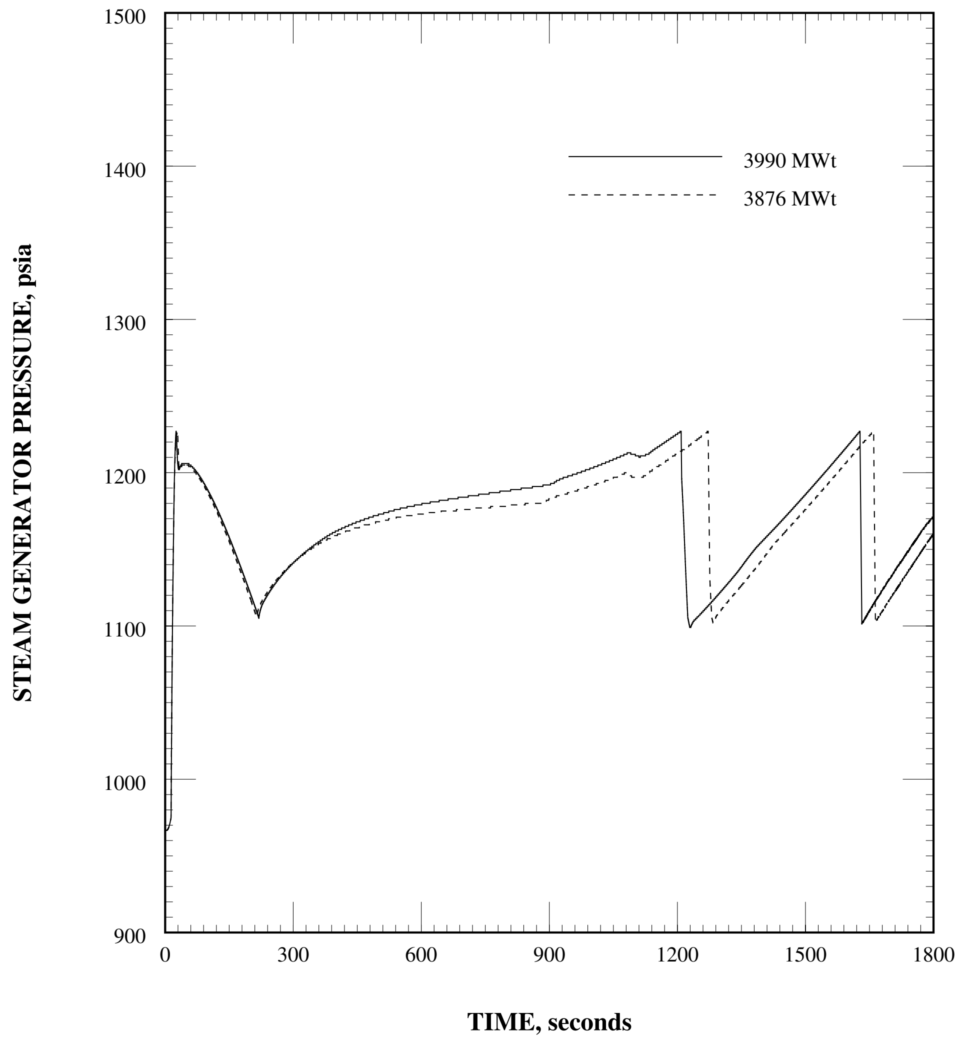


Figure 6.3-176
Uncontrolled CEAW at Power - DNBR vs. Time

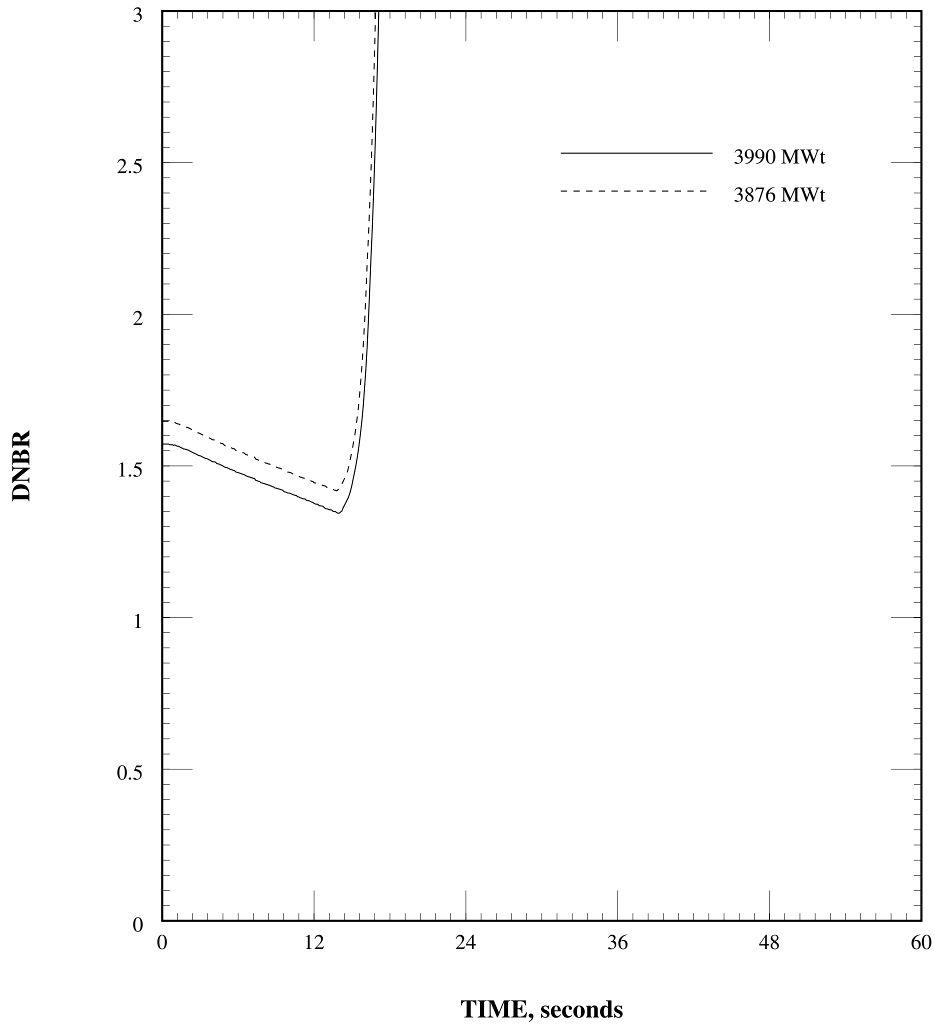
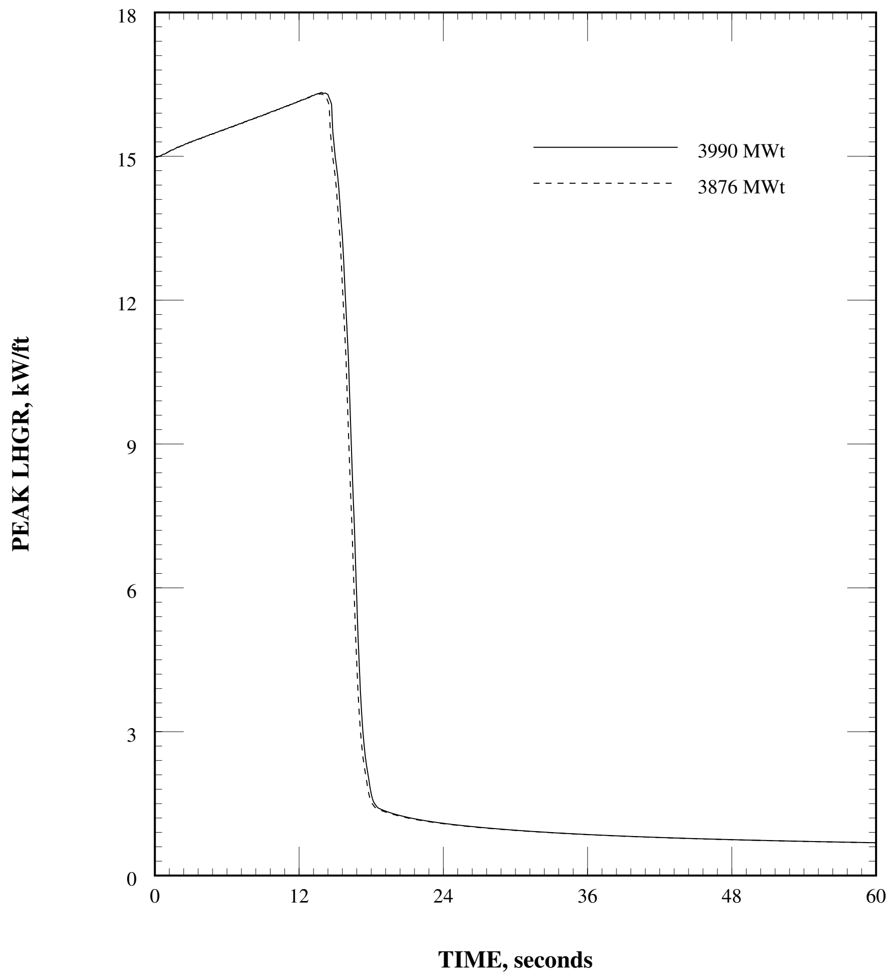


Figure 6.3-177
Uncontrolled CEAW at Power - Peak LHGR vs. Time



Section 6.3.4.3 Single Full Length Control Element Assembly Drop

Section 6.3.4.3.1 Identification of Event and Causes

As described in UFSAR Section 15.4.3, a single full-length CEA drop results from an interruption in the electrical power to the CEDM holding coil of a single full-length CEA. This interruption can be caused by a holding coil failure or loss of power to the holding coil. The limiting case is the CEA drop which does not cause a trip to occur but results in an approach to the DNBR SAFDL.

Acceptable results for all 4-finger CEA drops are ensured by the initial thermal margin preserved by the LCOs, and do not rely upon CEA position penalty factors contained within the CPC's calculations. The CEA position-related penalty factors for downward deviations of 12-fingered CEAs are calculated such that the CPCS will provide a trip when necessary. A part-length Power-Dependent Insertion Limit (PDIL) also restricts the part-length CEA insertion to less than 25% for power levels greater than 50%. From these initial conditions, the part-length single or subgroup drop inserts only negative reactivity (similar to a full-length single or subgroup drop event). For CEA subgroup drops, the CEA position-related penalty factors for downward deviations are used by the CPCS to provide a trip when necessary.

Section 6.3.4.3.2 Acceptance Criteria

As defined in the SRP Section 15.4.2, the specific acceptance criteria for this event are:

The following GDC apply:

a. Criterion 20 that requires that the RPS action be initiated automatically.
b. Criterion 25 that requires that the RPS be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

The following fuel design limits serve as the acceptance criteria for this event:

a. Minimum DNBR should remain above SAFDL.
b. Fuel temperature and clad strain limits consistent with the acceptance criteria of SRP 4.2 should not be exceeded. For steady state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate. For PWRs, a steady-state linear heat generation rate of 20-22 kW/ft would result in a centerline fuel temperature equal to or less than the melting point of UO_2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

Section 6.3.4.3.3 Description of Analysis

Representative NSSS response to the single full-length CEA drop was simulated using the CENTS code. The CETOP-D code utilizing CE-1 CHF correlation was employed to calculate the equivalent power change corresponding to the axial and radial power distortion.

Each reload cycle relies on hand calculations to verify acceptable results for a Full-Length Control Element Assembly Drop (FLCEAD). This is acceptable since the major effect considered to degrade thermal margin comes from the radially distorted power. A maximum radial distortion factor including Xenon redistribution resulting from a FLCEAD is obtained from the reload physics calculation. The ratio of pre- and post-drop radial distortion is converted to the equivalent power ratio (the required margin) by the POL partial derivative for radial distortion factor. The maximum value of the POL partial derivative within LCO parameters is used to maximize the required margin that must be reserved by COLSS or the other LCOs. The same methodology was used to analyze 12-finger and subgroup CEA drops when both CEACs are out of service. These margin analyses are performed each cycle as part of the reload analysis.

<u>Section 6.3.4.3.4</u> <u>Input Parameters, Initial Conditions, and Assumptions</u>

Table 6.3-43 contains the initial conditions used for the FLCEAD event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The RRS is assumed to be in the automatic mode. For this analysis, the choice of mode is inconsequential because there would be no regulating bank motion if the system were in manual mode; and in the automatic mode, the CEAW Prohibit (CWP), actuated on the DNBR pretrip signal, prevents the motion of any regulating bank following the drop of a single full-length CEA which could cause the CPC calculated minimum DNBR to approach the DNBR SAFDL.
3. At 15 minutes, the operator will take action to reduce power in accordance with the Technical Specifications, if the misaligned CEA has not been realigned.

Table 6.3-43
Parameters Used for the Full Length CEA Drop Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	95	95
Initial core inlet temperature (°F)	nominal	nominal
Initial pressurizer pressure (psia)	nominal	nominal
Initial RCS flow (% of design)	nominal	nominal
Initial pressurizer level (ft)	nominal	nominal
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-4.2E-04	-4.2E-04
FTC	least negative	least negative
Kinetics	maximum β	maximum β
Dropped CEA worth (%$\Delta\rho$)	-0.0015	-0.0015
CEA drop radial distortion factor	1.143	1.143
CEA drop time (sec)	1.0	1.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	1620	1620
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.4.3.5 Results

The release and subsequent drop of a full-length CEA initiated the transient. The resultant increase in the hot pin radial peaking factor coupled with a return to initial power (following a temporary power depression) results in a minimum DNBR greater than the DNBR SAFDL at approximately 900 seconds.

Table 6.3-44 presents a chronological list of events that occur during the single full-length CEA drop transient, from initiation to the attainment of steady state conditions. The behavior of the NSSS parameters following a CEA drop is presented in Figure 6.3-178 through Figure 6.3-194.

A minimum DNBR of greater than the DNBR SAFDL is obtained at 900 seconds, as determined from the initial radial power peaking increase following CEA drop plus Xenon redistribution at the final coolant conditions. At this time, the operator will take action to reduce power in accordance with the Technical Specifications, if the misaligned CEA has not been realigned. A maximum allowable initial linear heat generation rate of 18.0 kW/ft could exist as an initial linear heat rate condition without exceeding the acceptable fuel centerline melt limit.

Table 6.3-44
Sequence of Events for the Full Length CEA Drop Event

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	CEA begins to drop into core	---	---
1.0	1.0	CEA reaches fully inserted position	---	---
1.1	1.1	Core power level reaches minimum and begins to increase due to reactivity feedback (% of rated)	82.71	82.68
25.0	25.0	Minimum pressurizer pressure (psia)	2215	2213
25.0	25.0	Core power returns to maximum (% of rated)	91.0	91.0
900	900	Minimum DNBR is approached	>SAFDL	>SAFDL
900	900	Operator action - core power reduced if dropped CEA not aligned (min)	15	15

Section 6.3.4.3.6 Conclusions

The full-length CEA drop event meets GDC 20 and 25 as specified in SRP. These criteria require that the SAFDLs are not exceeded and that RPS action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than the DNBR SAFDL and a peak LHGR less than the LHGR SAFDL.

Figure 6.3-178
Full Length CEA Drop - Core Power vs. Time

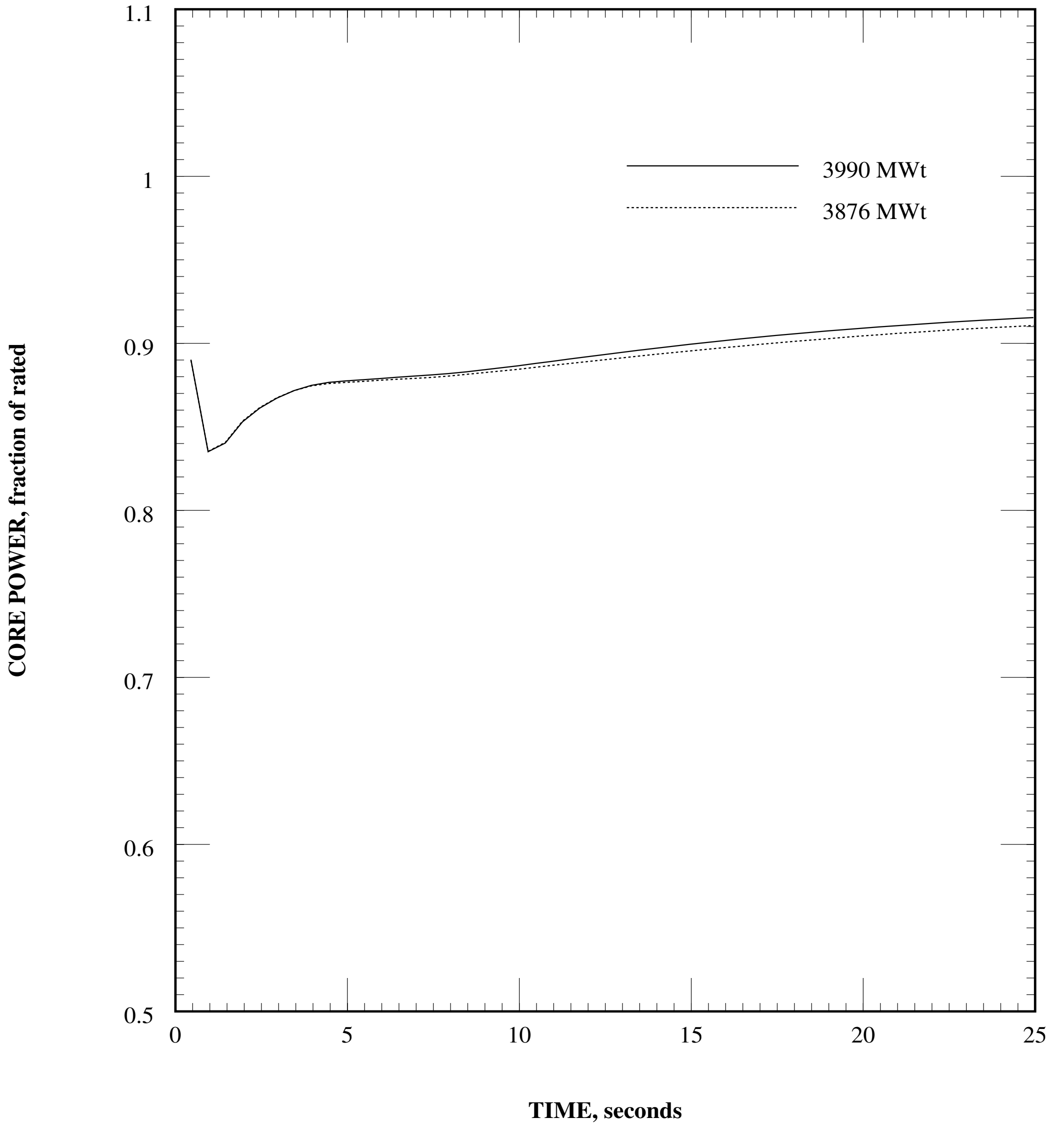


Figure 6.3-179
Full Length CEA Drop - Core Average Heat Flux vs. Time

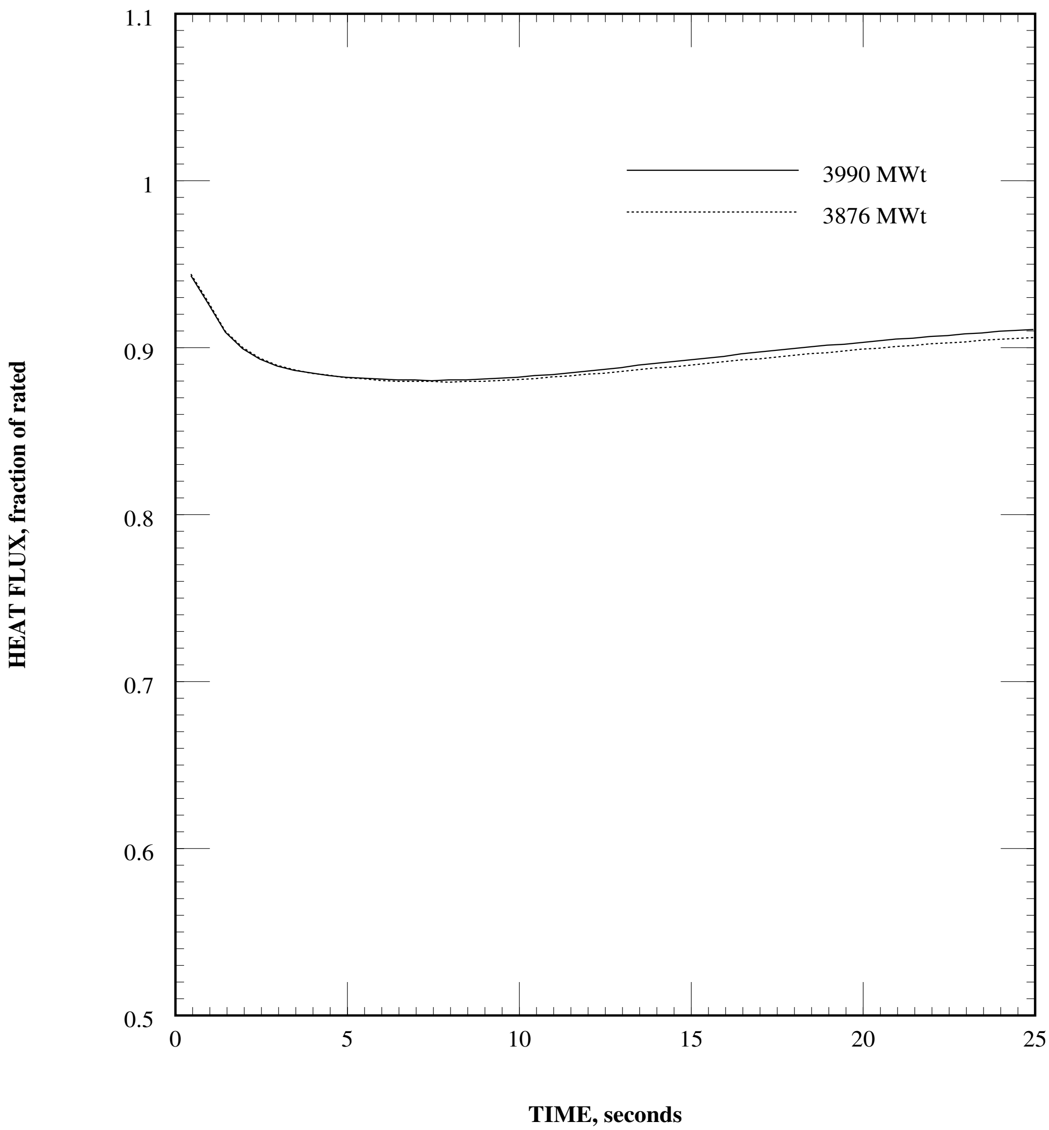


Figure 6.3-180
Full Length CEA Drop - RCS Temperature vs. Time

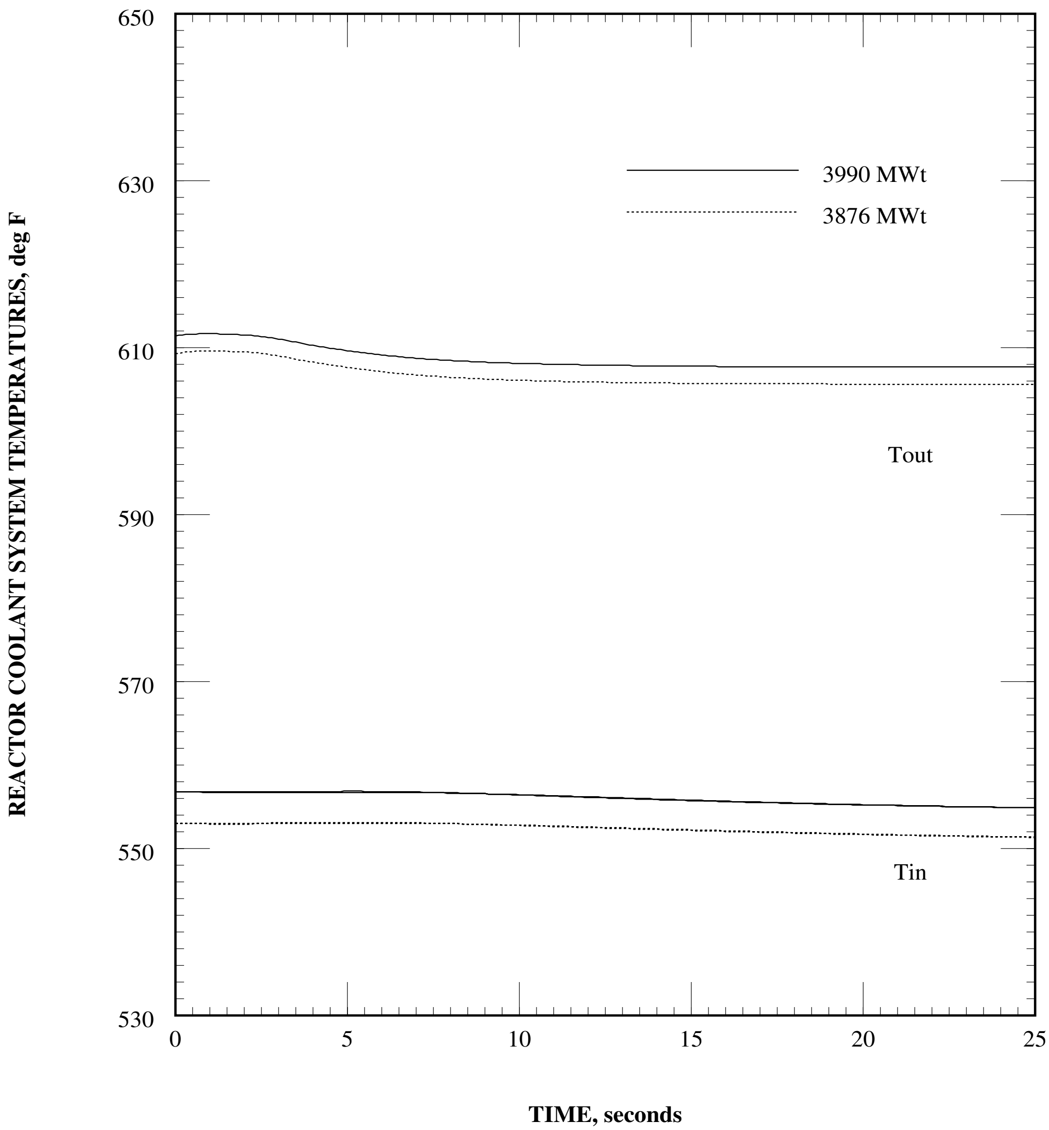


Figure 6.3-181
Full Length CEA Drop - Pressurizer Pressure vs. Time

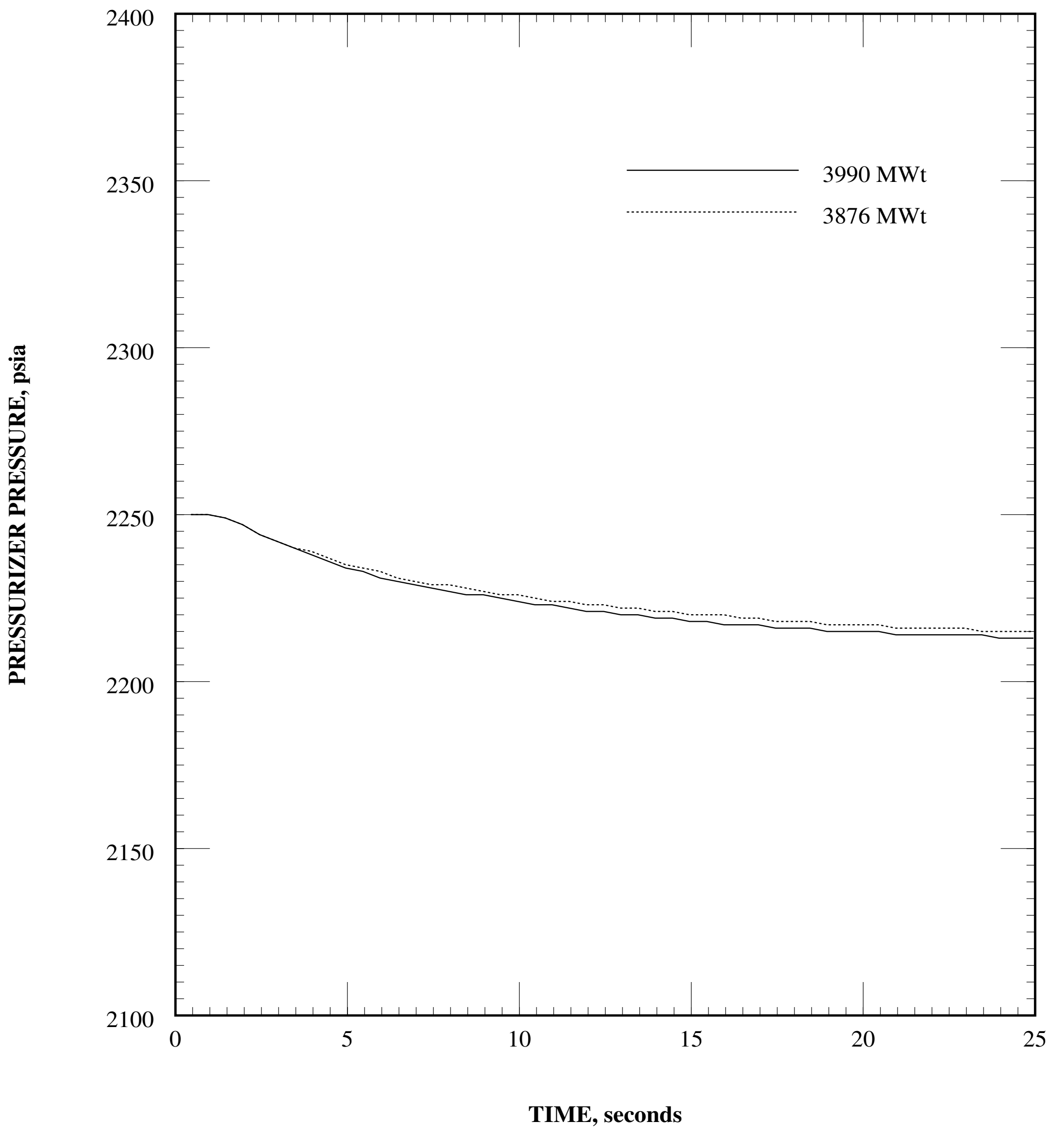


Figure 6.3-182
Full Length CEA Drop - Core Reactivity vs. Time

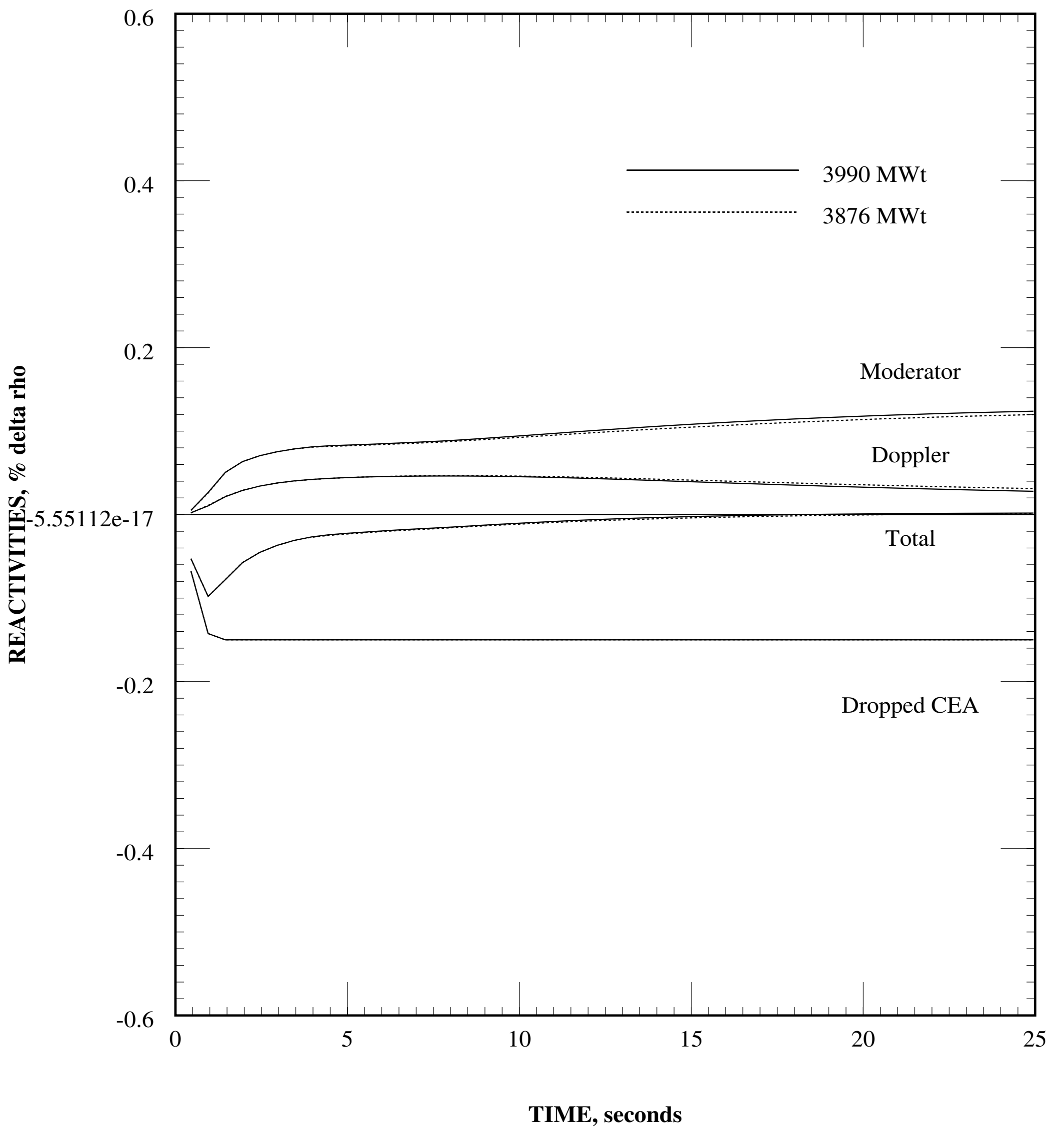
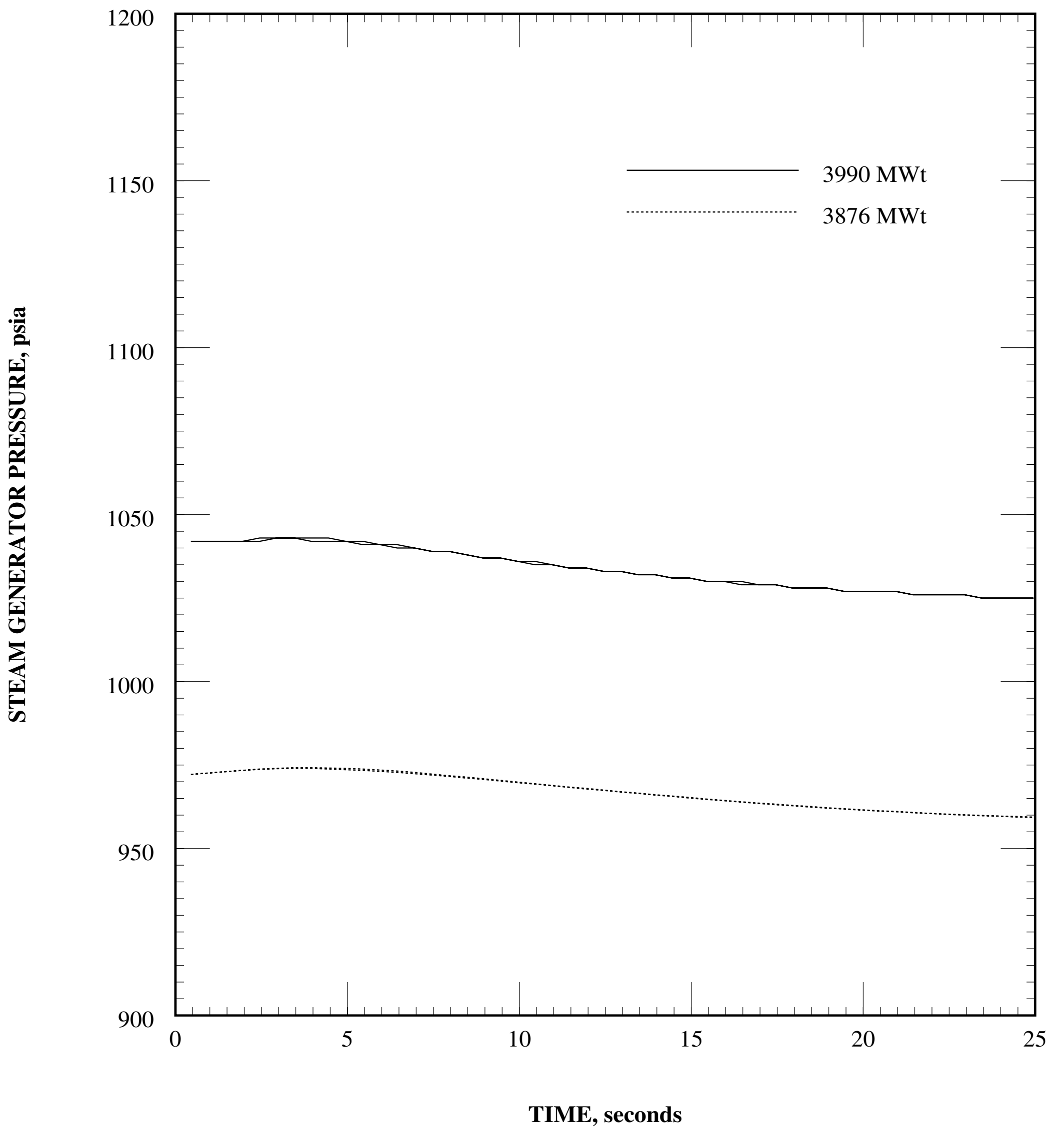


Figure 6.3-183
Full Length CEA Drop - SG Pressure vs. Time



Section 6.3.4.4 Startup of an Inactive Reactor Coolant Pump

As described in UFSAR Section 15.4.4, the Startup of an Inactive RCP (SIRCP) may result in a potential loss of subcriticality. This event also has the potential to challenge RCS pressure and fuel performance criteria. Administrative procedures govern the starting of RCPs and reduce the effects of RCP starts.

The startup of an inactive RCP event is classified as an AOO. This event was reviewed for the impacts of PUR. None of the impacts associated with PUR would necessitate re-analysis of this event. This event remains bounded by the AOR.

Section 6.3.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow

As described in UFSAR Section 15.4.5, this event is not applicable to PWRs and, therefore, is not included in this submittal.

Section 6.3.4.6 Inadvertent Deboration

As described in UFSAR Section 15.4.6, the ID event is presented here for the time available for operator corrective action before the reactor achieves criticality. Fuel integrity is not challenged by this event.

The ID event may be caused by improper operator action or by a failure in the boric acid makeup flow path that reduces the flow of borated water to the charging pump suction. Either cause can produce a boron concentration of the charging flow that is below the concentration of the RCS. The ID event is classified as an incident of moderate frequency.

This evaluation shows that Mode 5 (cold shutdown) with the RCS drained results in the least time available for detection and termination of an ID event. The combination of lowered RCS volume and three operating charging pumps results in a small dilution time constant and the fastest dilution rate, and therefore yields the shortest time interval between initiation of an ID event and the reactor achieving criticality.

Since RCS boron concentration is maintained under strict procedural controls, the probability of a sustained and erroneous dilution due to operator error is very low.

The ID event is classified as an AOO. This event was reviewed for the impacts of PUR. The PUR has no impact on the ID event. The increased RCS inventory due to larger SGs results in a longer dilution time constant that increases the time that the operator has to perform corrective actions. This event remains bounded by the AOR.

Section 6.3.4.7 Inadvertent Loading of a Fuel Assembly into the Improper Position

As described in UFSAR Section 15.4.7, the inadvertent loading of a fuel assembly into the improper position event is initiated by interchanging two fuel assemblies. The likelihood of an error in core loading is considered extremely remote because of the strict procedural control used during core loading.

The inadvertent loading of a fuel assembly into the improper position event is classified as an AOO. This event was reviewed for the impacts of PUR. The PUR will neither increase the probability of an inadvertent loading of a fuel assembly event or the consequences of this event. This event remains bounded by the AOR.

Section 6.3.4.8 Control Element Assembly Ejection

Section 6.3.4.8.1 Identification of Event and Causes

As described in UFSAR Section 15.4.8, CEA ejection event results from a circumferential rupture of the CEDM housing or of the CEDM nozzle. The CEA ejection may lead to a rapid positive reactivity addition resulting in a rapid power excursion. A reactor trip on CPC and/or PPS VOPT occurs in a few seconds.

Section 6.3.4.8.2 Acceptance Criteria

As defined in the SRP Section 15.4.8, the specific acceptance criteria for this event are:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.
2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the Service Level C as defined in the ASME Code.

The radiological consequences of the SRP Section 15.4.8 states:

- Higher doses may be acceptable at the Operating License review stage, up to the guidelines of 10 CFR Part 100.

Section 6.3.4.8.3 Description of Analysis

Section 6.3.4.8.3.1 Fuel Performance Case

Fuel enthalpy (as well as temperature) rises rapidly because of the CEA ejection. The power excursion is turned around quickly due to the effect of Doppler feedback followed by the insertion of scram CEAs after reactor trip. The STRIKIN-II code was used to simulate response of the fuel during the transient and was used to determine the energy deposition in the fuel.

In order to maximize the power excursion, the largest worth CEA was ejected. The CE-1 model in STRIKIN-II was utilized to calculate the minimum DNBR during the transient.

Section 6.3.4.8.3.2 Peak Reactor Coolant System Pressure Case

CENTS was utilized to simulate the NSSS response during the CEA ejection transient and calculate the maximum RCS pressure. The tables and figures provided in this section detail the CENTS peak pressure simulation. The input parameters and initial conditions were selected to maximize the RCS pressure, and demonstrate that the peak RCS pressure remains within 120% of the design pressure.

Section 6.3.4.8.3.3 Transient Simulation

The system is initialized at 102% power using the most limiting initial parameters. At time equal zero, the CEA was ejected and exits the core in 0.05 seconds. Reactor trip occurs on HPPT that is followed by turbine trip. Although not credited, an immediate CPC and/or RPS VOPT occur. The lifting of PSVs and MSSVs, at their most adverse lift setpoints follows the HPPT.

Section 6.3.4.8.4 Input Parameters, Initial Conditions, and Assumptions

Regulatory Guide 1.77 (Reference 6-26) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident.

Table 6.3-45 contains the initial conditions used for the peak RCS pressure CEA ejection event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. Only the HPPT is credited. Although a PPS or CPCs VOPT may occur much earlier than the HPPT, no credit is taken for this trip.
3. For the purposes of evaluating fuel performance and peak RCS pressure, it was assumed that the CEDM rupture was plugged by the ejected CEA. For calculating radiological consequences, the CEDM rupture was not assumed plugged and provides a direct release path into containment.
4. There is no operator action for the first 30 minutes of the event.

Table 6.3-45
Parameters Used for the CEA Ejection Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temp (°F)	548	548
Initial pressurizer pressure (psia)	2100	2100
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	11.4	11.4
Initial SG level (ft)	24.5	25.7
MTC (Δρ/°F)	0.0E-04	0.0E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
Ejected CEA worth (%Δρ)	0.157	0.157
Postulated CEA ejection time (sec)	0.05	0.05
Scram worth at trip, N-2 (%Δρ)	5.5	5.5
Fuel rod gap conductance (Btu/hr-ft^2-°F)	6984	6984
Plugged SG tubes (% of tubes/SG)	0	0
PSV tolerance	+3%	+3%
MSSV tolerance	+3%	+3%
Single failure	none	none
LOP	no	no

Section 6.3.4.8.5 Results

Fuel Performance Case

A radially averaged fuel specific enthalpy is found to be less than 280 cal/gm at the hottest axial location of the hot fuel pin and the fuel centerline enthalpy is less than 250 cal/gm. Thus, fuel rod integrity, based upon enthalpy, is maintained throughout the event.

The hot channel minimum DNBR temporarily drops below the SAFDL during the brief power excursion. Due to the short duration within DNB, no DNB propagation is predicted to occur. The number of fuel rod failures, based upon DNBR, is limited to ensure acceptable dose consequences. Section 6.4.4.1 details the dose calculation.

Primary Peak Pressure Case

Table 6.3-46 presents a sequence of events that occur following the CEA ejection until operator action is initiated for the primary peak pressure cases. Figure 6.3-184 through Figure 6.3-195 show the NSSS response to the CEA ejection event under the most adverse initial and transient conditions that maximize RCS peak pressure.

The sudden reduction of steam flow caused by the turbine trip leads to a reduction of the primary-to-secondary heat transfer and a resultant RCS and secondary system temperature and pressure increase. Coupled with the power excursion, the rapid heatup of the RCS results in a reactor trip on high pressurizer pressure. The PSVs open, maintaining primary pressure below 120% of the design value.

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 ejected CEA worth is similar between the two plant configurations. Therefore, the power excursion resulting from the ejected CEA and the associated fuel enthalpy, minimum DNBR, and peak RCS pressure are similar.

Radiological consequences for this event are presented in Section 6.4.4.1.

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

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0.00	0.00	Failure of CEDM causes CEA to eject	---	---
0.05	0.05	CEA fully ejected	---	---
0.07	0.07	Maximum core power (% of rated)	149.5	149.5
19.70	19.66	Pressurizer pressure trip setpoint (psia)	2450	2450
20.45	20.41	Reactor trip breakers open	---	---
20.45	20.41	Turbine trip occurs	---	---
21.05	21.01	Scram CEAs begin falling	---	---
21.74	21.62	PSVs open (psia)	2550	2550
22.14	22.10	Maximum RCS pressure (psia)	2678	2698
23.42	23.45	PSVs closed (psia)	2422	2422
25.34	24.82	MSSV bank 1 open	1303	1303
27.85	27.89	Maximum SG pressure (psia)	1348	1350
32.74	30.58	SG level at AFAS setpoint (%WR)	20	20
<1800	<1800	Long-term automatic plant system actions and NSSS response to this transient are similar to the CEAW at power event	---	---
1800	1800	Operator initiates cooldown (min)	30	30

Section 6.3.4.8.6 Conclusions

For the CEA ejection event, all acceptance criteria are met. The rupture of a CEDM nozzle or housing and the subsequent ejection of a CEA will not result in a radially averaged fuel enthalpy greater than 280 cal/gm at any location in any fuel rod.

The peak primary and secondary pressures remain below 120% of design at all times, thus ensuring the integrity of the RCS or the main steam system.

The radiological consequences remain below 10 CFR Part 100 guidelines.

Figure 6.3-184
CEA Ejection - Core Power vs. Time

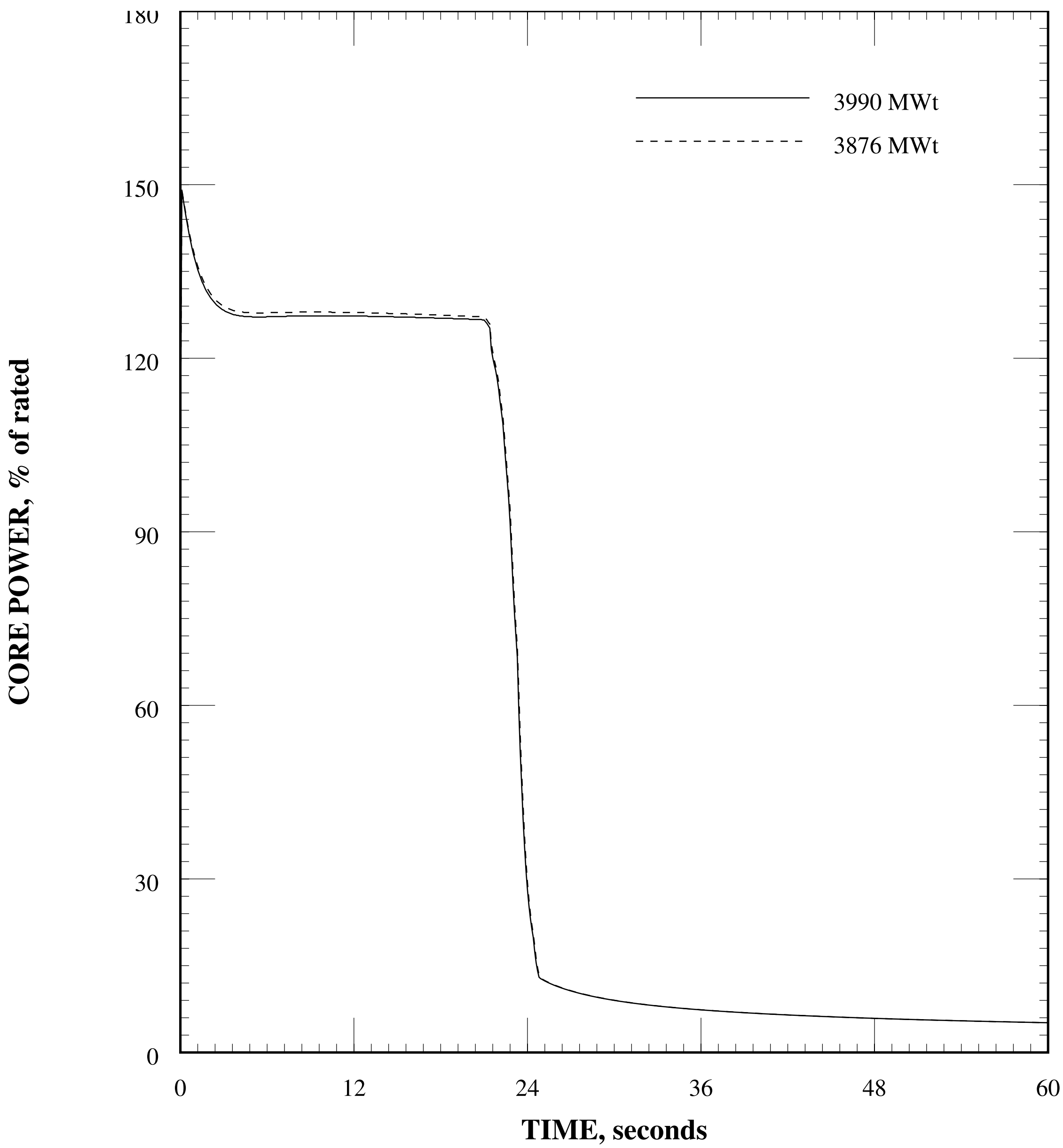


Figure 6.3-185
CEA Ejection - Core Average Heat Flux vs. Time

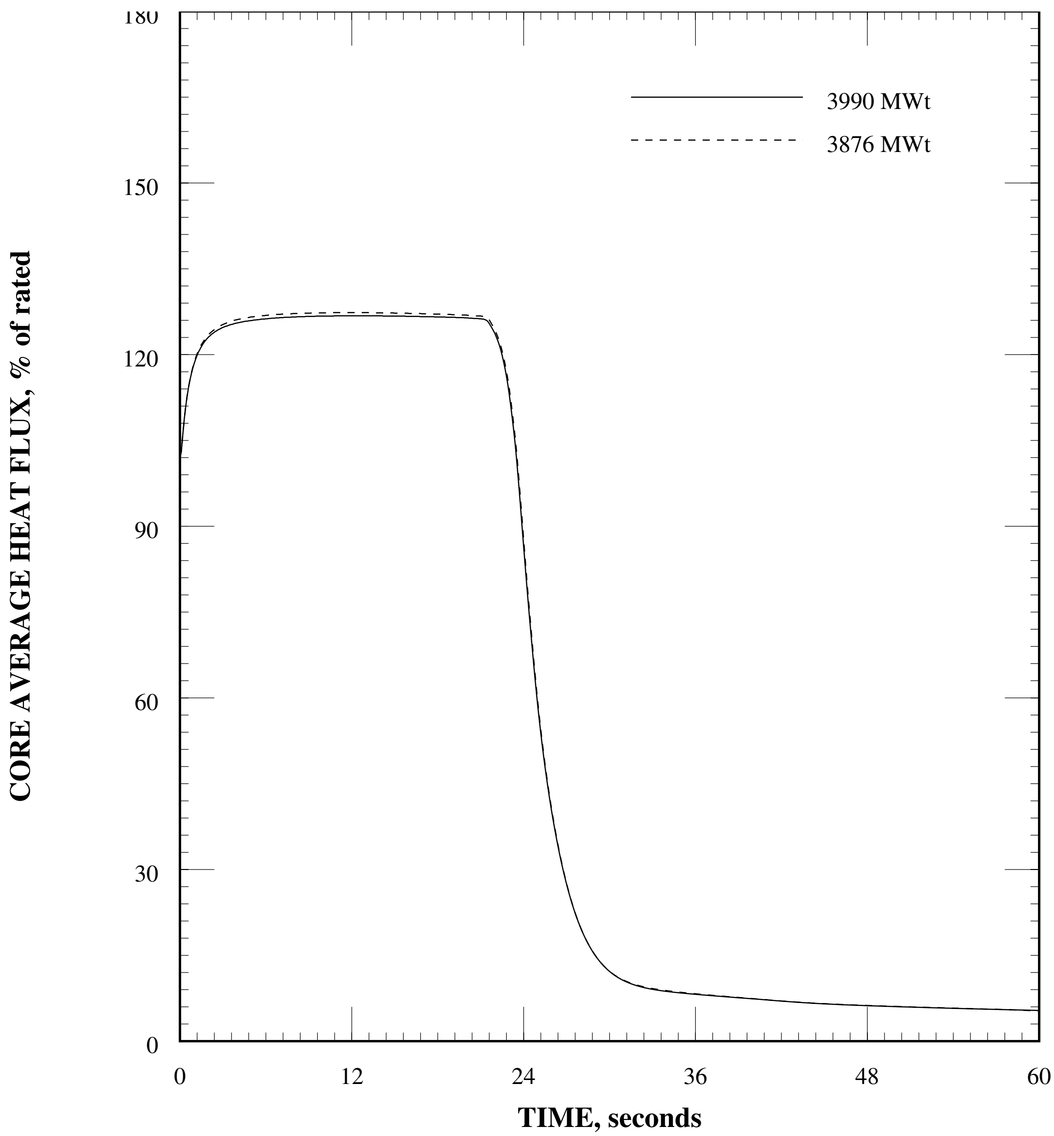


Figure 6.3-186
CEA Ejection - Core Reactivities vs. Time

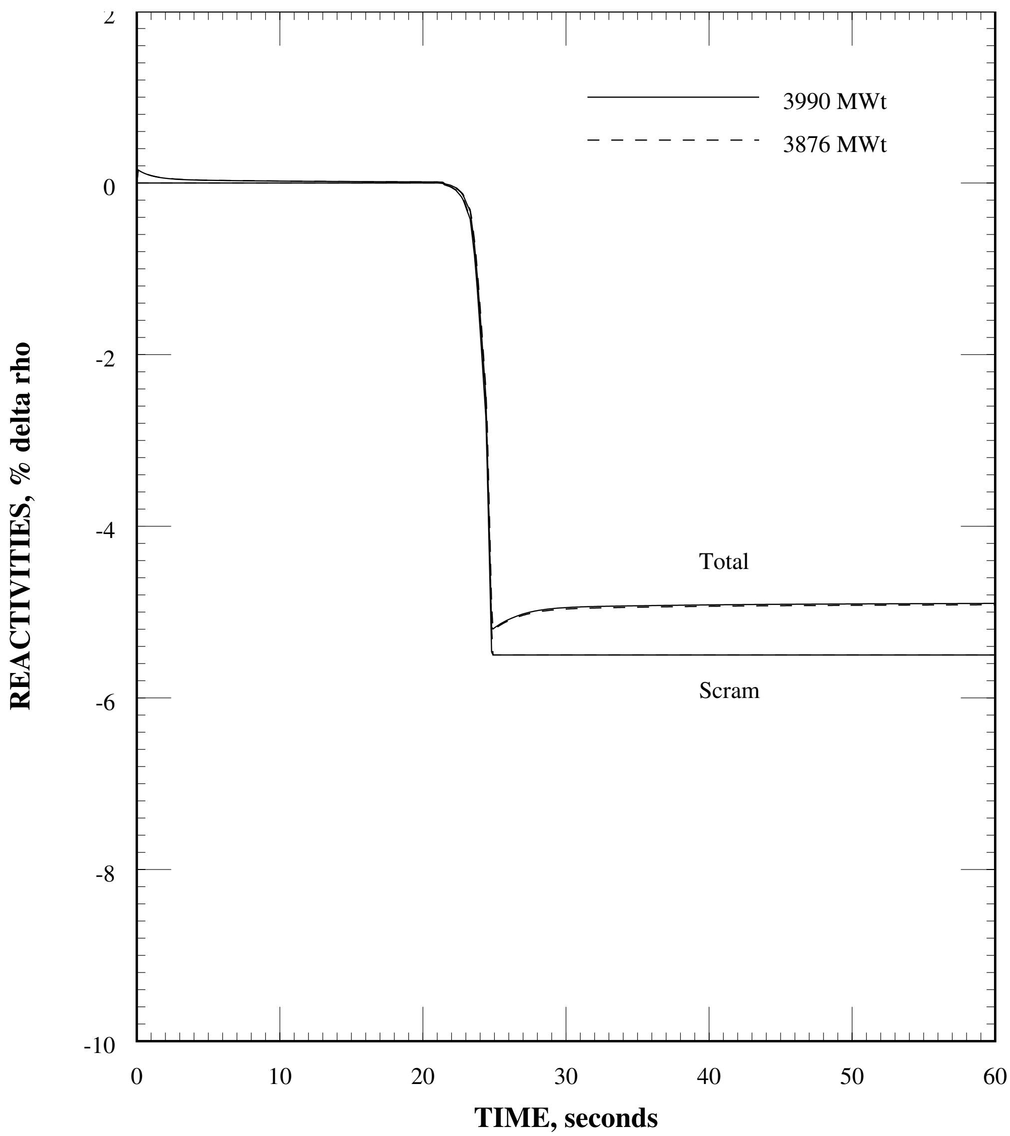


Figure 6.3-187
CEA Ejection - RCS Temperatures vs. Time

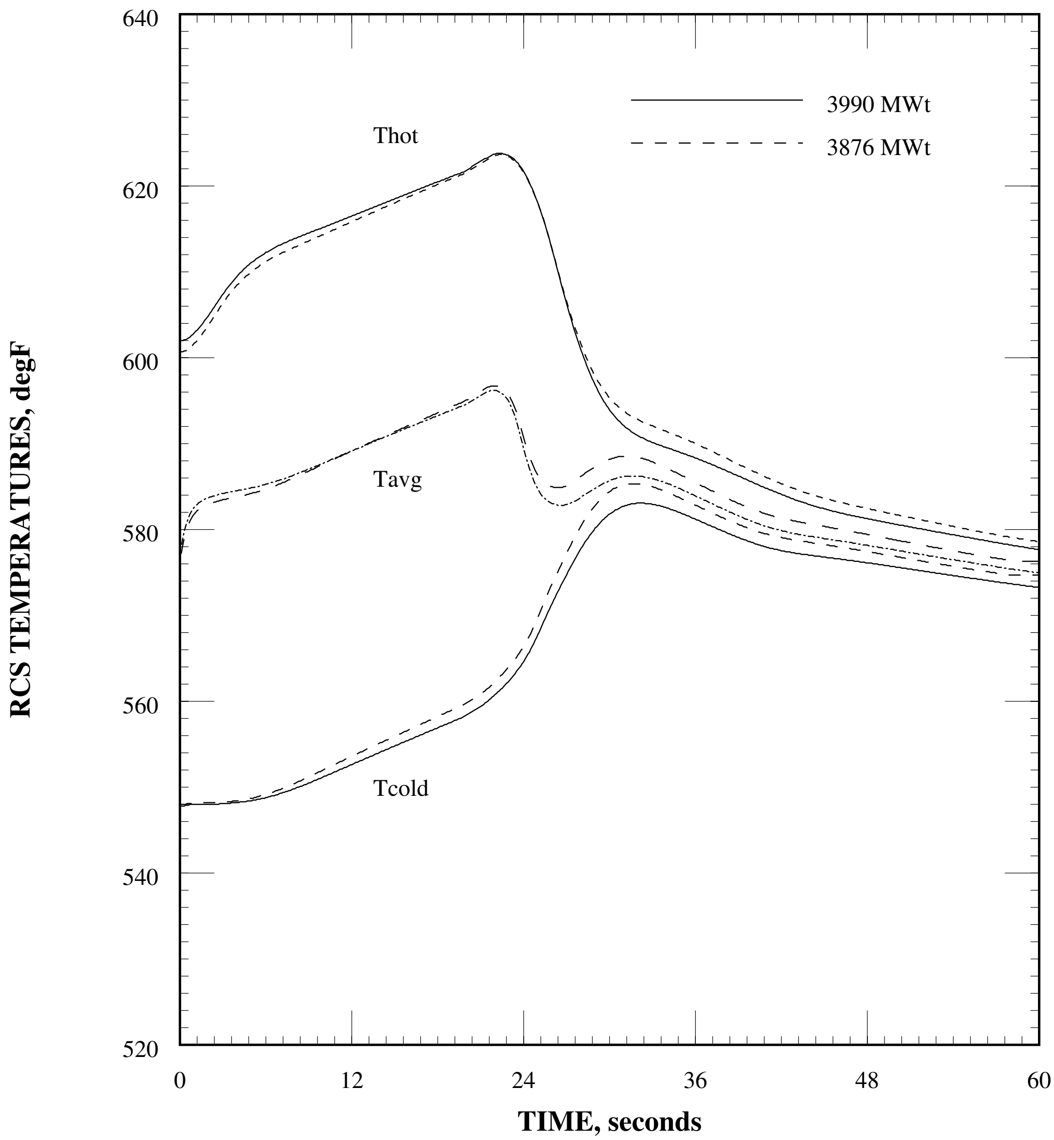


Figure 6.3-188
CEA Ejection - RCS Pressure vs. Time

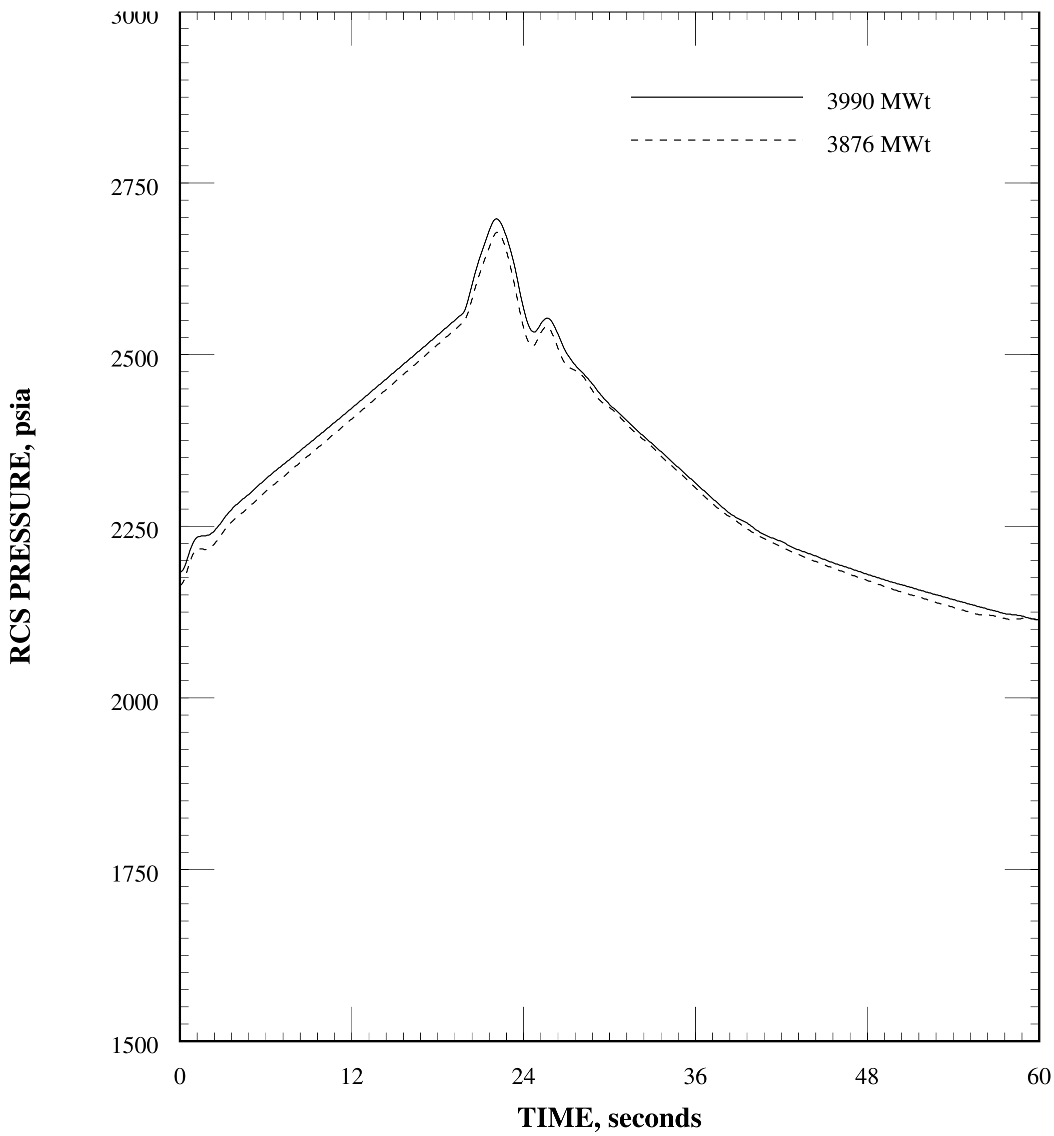


Figure 6.3-189
CEA Ejection - Pressurizer Pressure vs. Time

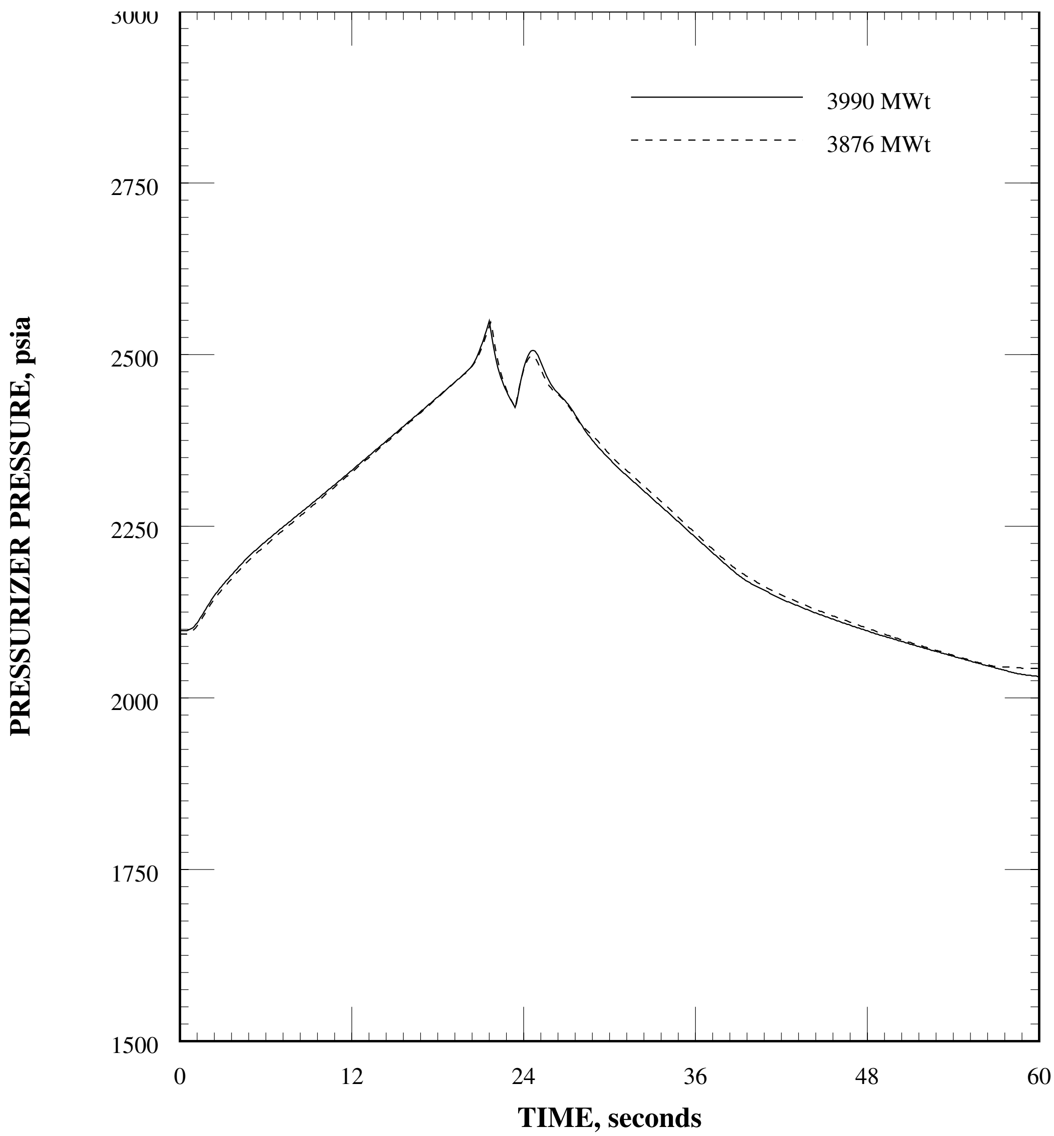


Figure 6.3-190
CEA Ejection - Pressurizer Liquid Volume vs. Time

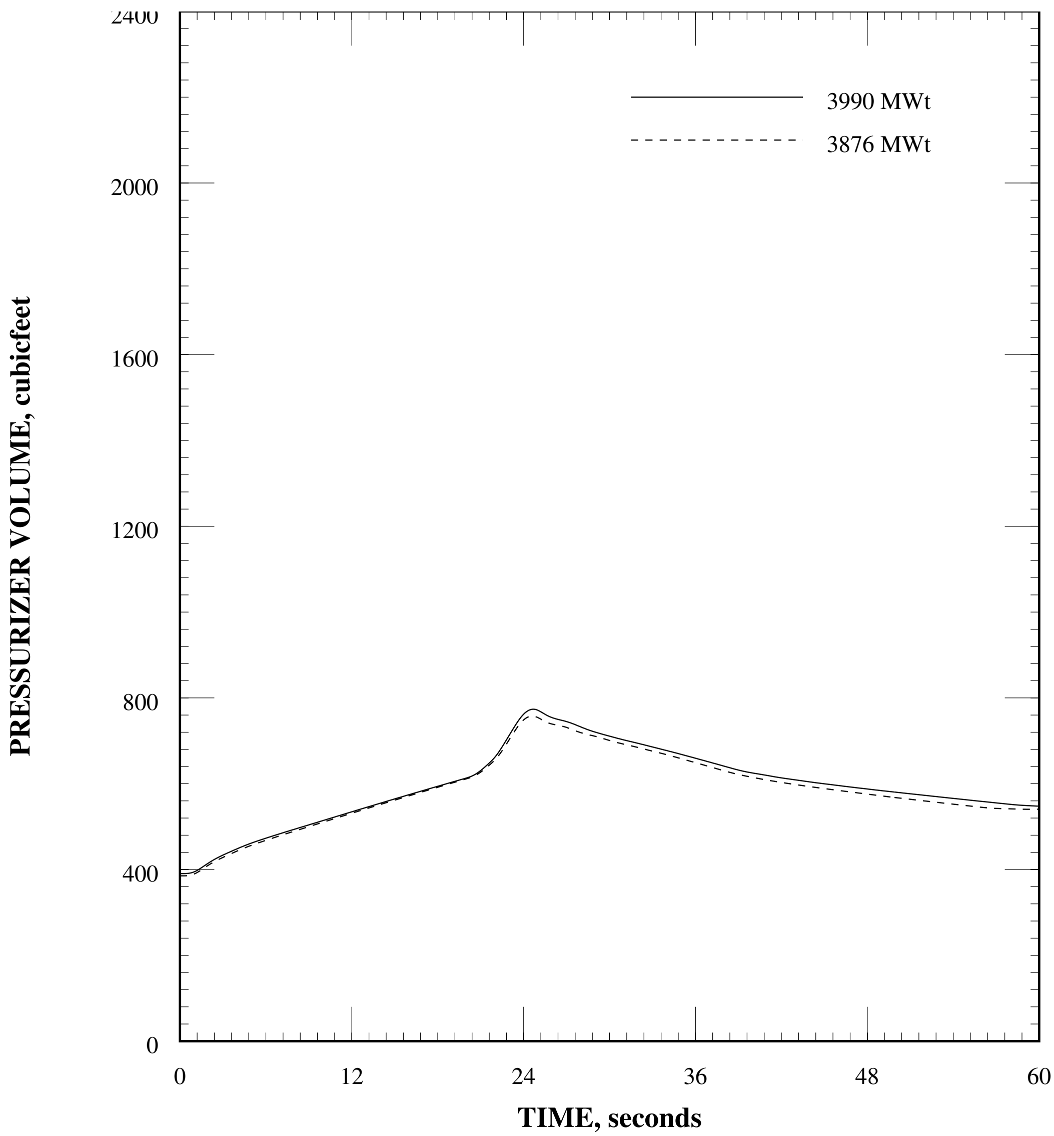


Figure 6.3-191
CEA Ejection - SG Pressure vs. Time

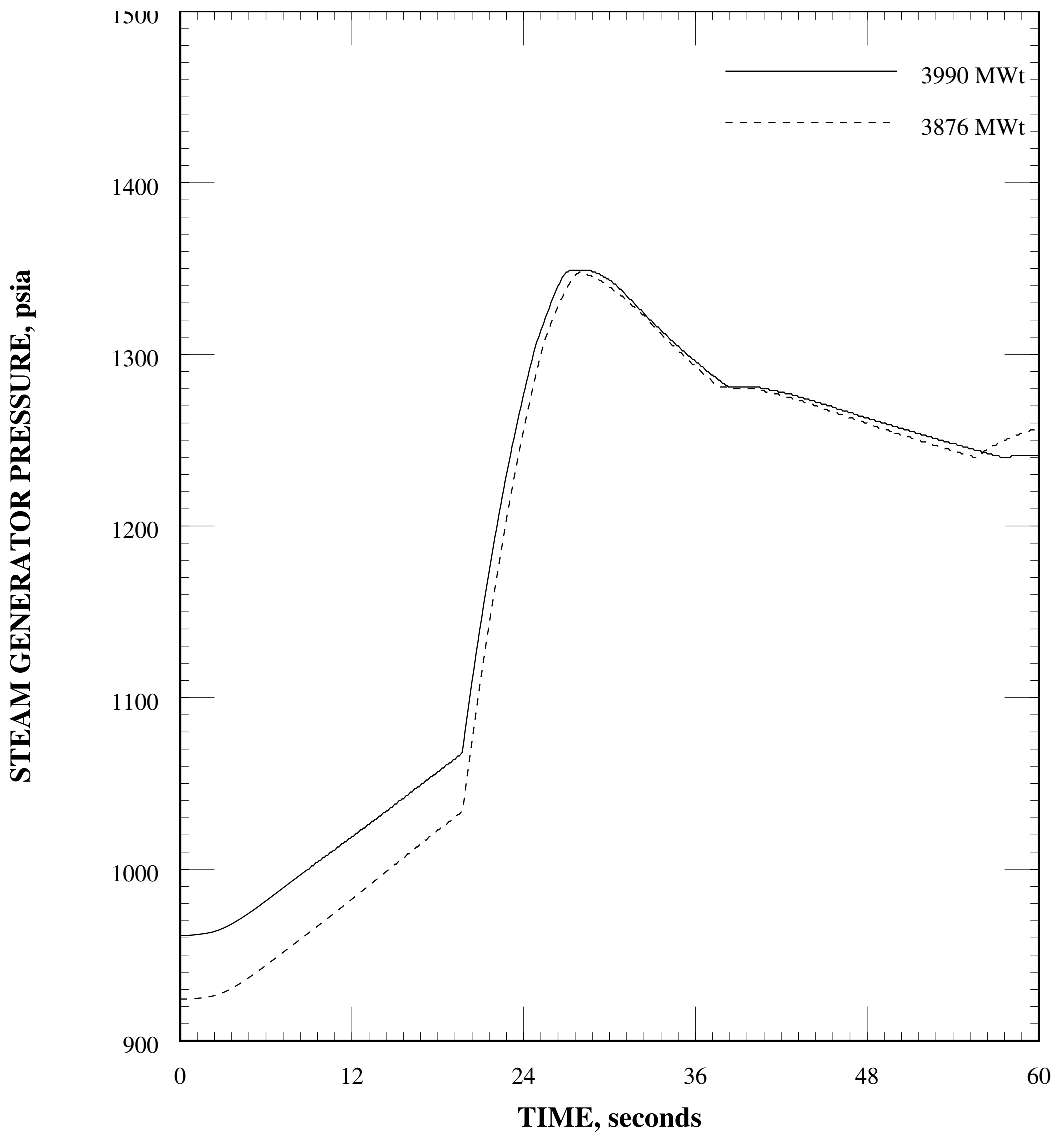


Figure 6.3-192
CEA Ejection - SG Level vs. Time

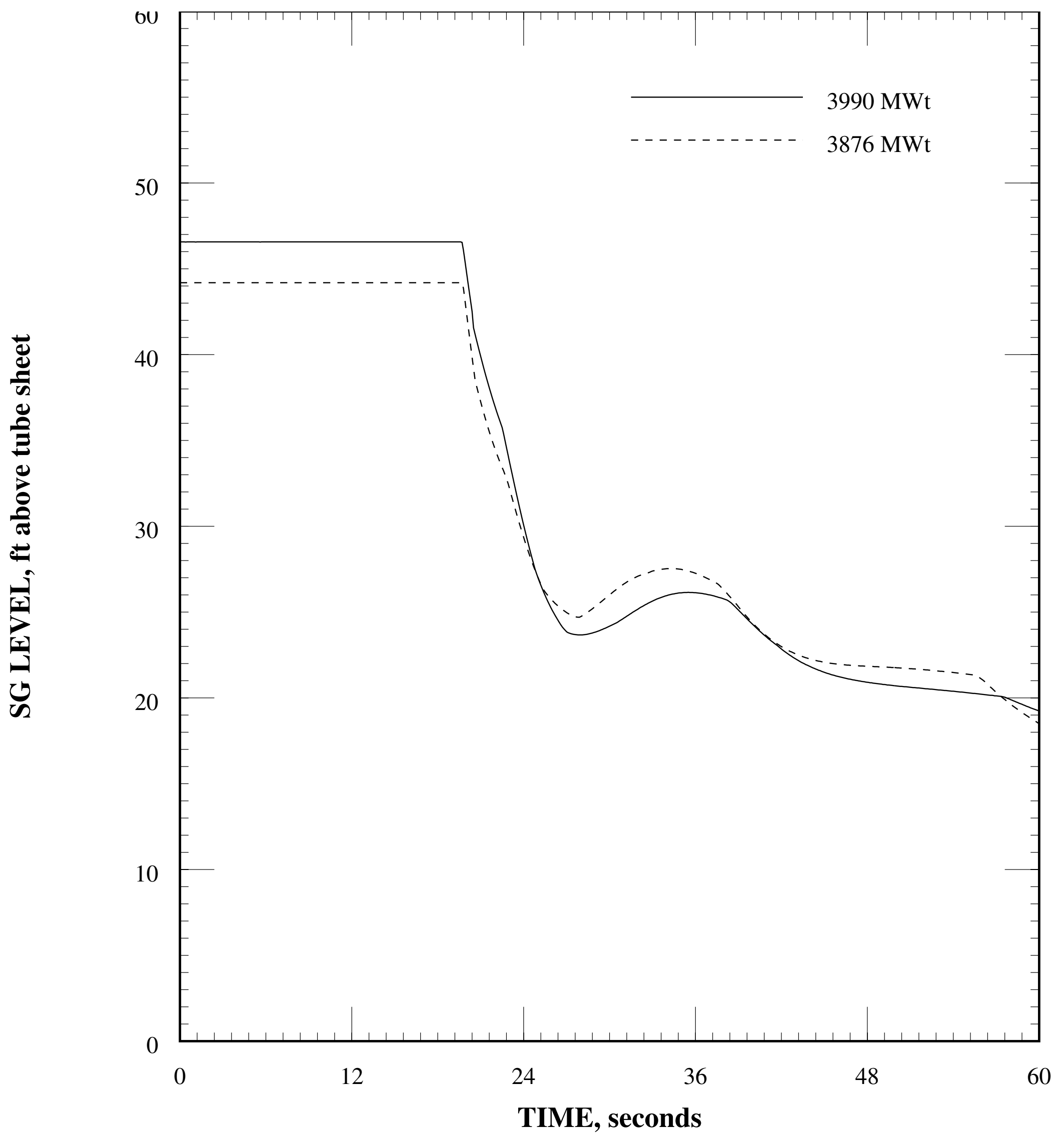


Figure 6.3-193
CEA Ejection - SG Liquid Mass Inventory vs. Time

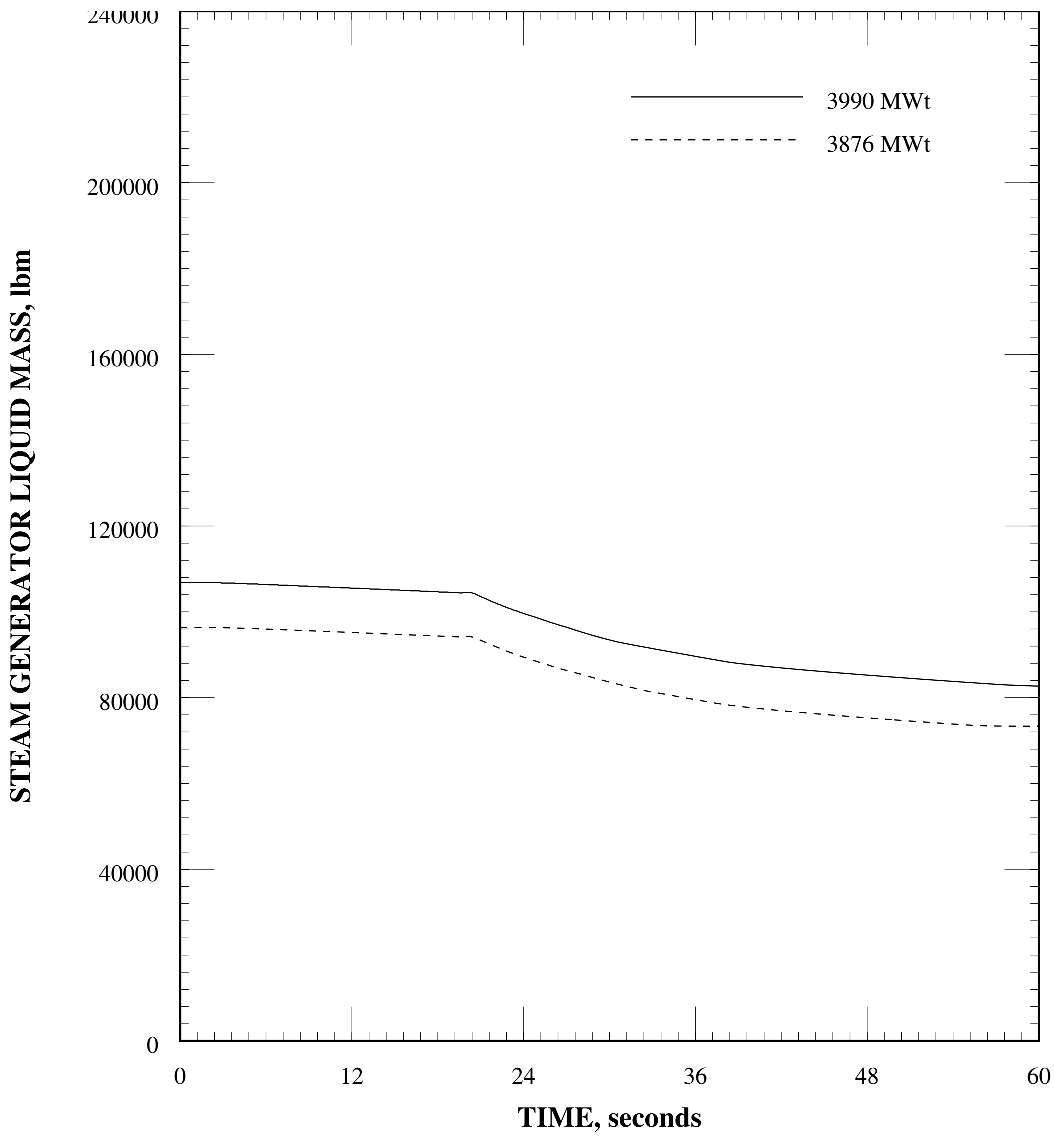


Figure 6.3-194
CEA Ejection - Integrated Steam Flow vs. Time

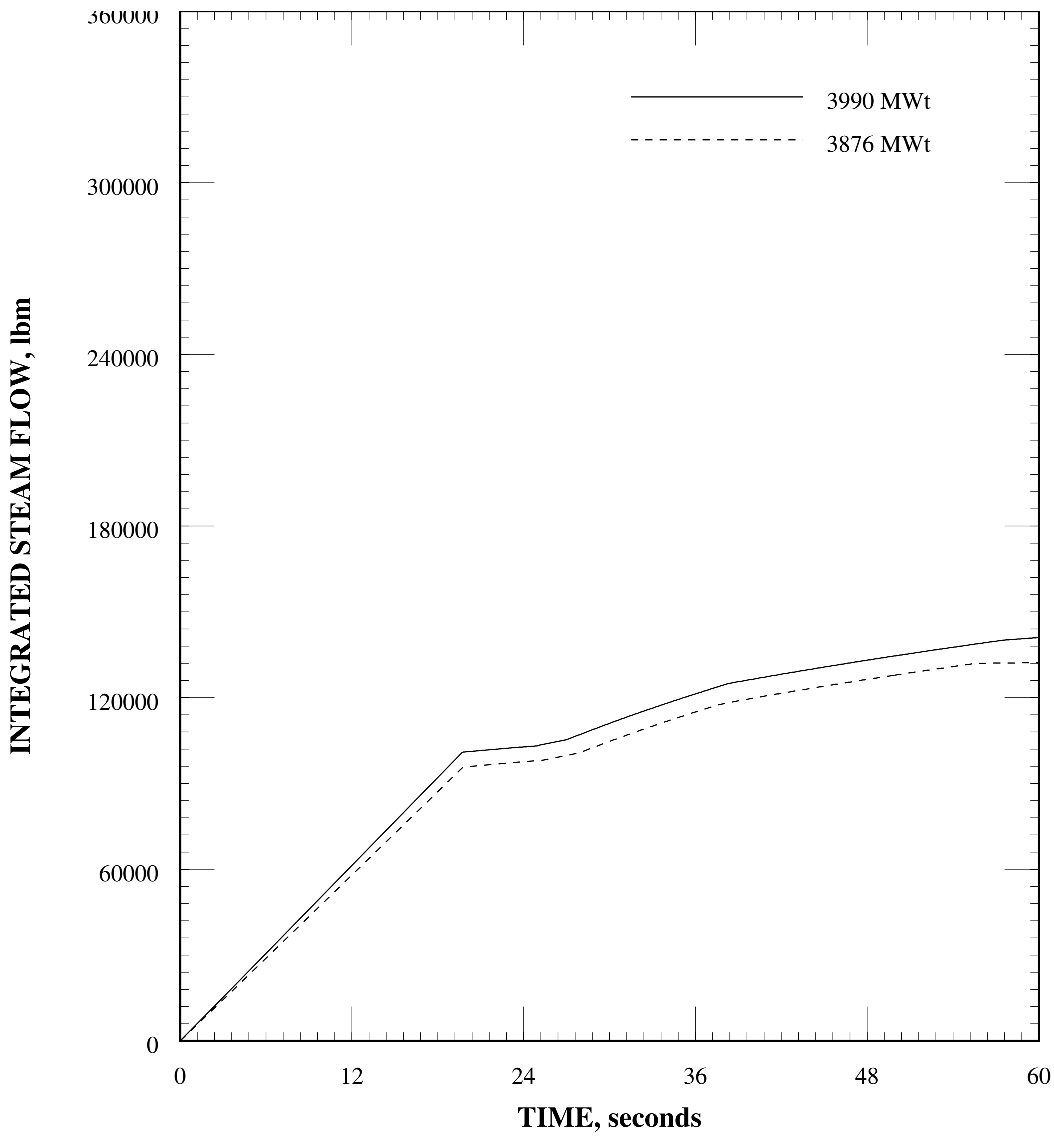
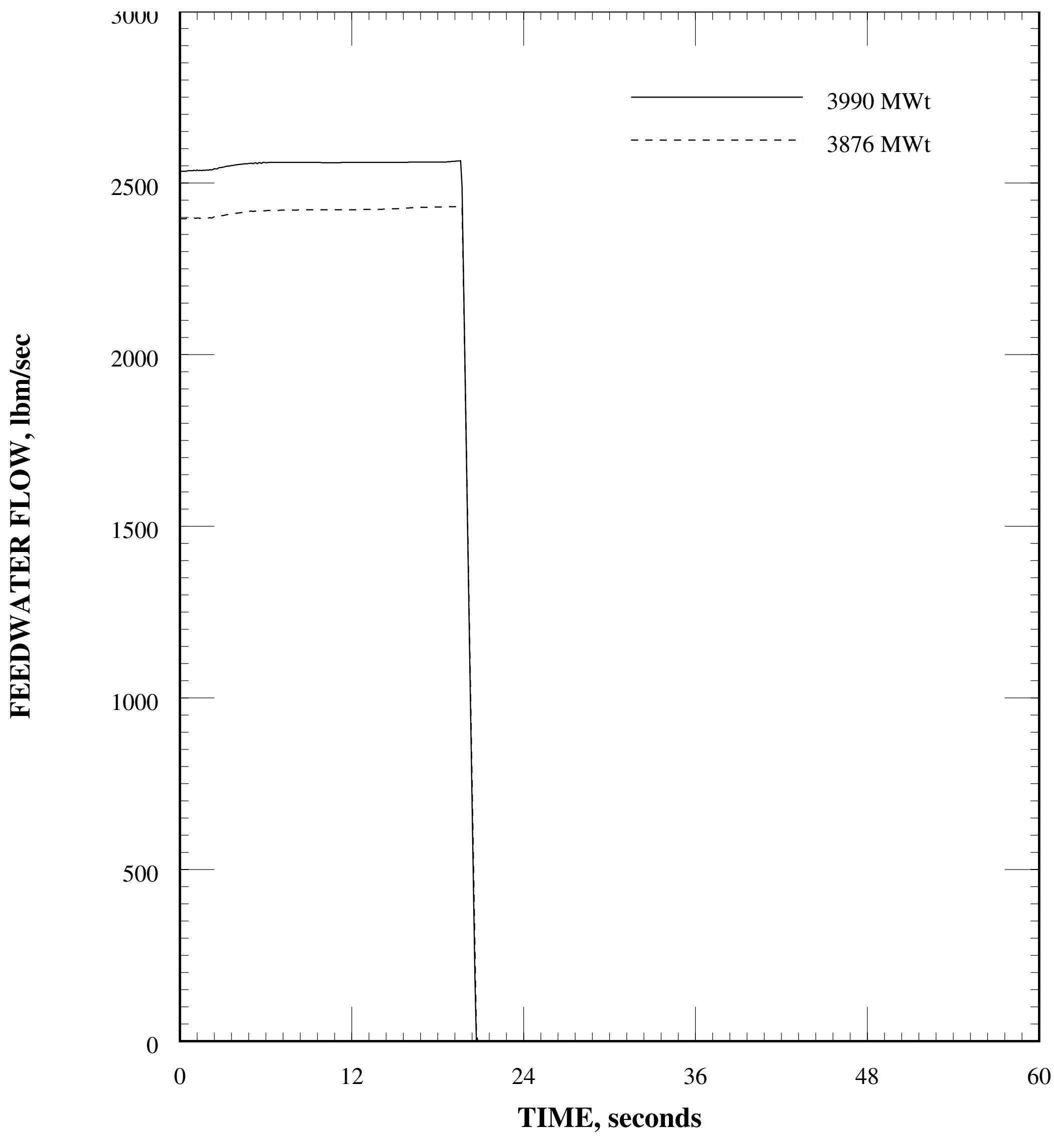


Figure 6.3-195
CEA Ejection - Total FW Flow vs. Time



Section 6.3.5 Increase in Reactor Coolant System Inventory

Section 6.3.5.1 Inadvertent Operation of the Emergency Core Cooling System

As described in UFSAR Section 15.5.1, the inadvertent operation of the ECCS is assumed to actuate the two HPSI pumps and open the corresponding discharge valves. This operation occurs because of a spurious signal to the system or operator error.

The inadvertent operation of the ECCS event is classified as an AOO. This event was reviewed for the impacts of PUR. All PUR conditions are within the assumptions in UFSAR Section 15.5.1. This event remains bounded by the AOR.

Section 6.3.5.2 Chemical and Volume Control System Malfunction - Pressurizer Level Control System Malfunction with a Concurrent Loss of Offsite Power

As described in UFSAR Section 15.5.2, the limiting event in this category is a PLCS malfunction, in combination with the LOP because of a grid failure after a turbine trip.

When in the automatic mode, the PLCS responds to changes in pressurizer level by changing charging and letdown flows to maintain the programmed level. Normally, two charging pumps are running with one charging pump available for automatic startup. If the pressurizer level controller fails low or the level setpoint generated by the RRS fails high, a low level signal can be transmitted to the controller. In response, the controller will start all the charging pumps and close the letdown control valve to its minimum opening, resulting in the maximum mass addition to the RCS.

The CVCS malfunction - PLCS malfunction with a concurrent LOP event is classified as an infrequent event. This event was reviewed for the impacts of PUR. All PUR conditions are within the assumptions in UFSAR Section 15.5.2. This event remains bounded by the AOR.

Section 6.3.6 Decrease in Reactor Coolant System Inventory

Section 6.3.6.1 Inadvertent Opening of a Pressurizer Safety/Relief Valve

As described in UFSAR Section 15.6.1, the inadvertent opening of a PSV event as described in SRP is evaluated in the ECCS analyses (Section 6.1).

Section 6.3.6.2 Double-Ended Break of a Letdown Line Outside Containment

Section 6.3.6.2.1 Identification of Event and Causes

As described in UFSAR Section 15.6.2, direct release of RCS coolant may result from a break or leak outside containment in a letdown line, instrument line, or sample line. The Double-Ended Break of a Letdown Line Outside Containment of the letdown line control valve (DBLLOCUS) was selected for this analysis because it is the largest line, and results in the largest release of reactor coolant outside the containment.

The single active failure of an isolation valve was not considered in the analysis because the letdown line includes two isolation valves in series situated inside the containment. Hence, failure of one isolation valve does not make the consequences of the event more severe.

A DBLLOCUS can range from a small crack in the piping to a complete double-ended break. The cause of the event may be attributed to corrosion, or to fatigue cracks resulting from vibration or inadequate welds.

Section 6.3.6.2.2 Acceptance Criteria

As defined in the SRP Section 15.6.2, the specific acceptance criteria for this event are:

- The plant is considered adequately designed against the radiological consequences of a failure of containment-penetrating small line carrying RCS (and the Technical Specifications for primary coolant activity and isolation time and maximum allowable leak rate of isolation valves in these lines is appropriately limited) if calculations show that the resulting doses at the Exclusion Area Boundary (EAB) are small fractions (no more than 10%) of the 10 CFR Part 100 exposure guidelines.

Section 6.3.6.2.3 Description of Analysis

The NSSS response to the DBLLOCUS event was simulated using the CENTS code. The initial and transient DNBR is calculated using the CETOP-D code which uses the CE-1 CHF correlation.

Both the existing configuration and PUR were analyzed in order to compare the NSSS response to the DBLLOCUS event. Input parameters and initial conditions were selected to maximize leak flow that subsequently maximizes RCS depressurization and DNBR degradation.

Section 6.3.6.2.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-47 contains the initial conditions used for the DBLLOCUS event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. Due to the number of available alarms that will actuate during a postulated DBLLOCUS, operator action to isolate the letdown line is credited at 10 minutes.

Table 6.3-47
Parameters Used for DBLLOCUS Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temp (°F)	548	548
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	116	116
Initial pressurizer level (ft)	23.7	23.7
Initial SG level (ft)	nominal	nominal
MTC ($\Delta\rho$/°F)	-4.2E-04	-4.2E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	no trip	no trip
Fuel rod gap conductance (Btu/hr-ft^2-°F)	5755	5755
Plugged SG tubes (% of tubes/SG)	0	0
Single failure	none	none
LOP	no	no

Section 6.3.6.2.5 Results

Table 6.3-48 presents a sequence of events that occur following the DBLLOCUS event until operator action is initiated at 10 minutes. Figure 6.3-196 through Figure 6.3-207 presents the behavior of NSSS parameters following the DBLLOCUS event.

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 limiting initial conditions and break sizes are identical between the two plant configurations. Therefore, the integrated break flow is equivalent.

Radiological consequences for this event are presented in Section 6.4.6.1.

Table 6.3-48
Sequence of Events for DBLLOCUS Event

Time (sec)			Value	
3876 MW_t	3990 MW_t	Event	3876 MW_t	3990 MW_t
0.00	0.00	DBLLOCUS occurs	---	---
46.38	34.39	Second charging pump starts (ft)	22.3	22.7
48.88	49.78	Pressurizer backup and proportional heaters on (psia)	2284	2284
440.44	447.89	Pressurizer backup and proportional heaters off (psia)	2216	2182
600	600	RCS inventory release (lb_m)	28000	28000
600	600	Operator isolates the DBLLOCUS and takes steps for a controlled shutdown (min)	10	10

Section 6.3.6.2.6 Conclusions

For the DBLLOCUS event, all acceptance criteria are met. Offsite doses remained below the acceptance criteria for this category of event. Specifically, a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body).

Figure 6.3-196
DBLLOCUS Event - Core Power vs. Time

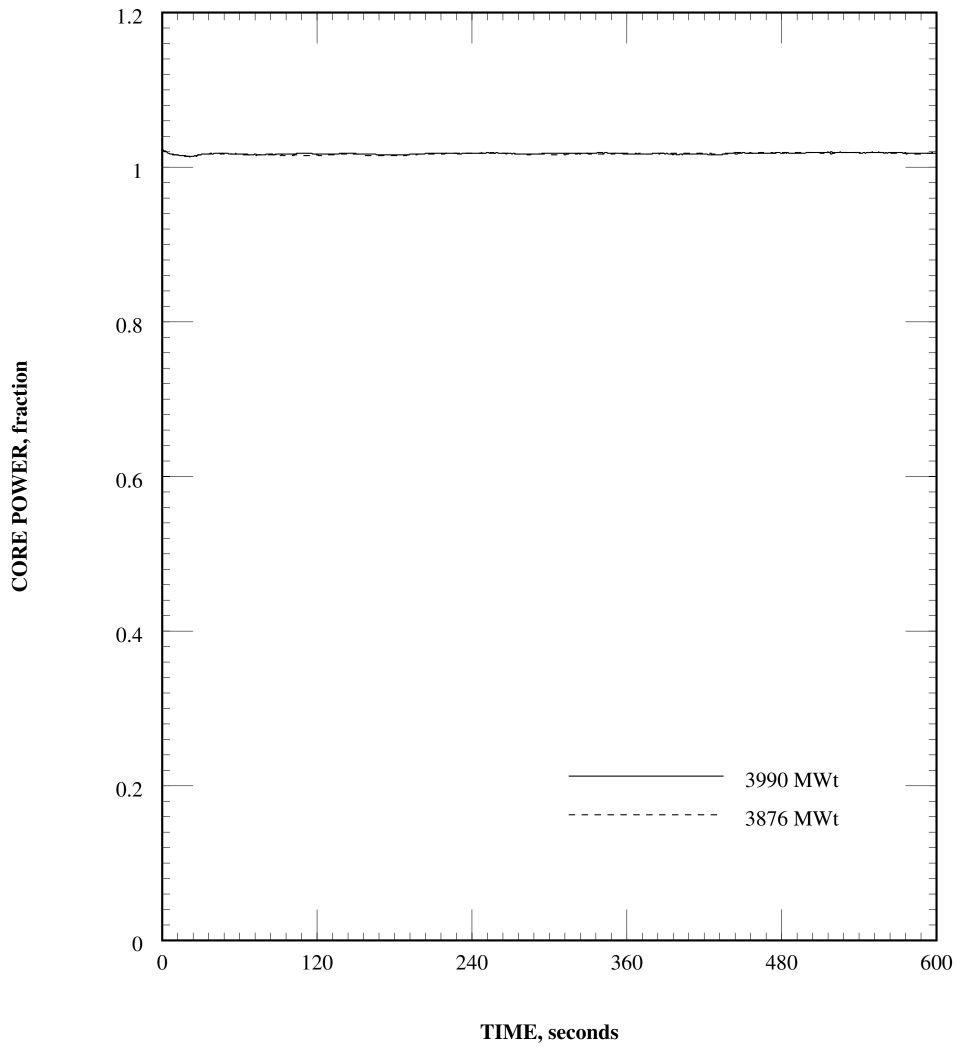


Figure 6.3-197
DBLLOCUS Event - Core Average Heat Flux vs. Time

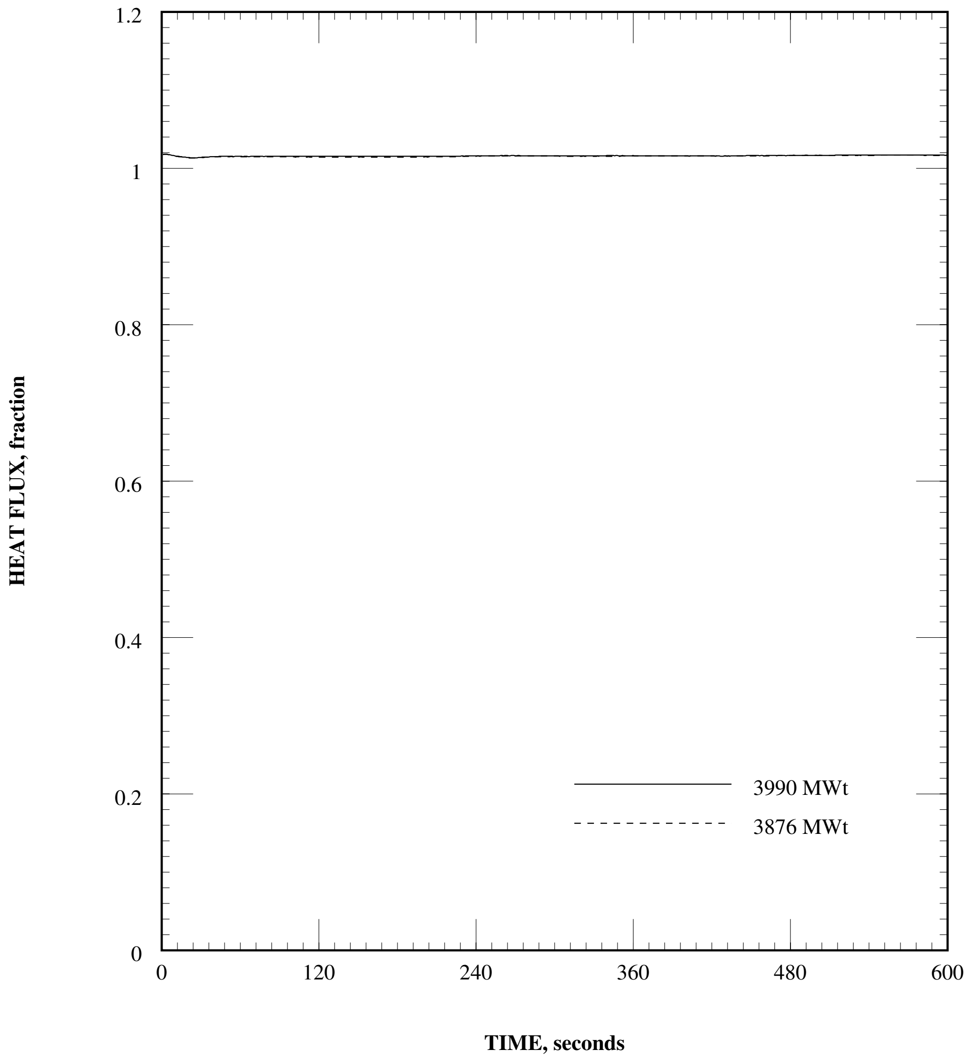


Figure 6.3-198
DBLLOCUS Event - Minimum DNBR vs. Time

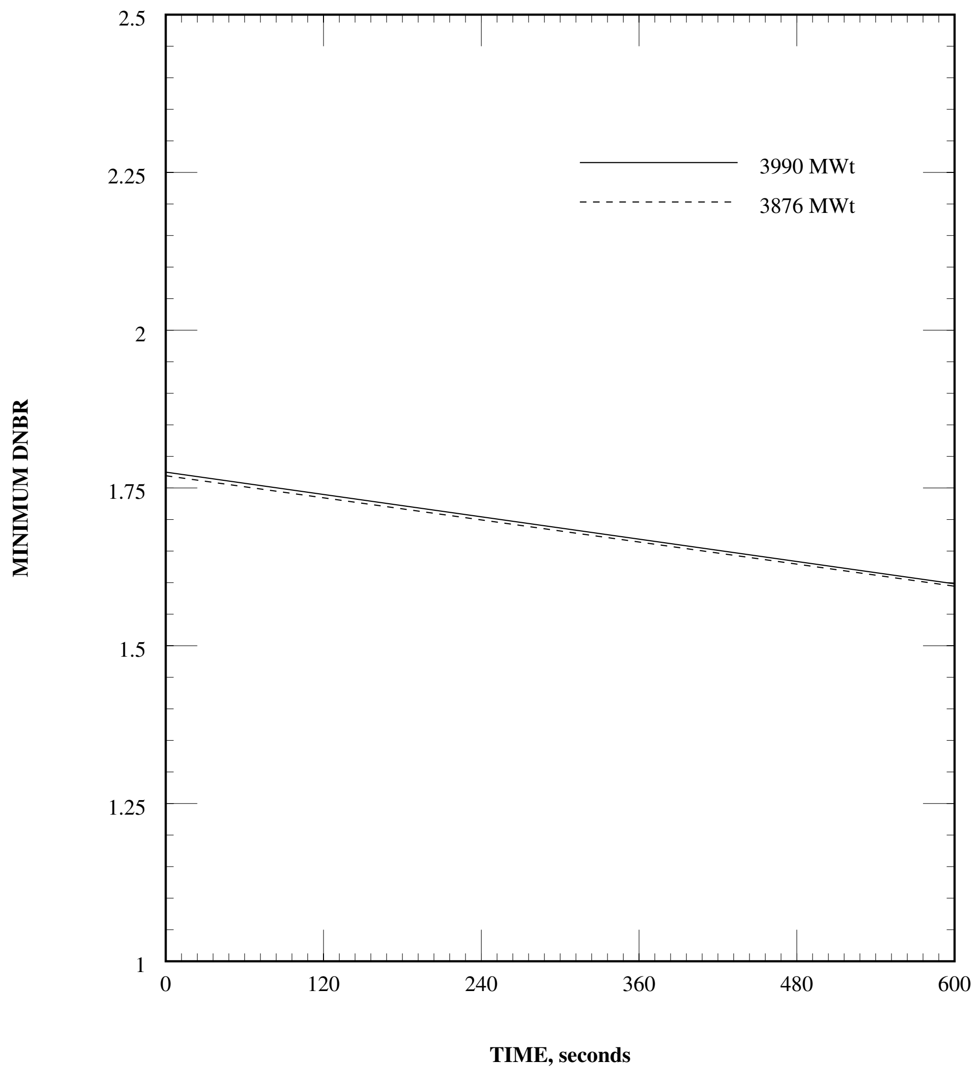


Figure 6.3-199
DBLLOCUS Event - RCS Temperatures vs. Time

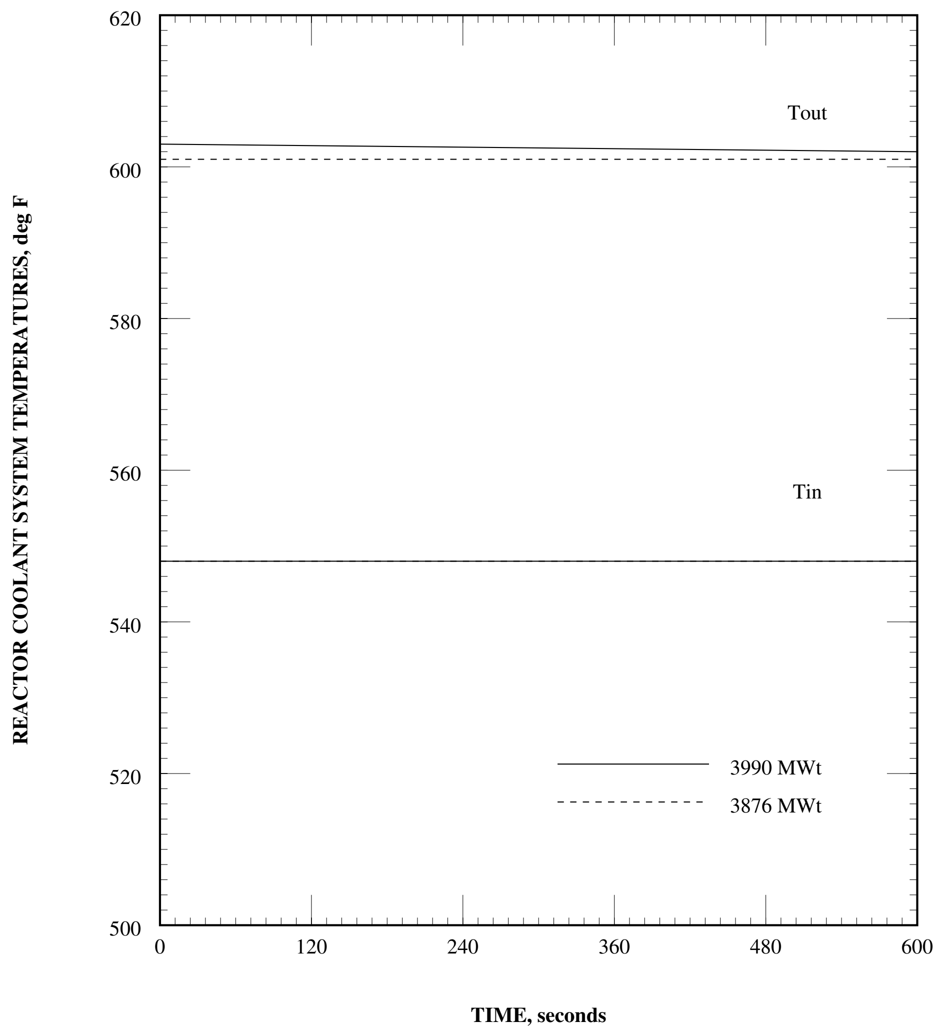


Figure 6.3-200
DBLLOCUS Event - Pressurizer Pressure vs. Time

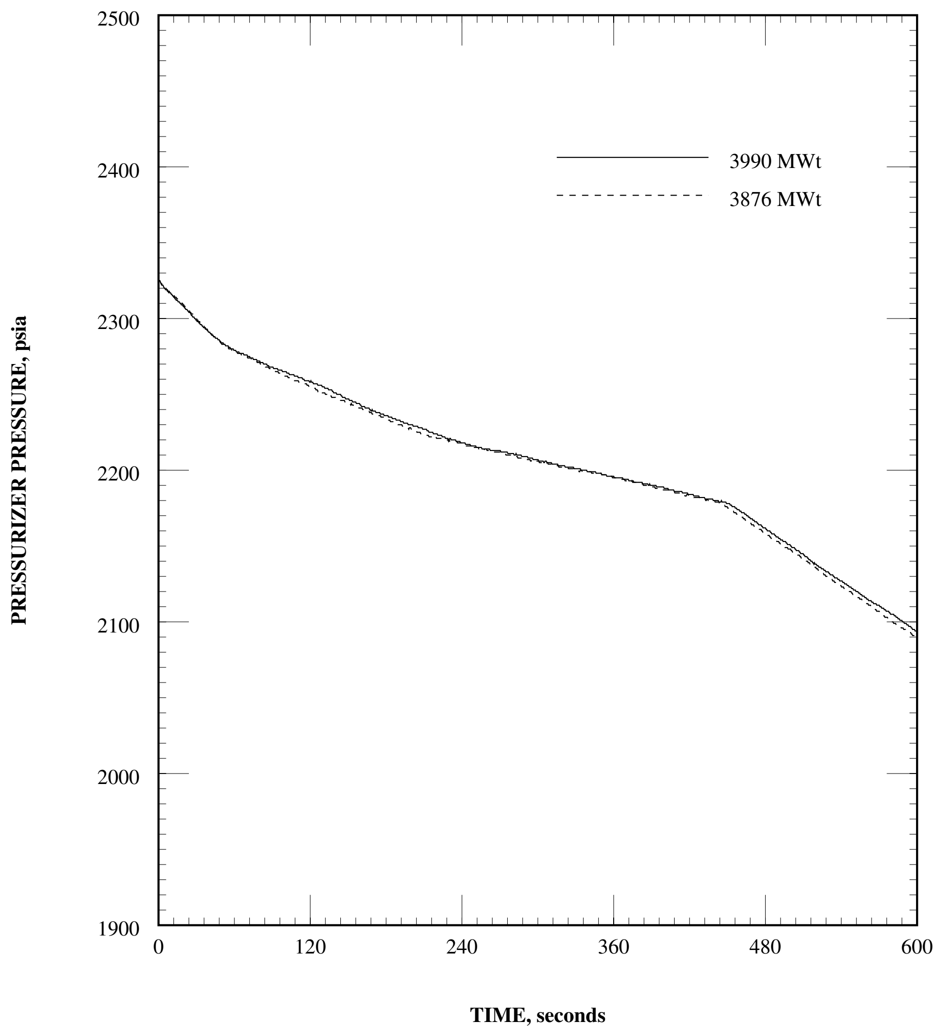


Figure 6.3-201
DBLLOCUS Event - SG Pressure vs. Time

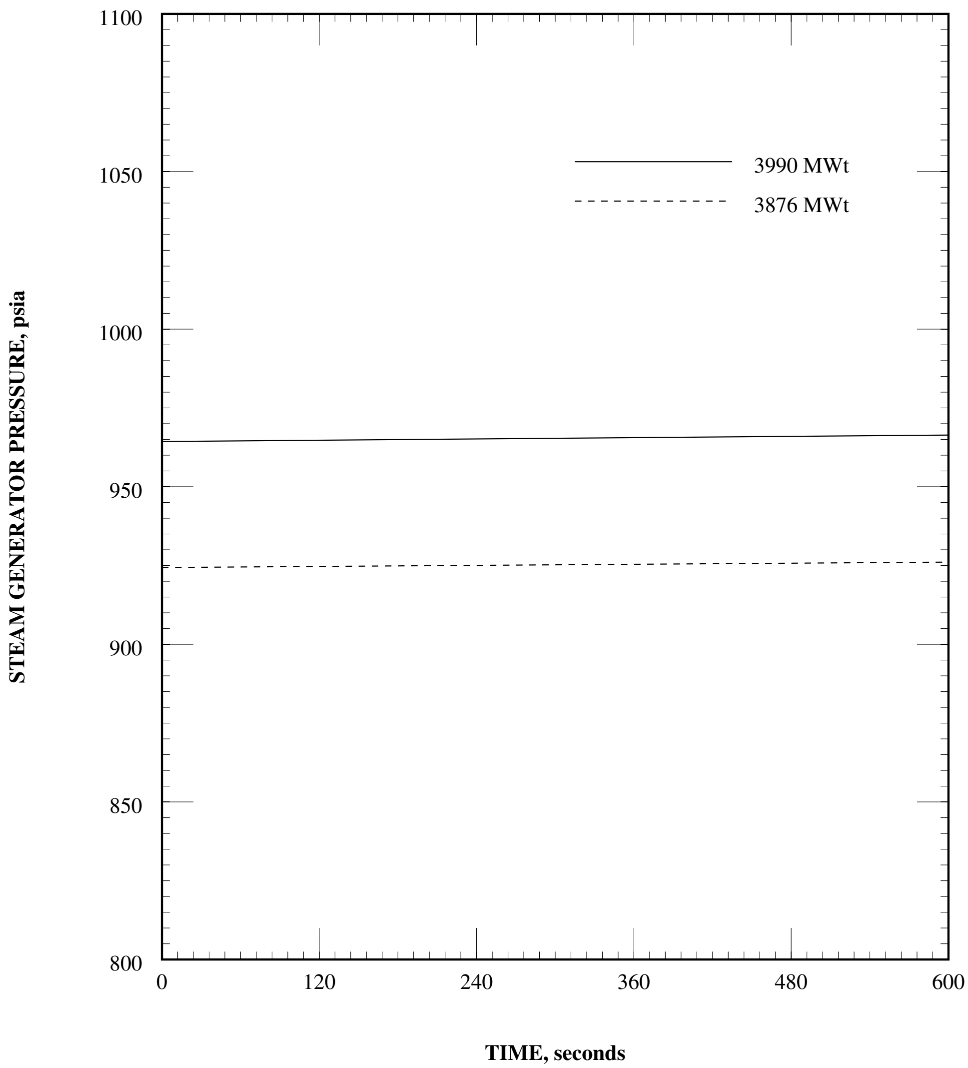


Figure 6.3-202
DBLLOCUS Event - FW Enthalpy vs. Time

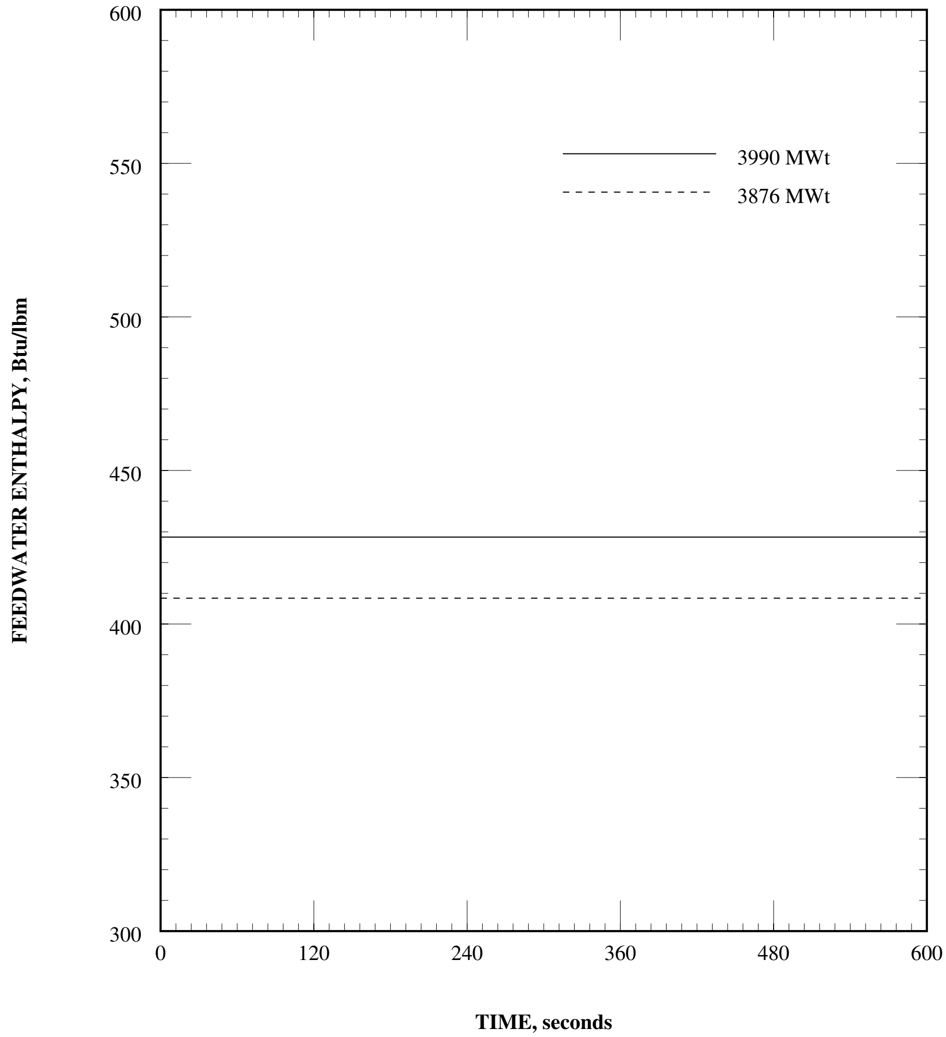


Figure 6.3-203
DBLLOCUS Event - FW Flow vs. Time

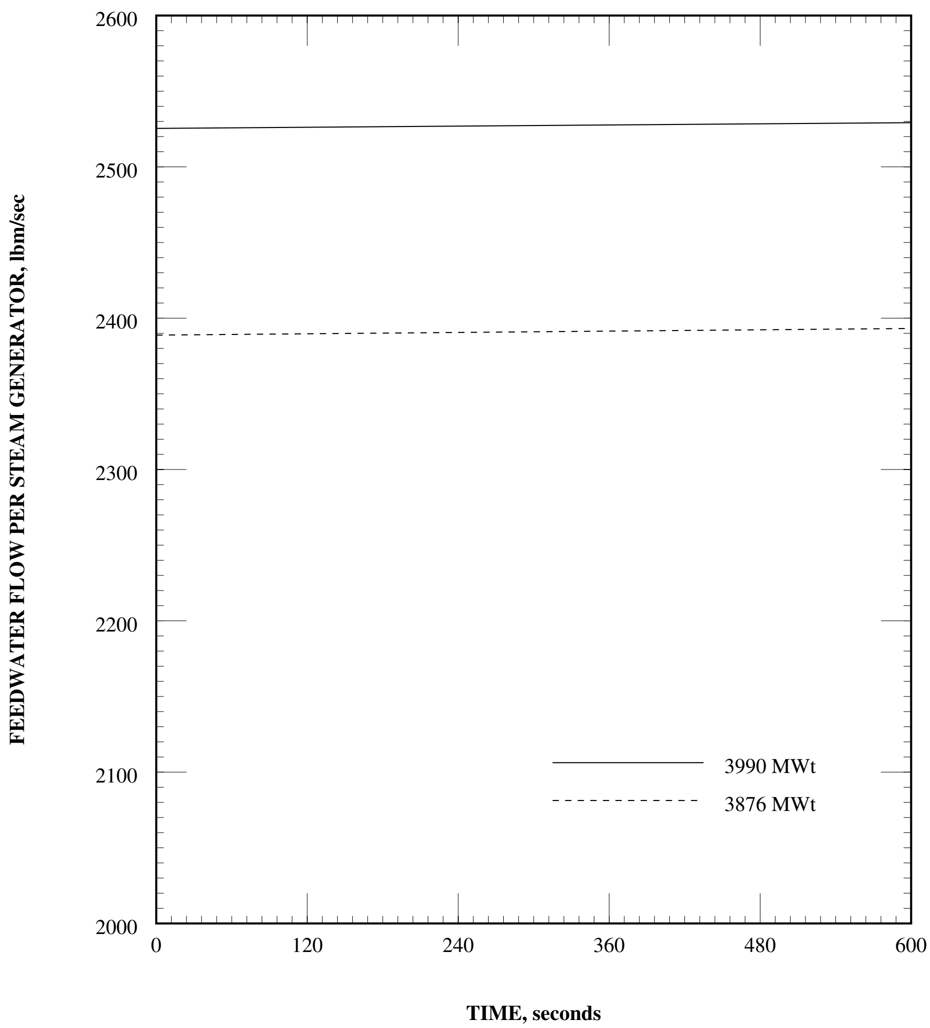


Figure 6.3-204
DBLLOCUS Event - Steam Flow per SG vs. Time

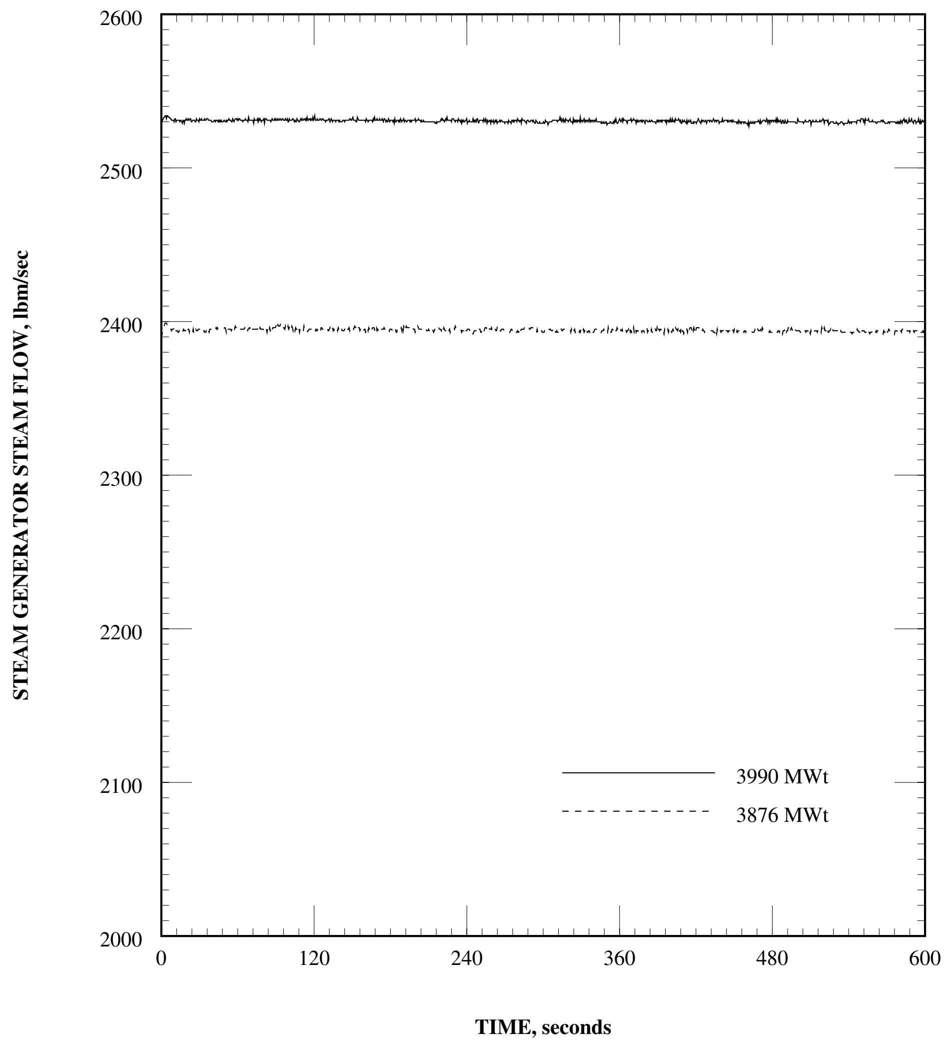


Figure 6.3-205
DBLLOCUS Event - SG Level vs. Time

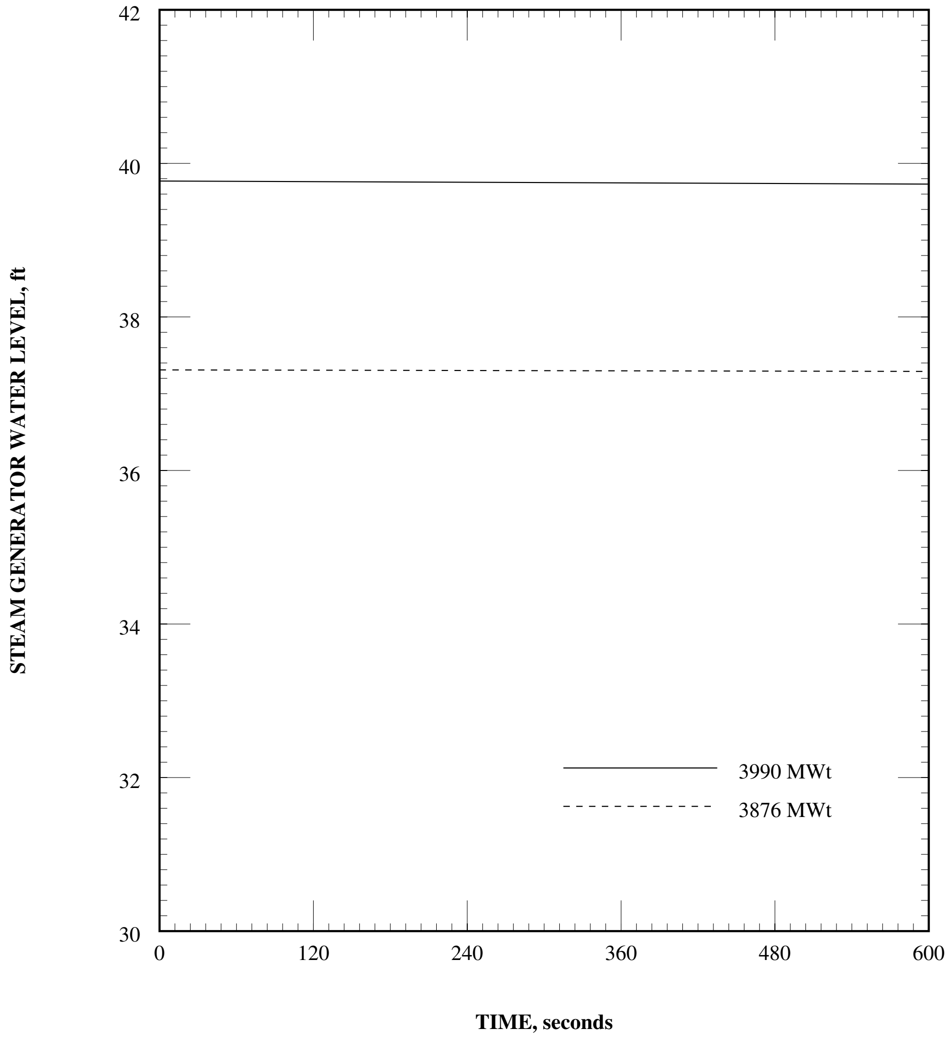


Figure 6.3-206
DBLLOCUS Event - RCS Inventory vs. Time

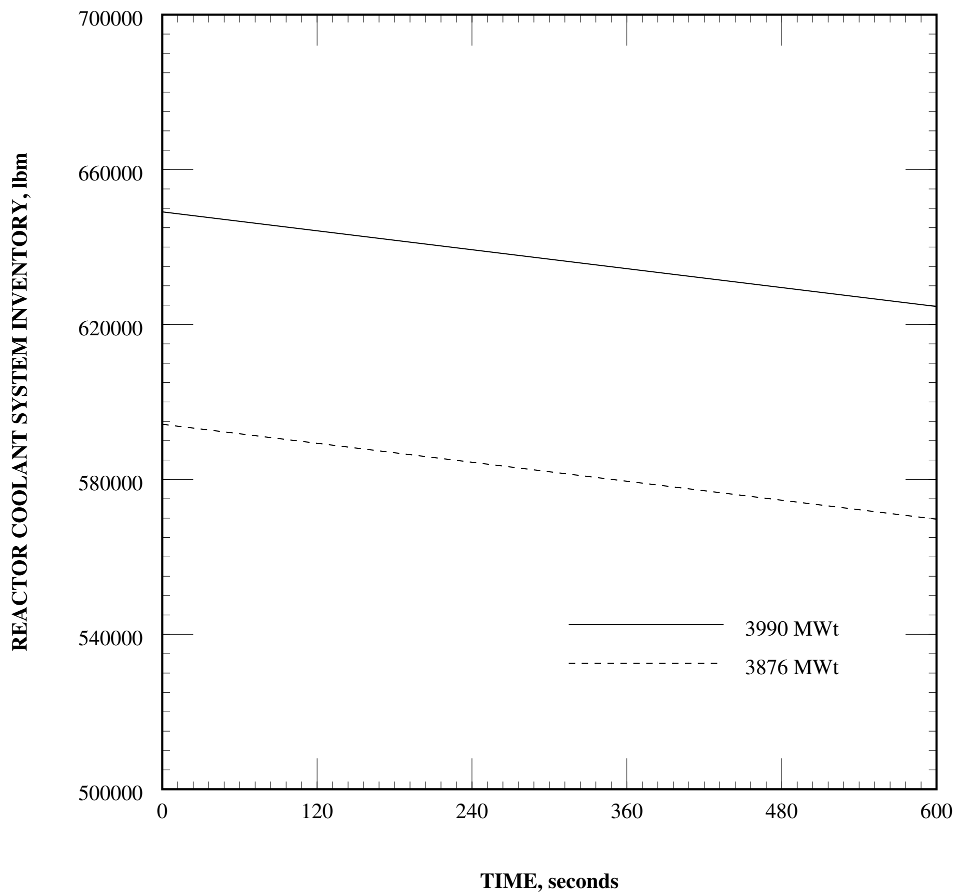


Figure 6.3-207
DBLLOCUS Event - Pressurizer Water Volume vs. Time

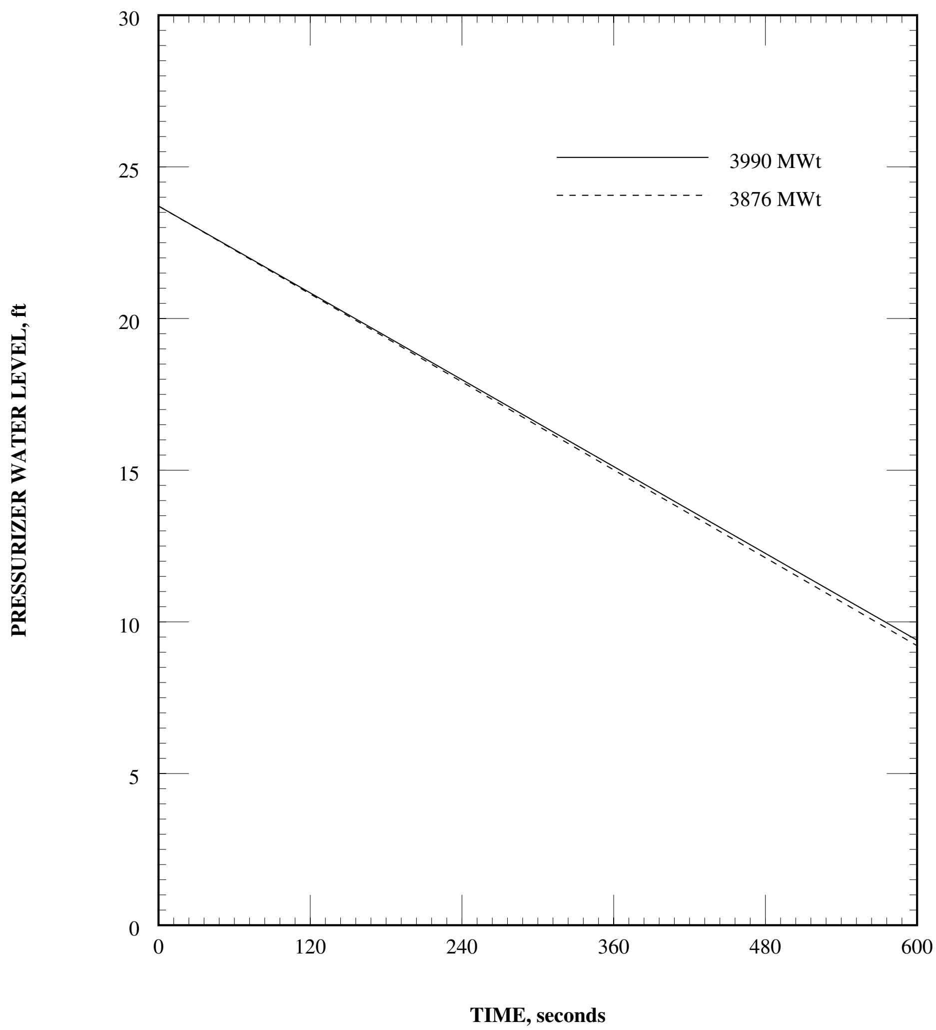
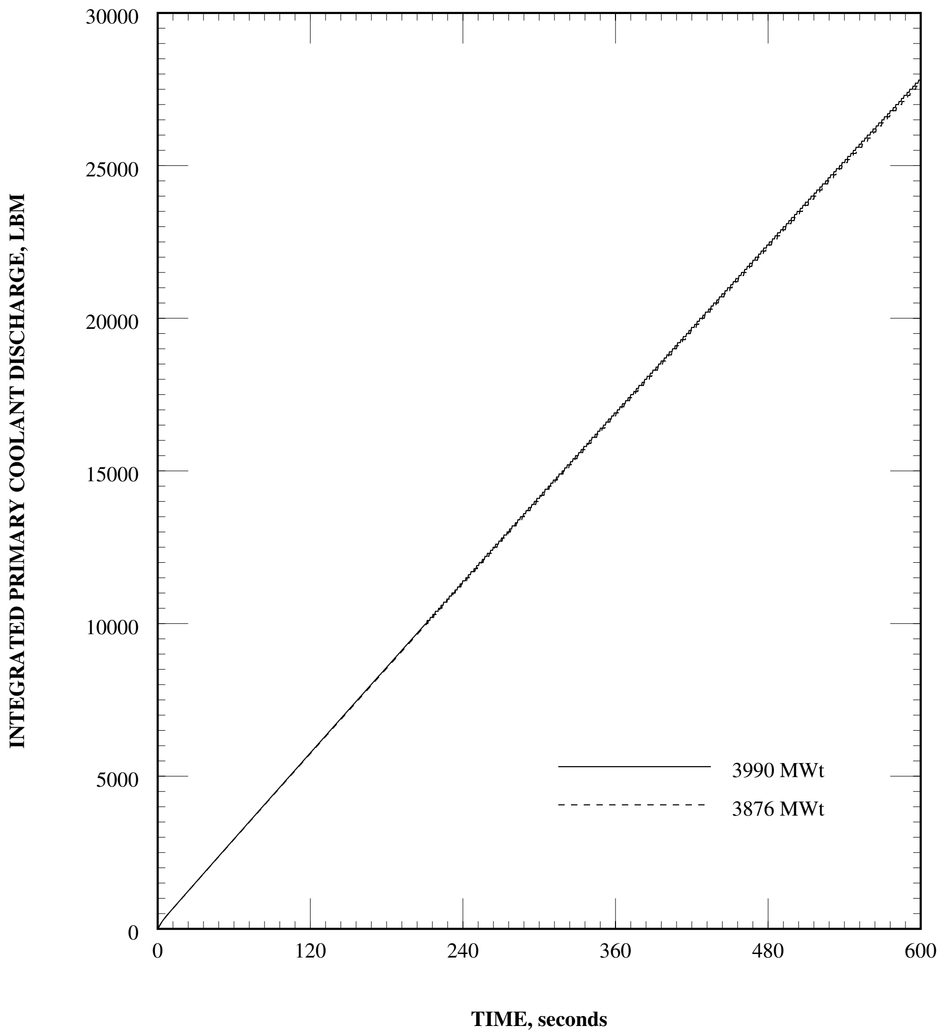


Figure 6.3-208
DBLLOCUS Event - Integrated Primary Coolant Discharge vs. Time



Section 6.3.6.3 Steam Generator Tube Rupture

Section 6.3.6.3.1 Steam Generator Tube Rupture without a Concurrent Loss of Offsite Power

As described in UFSAR Section 15.6.3, the SGTR accidents with and without concurrent LOP both fall in the category of infrequent event for an accident Generated Iodine Spiking factor (GIS) case or a limiting fault event for a Pre-existing Iodine Spiking factor (PIS) case. The results of a SGTR without LOP are less limiting than that of the SGTR with a concurrent LOP (SGTRLOP) event. The SGTRLOP event includes a loss of forced circulation and condenser. These two factors predominantly result in dose consequences for the SGTRLOP being worse than the SGTR.

Section 6.3.6.3.2 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power and a Single Failure

Section 6.3.6.3.2.1 Identification of Event and Causes

As described in UFSAR Section 15.6.3, the SGTR accident is a penetration of the barrier between the RCS and the main steam system, and results from the failure of a SG U-tube. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. Before turbine trip, the radioactivity is transported through the turbine to the condenser where the noncondensable radioactive materials would be released via the condenser air removal pumps. Following reactor/turbine trip, the MSSVs open to control the main steam system pressure. The operator can isolate the affected SG any time after reactor trip occurs. Since a LOP occurs concurrent with a SGTR, electrical power would be unavailable for the station auxiliaries. The plant would experience a loss of the following:

- turbine load,
- normal FW flow,
- forced RCS flow, and
- condenser.

With the SBCS unavailable, NSSS cooldown is accomplished by use of AFW flow and ADVs. Heat removal must be accomplished by natural circulation, resulting in a higher core outlet temperature for much of the transient. The higher core outlet temperature, as well as steaming to the atmosphere via the ADV, contributes to higher offsite doses.

This analysis assumes that the plant is challenged by a SGTR that includes additional events and failures beyond those postulated by the SRP Section 15.6.3. In addition to the assumptions of the SRP (LOP, accident meteorology, iodine spiking, etc.), this analysis postulates that the operators trip the plant manually prior to reaching reactor trip setpoint, open an ADV on the affected SG fully and that the ADV sticks full open for the duration of the transient as the single failure.

Radiation monitors, initiate alarms and inform the operator of abnormal activity levels. The radiation monitors facilitate diagnosis of the SGTR accident and that corrective operator action is required. Additional diagnostic information is provided by RCS pressure and pressurizer level response indicating a leak, and by level response in the affected SG.

Experience with SGs indicates that the probability of complete severance of the Inconel vertical U-tubes is remote. The more probable modes of failure result in considerably smaller penetrations of the pressure barrier. These involve the formation of etch pits or small cracks in the U-tubes or cracks in the welds joining the tubes to the tube sheet.

The most limiting SGTR event is for a leak flow equivalent to a double-ended rupture of a U-tube at full power conditions.

<u>Section 6.3.6.3.2.2</u> <u>Acceptance Criteria</u>

As defined in the SRP Section 15.6.3, the specific acceptance criteria for this event are:

- The plant is considered adequately designed against a SG tube failure, and the primary and secondary coolant activities adequately limited, if calculations show that the resulting doses at the EAB and Low Population Zone (LPZ) boundaries are less than a small fraction (less than 10%) of the 10 CFR Part 100 exposure guidelines, and are within 10 CFR Part 100 guidelines for the case of a PIS or for the case of one rod held out of the core.

The SGTRLOP single failure is considered a limiting fault event. Since the consideration of a single failure was an additional requirement to the analysis guidelines stipulated in the SRP, the acceptance criteria for the SGTR with a LOP and a fully stuck open ADV (SGTRLOP single failure) is:

- dose consequences at the EAB and the LPZ for the PIS and GIS case are within the 10 CFR Part 100 exposure guidelines.

<u>Section 6.3.6.3.2.3</u> <u>Description of Analysis</u>

The NSSS response to a SGTRLOP single failure event was simulated using the CENTS code. The input parameters and initial conditions were selected to maximize the dose consequences.

<u>Section 6.3.6.3.2.3.1</u> <u>Transient Simulation</u>

The system was initialized at 102% power using the most limiting initial parameters. At time equal zero, the SGTR was simulated by a break at the top of the tube sheet at the hot side. This results in the pressurizer level and pressure decreasing, letdown flow going to minimum and the backup heaters energizing. The reactor trip setpoint based on CPC hot leg saturation margin trip was approached. However, an early manual reactor trip was simulated at 100 sec in order to maximize the integrated steam flow out of the stuck open ADV. A LOP occurs due to grid instability three seconds after turbine trip.

After reactor trip, stored and fission product decay heat energy must be removed by the RCS and main steam systems. In the absence of forced RCS flow, convective heat transfer out of the reactor core is supported by natural circulation. Initially, the water inventory in the SGs is used to cooldown the RCS with the resultant steam released to atmosphere via the MSSVs.

Two minutes post-trip, the operator takes action to partially open an ADV on each SG. At which point an ADV on the affected SG fails full open and remains open for the duration of the event. AFAS actuation results in AFW flow initiated to both SGs. MSIS actuation on LSGP causes a subsequent termination of AFW to the affected SG due to pressure differential lockout.

SGTR with a stuck open ADV results in a dual event, a SGTR concurrent with an Excess Steam Demand. The functional recovery procedures contain explicit instructions to help the operators manage the cooldown and keep the SG tubes covered for this dual event. The timing of the operator actions in the model are based on the ANS/ANSI-N58.8-1984 which specify response times for safety related operator actions. The major operator actions were simulated to maximize the analysis radiological dose results, and to bound the radiological doses from a postulated SGTR with a stuck open ADV.

The major post-trip EOP assumptions regarding operator actions are the following:

1. Preclude challenge to MSSVs

The analysis assumes operator action to open the ADVs on both SGs to preclude a challenge to the MSSVs two minutes after the reactor trip. The ADVs are used due to the unavailability of the SBCS due to LOP. The ADV on the affected SG remains full open.

2. Stabilize the Plant and Diagnose the Event

Seven minutes post-trip, the operator action is credited in closing the unaffected SG ADV to prevent excessive cooldown. This action is consistent with expected operator action while ensuring adequate RCS heat removal. The analysis assumes that a diagnosis of a SGTR with an ESD will take 10-12 minutes post-trip at which point operators follow guidance from the procedures.

3. Functional Recovery Strategy

After 15 minutes post-trip, the operators are assumed to override the AFAS on the affected SG and establish dedicated flow of 1360 gpm until SG level recovers above 40% NR. This action is consistent with the procedural strategy in response to indications of a SGTR with uncontrolled steaming to atmosphere from the affected SG.

1360 gpm flow is the minimum AFW flow in the range instructed in the procedures (1360 - 1600 gpm).

4. Post-Tube Coverage Strategy

After the affected SG is above 40% NR, the operators are assumed to initiate AFW flow of 500 gpm to the unaffected SG. This calculation assumes that the operator action maintains the affected SG level between 40% to 60% NR for the duration of the event by adjusting AFW flow as necessary.

5. Maintain adequate RCS inventory, HPSI throttle criteria

Besides maintaining adequate subcooling, the operator is simultaneously responsible for assuring adequate RCS inventory is maintained. Specifically, the EOPs require the operator to retain specified levels in the pressurizer and the upper head before throttling back the HPSI flow. Accordingly, the pressurizer level and the subcooling margin in the analysis are maintained above the level required by the EOPs.

6. Cooldown and Depressurize to SCS Entry Conditions

This analysis assumes that the cooldown and depressurization of the RCS is primarily accomplished due to the stuck open ADV. Operator actions minimize the cooldown by closing the unaffected SG ADV and use of a minimum AFW flow of 500 gpm. Cooldown rate during this phase was restricted to within those allowed by the EOP guideline. Although SCS entry conditions were reached before eight hours, the event was simulated for 8 hours to maximize the dose consequences

Section 6.3.6.3.2.4 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-49 contains the initial conditions used for the SGTRLOP single failure event.

The following assumptions were 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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The CPC T_{hot} saturation margin trip is enabled. However, a conservative manual trip based on operator action is credited at 100 seconds, before the CPC trip.
3. The RCPs, main FW pumps, PPCS, and charging pumps are assumed unavailable post-LOP. Class 1E back-up heaters at a capacity of 250 kW are credited for maintaining harsh Subcooling Margin (SCM) criteria.
4. Auxiliary spray was assumed unavailable, use of pressurizer head vents was credited for de-pressurization.

Table 6.3-49
Parameters Used for SGTRLOP Single Failure Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temp (°F)	562	568
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	21.85	21.85
Initial SG level (ft)	26.75	25.7
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.0	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	518	518
Plugged SG tubes (% of tubes/SG)	0	0
SGTR break location	at the tube sheet	at the tube sheet
Single failure	stuck open ADV	stuck open ADV
LOP	yes	yes

Section 6.3.6.3.2.5 Results

Table 6.3-50 presents the sequence of events for the simulation of the SGTRLOP single failure. The representative behaviors of NSSS parameters following the SGTR single failure are presented in Figure 6.3-209 through Figure 6.3-223.

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 timing and operator actions between the two plant configurations are similar. The dose consequences are more severe for the larger SGs due to the height of the U-tubes and the extra time required to submerge these U-tubes during SG refill.

Radiological consequences for this event are presented in Section 6.4.6.2.

Table 6.3-50
Sequence of Events for the SGTRLOP Single Failure Event
(Page 1 of 2)

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0	0	SGTR occurs	---	---
43	43	Letdown control valve throttled to minimum (gpm)	35	35
73	79	Backup pressurizer heaters energized (psia)	2275	2275
100	100	Manual reactor trip	---	---
101	101	Reactor trip breakers open	---	---
101	101	Turbine trip occurs	---	---
101.6	101.6	Scram CEAs begin falling	---	---
104	104	LOP occurs	---	---
104	102	MSSVs open (psia)	1227	1227
108	104.3	Maximum SG pressures (psia)	1296	1299
113	109	SG level reaches AFAS setpoint in unaffected SG (%WR)	20	20
114	110	SG level reaches AFAS setpoint in affected SG (%WR)	20	20
159	155	AFW initiated to unaffected SG (gpm)	779	779
160	156	AFW initiated to the affected SG (gpm)	779	779
165	162	MSSVs close (psia)	1104	1104
220	220	Operator initiates plant cooldown by opening one ADV on each SG; ADV on affected SG instantaneously opens fully	---	---
265	245	Pressurizer pressure reaches SIAS setpoint (psia)	1837	1837
265	245	SI flow initiated with no delay	---	---
266	251	MSIS actuation, secondary pressure (psia)	940	1005
283	268	AFAS 1 lockout on high DP (psid)	100	100
352	317	Voids begin to form in the upper head	---	---
520	520	Operator shuts ADV on the unaffected SG	---	---
648	608	Voids collapsed in the upper head	---	---

Table 6.3-50
Sequence of Events for the SGTRLOP Single Failure Event

(Page 2 of 2)

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
633	847	AFAS 2 reset on high SG level (%WR)	30%	30%
1000	1000	Operator overrides the DP lockout and initiates dedicated AFW flow 1360 gpm to affected SG	---	---
1900	1900	Operator opens pressurizer head vent	---	---
2593	2779	Level in the affected SG is above the top of U-tubes; operator secures AFW flow to affected SG and initiates AFW to unaffected SG	---	---
2716	2716	HPSI flow throttled to maintain SCM less than the limit	---	---
11850	12610	Class back-up heaters energized to maintain target harsh SCM criteria (°F)	85	85
25436	26014	ADV opened in the unaffected SG in preparation of approaching SCS entry conditions	---	---
26640	27260	SCS entry conditions reached in the affected loop (psia/°F)	> 395/350	> 395/350
28800	28800	Operator activates SCS system (hr)	8	8

Section 6.3.6.3.2.6 Conclusions

For the SGTRLOP single failure event, all acceptance criteria are met. Dose consequences at the EAB and the LPZ for the PIS and GIS case are within the 10 CFR Part 100 exposure guidelines.

Section 6.3.6.3.3 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

Section 6.3.6.3.3.1 Identification of Event and Causes

As described in UFSAR Section 15.6.3, this transient is similar to that described in previous section for SGTRLOP single failure with the exception of ADV remaining open. It assumes that the plant is challenged by a SGTR. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. Before turbine trip, the radioactivity is transported through the turbine to the condenser where the noncondensable radioactive materials would be released via the condenser air removal pumps. Following reactor/turbine trip, the MSSVs open to control the main steam system pressure. The operator can isolate the damaged SG any time after reactor trip occurs. Since a LOP occurs concurrent with a SGTR, electrical power would be unavailable for the station auxiliaries. The plant would experience a loss of the following:

- turbine load,
- normal FW flow,
- forced RCS flow, and
- condenser.

With the SBCS unavailable, NSSS cooldown is accomplished by use of AFW flow and ADVs. Heat removal must be accomplished by natural circulation, resulting in a higher core outlet temperature for much of the transient. The higher core outlet temperature as well as steaming to the atmosphere via use of ADV contributes to higher offsite doses. Thus with the SGTRLOP being more limiting in term of dose consequences than the SGTR with offsite power available, the SGTRLOP is described in detail.

Radiation monitors that initiate alarms and inform the operator of abnormal activity levels, facilitate diagnosis of the SGTR accident and that corrective operator action is required. Additional diagnostic information is provided by RCS pressure and pressurizer level response indicating a leak, and by level response in the affected SG.

Experience with SGs indicates that the probability of complete severance of the Inconel vertical U-tubes is remote. The more probable modes of failure result in considerably smaller penetrations of the pressure barrier. They involve the formation of etch pits or small cracks in the U-tubes or cracks in the welds joining the tubes to the tube sheet.

The most limiting SGTR event is for a leak flow equivalent to a double-ended rupture of a U-tube at full power conditions.

Section 6.3.6.3.3.2 Acceptance Criteria

As defined in the SRP Section 15.6.3, the specific acceptance criteria for this event are:

- The plant is considered adequately designed against a SG tube failure, and the primary and secondary coolant activities adequately limited, if calculations show that the resulting doses at the EAB and LPZ boundaries are less than a small fraction (less than 10%) of the 10 CFR Part 100 exposure guidelines, and are within 10 CFR Part 100 guidelines for the case of a PIS or for the case of one rod held out of the core.

Section 6.3.6.3.3.3 Description of Analysis

The NSSS response to a SGTRLOP event is simulated using the CENTS code. The input parameters and initial conditions were selected to maximize the dose consequences and demonstrate that the applicable SRP criteria were met.

Section 6.3.6.3.3.3.1 Transient Simulation

The system is initialized at 102% power using the most limiting initial parameters. At time equal zero, the SGTR was simulated by a break at the top of the tube sheet at the hot side. This results in the pressurizer level and pressure to decrease, letdown flow going to minimum and the third charging pump starts. Pressurizer level reaches the low-pressure level heater cut-off de-energizing all heaters, and accelerating the primary depressurization. The CPC reactor trip occurs on approach to hot leg saturation, a turbine trip follows within one second. A LOP occurs due to grid instability three seconds after turbine trip.

After reactor trip, stored and fission product decay heat energy must be removed by the RCS and main steam systems. In the absence of forced RCS flow, convective heat transfer out of the reactor core is supported by natural circulation. Initially, the water inventory in the SGs is used to cooldown the RCS with the resultant steam released to atmosphere via the MSSVs.

The EOPs contain explicit instructions to help the operator manage the cooldown following a SGTR event. Accordingly, the required operator actions to mitigate the effects of the SGTR event, and bring the plant to SCS entry conditions have been simulated based on the EOPs with bounding assumptions. The timing of the operator actions in the model are based on the ANS/ANSI-N58.8-1984 (Reference 6-35) that specifies response times for safety related operator actions. The major operator actions were simulated to maximize the analysis radiological dose results and, therefore, bound the radiological doses from a postulated SGTR with a LOP.

The major post-trip EOP analysis assumptions regarding operator actions are the following:

1. Preclude challenge to MSSVs

The analysis conservatively assumes operator action to open the ADVs (on both SGs) two minutes after the reactor trip. The ADVs are used due to the unavailability of the SBCS due to LOP.

2. Diagnose the event and stabilize the plant

Procedures are oriented towards quickly diagnosing the event and stabilizing the RCS at a temperature that precludes a challenge to the MSSVs. The analysis assumes this diagnosis and stabilization period will take 22 minutes that is consistent with ANSI/ANS standards for this category of event. The operator is assumed to use the ADVs (on both SGs) and the AFW system to maintain the post-trip T_{cold}. Both essential AFW pumps are operable, but only approximately half of their available capacity is assumed to be delivered to SGs during this interval. This assumption is conservative because it results in a continuous steaming path from the affected SG up to the time of initial isolation.

3. Cooldown the RCS before isolation of affected SG

After the 22-minute diagnosis and stabilization period, the operators are assumed to cool the RCS at a cooldown rate of approximately 80 ^{o}F/hr. The cooldown continues via the ADVs on both SGs until the affected T_{hot} reaches the isolation temperature. A conservatively lower temperature is assumed in the analysis in order to delay isolation of the affected SG. Additionally, during this period, AFW would be delivered to each SG as needed in order to maintain the level in both SGs per the requirements in the EOPs.

4. Manual MSIS

During the cooldown phase the operator is assumed to initiate a manual MSIS per EOP guidelines due to LOP. The initiation of MSIS is conservatively delayed as it maximizes dose consequences.

5. Isolate the affected SG

The operator is assumed to isolate the affected SG after the affected loop temperature has reached the isolation temperature of 515 ^{o}F. This isolation criterion is conservative for the EOP guidelines of 540 ^{o}F. During the cooldown phase, primary pressure is reduced with the aid of pressurizer head vent. Use of variable harsh SCM criteria is used to maximize leak rate and hence the dose consequences.

6. Cooldown the RCS

The analysis assumes that post-isolation of the affected SG, cooldown to SCS is conducted via feeding and steaming the unaffected SG. The affected SG level increases approaching full conditions. However, no liquid enters the main steam lines. Throttling HPSI as necessary and use of pressurizer head vents reduce primary pressure. After the leak is reduced, primary to secondary pressure differential is minimized to less than 50 psid to facilitate leak isolation and to ensure that the reverse leak is kept to a minimum.

The natural circulation cooling with the unaffected loop is maintained less than 30 ^{o}F/hr until the entry conditions for SCS is reached at 8 hours.

7. Maintain adequate RCS inventory, HPSI throttle criteria

Besides maintaining adequate subcooling, the EOPs require the operator to assure adequate RCS inventory, specifically, to retain minimum specified levels in the pressurizer and the upper head before throttling back the HPSI flow. Accordingly, the pressurizer level in the analysis is maintained above the level required by the EOPs.

<u>Section 6.3.6.3.3.4</u> <u>Input Parameters, Initial Conditions, and Assumptions</u>

Table 6.3-51 contains the initial conditions used for the SGTRLOP event.

The following assumptions were 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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. The RCPs, main FW pumps, PPCS, and charging pumps are assumed unavailable post-LOP. Class 1E back-up heaters at a capacity of 250 kW are credited for maintaining harsh SCM criteria.
3. Auxiliary spray is assumed unavailable; use of pressurizer head vents was credited for de-pressurization.

Table 6.3-51
Parameters Used for SGTRLOP Event

PARAMETER	Value	
	3876 MW_t	3990 MW_t
Initial core power (% of rated)	102	102
Initial core inlet temp (°F)	562	568
Initial pressurizer pressure (psia)	2325	2325
Initial RCS flow (% of design)	95	95
Initial pressurizer level (ft)	21.85	21.85
Initial SG level (ft)	26.75	25.7
MTC ($\Delta\rho$/°F)	-4.0E-04	-4.0E-04
FTC	least negative	least negative
Kinetics	minimum β	minimum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.0	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	518	518
Plugged SG tubes (% of tubes/SG)	0	0
SGTR break location	at the tube sheet	at the tube sheet
Single failure	none	none
LOP	yes	yes

Section 6.3.6.3.3.5 Results

Table 6.3-52 presents a sequence of events for the simulation of the SGTRLOP event. The representative behaviors of NSSS parameters following the SGTR single failure are presented in Figure 6.3-224 through Figure 6.3-240.

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 timing and operator actions between the two plant configurations are similar. The timing of the reactor trip and subsequent actions is different due to the higher initial temperature associated with the PUR operating point.

Radiological consequences for this event are presented in Section 6.4.6.2.

Table 6.3-52
Sequence of Events for the SGTRLOP Event
(Page 1 of 3)

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
0	0	SGTR occurs	---	---
43	43	Letdown control valve throttled to minimum (gpm)	35	35
73	79	Backup pressurizer heaters energized (psia)	2275	2275
338	350	Third charging pump is turned on	---	---
406	420	Pressurizer heaters de-energized on low level in the pressurizer (%)	25	25
898	759	Reactor trip reached on CPC hot leg saturation margin reached (°F)	8	8
899	760	Reactor trip breakers open	---	---
899	760	Turbine trip occurs	---	---
899.6	760.6	Scram CEAs begin falling	---	---
902	763	LOP occurs	---	---
903	761	MSSVs open (psia)	1227	1227
907	763	Maximum SG pressure (psia)	1297	1299
911	767	SG level reaches AFAS setpoint in unaffected SG (%WR)	20	20
918	780	Pressurizer pressure reaches SIAS setpoint (psia)	1837	1837
918	780	SIAS generated, SI flow initiated	---	---
920	780	Pressurizer empties	---	---
941	802	Voids begin to form in the upper head	---	---
957	813	AFW initiated to unaffected SG (gpm)	779	650
---	815	SG level reaches AFAS setpoint in affected SG (%WR)	20%	20%
---	861	AFW initiated to the affected SG (gpm)	---	650
962	878	MSSVs close (psia)	1104	1104
1004	864	Voids collapsed in the upper head	---	---
1018	878	Operator opens one ADV in each SG to prevent cycling of safeties	---	---

Table 6.3-52
Sequence of Events for the SGTRLOP Event

(Page 2 of 3)

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
1035	891	Pressurizer begins to refill	---	---
1138	998	Operator takes manual control of the AFW system, feeds each SG at the rate of 325 gpm, and stabilizes the plant	---	---
2219	2081	Operator initiates plant cooldown at the rate of 80 °F/hr, by adjusting the ADVs and using one AFW pump per SG (gpm)	650	650
2340	2202	Operator opens pressurizer head vents	---	---
2466	2324	Operator initiates a manual MSIS	---	---
2701	2571	Operator reduces ADV flow to slow cooldown rate	---	---
3785	4012	Operator throttles back HPSI flow to maintain RCS inventory control	---	---
3906	3170	Operator cycles class backup heaters (kW) to maintain SCM	250	250
5642	5650	Operator isolates the affected SG, at the analytical affected loop temperature (°F)	515	515
5762	5775	Operator opens the pressurizer head vent to resume depressurization	---	---
6724	6551	Affected SG dome temperature exceeds affected loop T_{hot} temperature; eliminates leak flashing in the affected SG	---	---
9361	7656	Onset of reverse heat transfer in the affected loop: T_{cold} greater than the loop T_{hot}	---	---
11585	---	Leak isolated, operator action maintains RCS pressure to affected SG ΔP minimum (psid)	50	50
20849	15929	Operator opens the first unaffected SG ADV full open	---	---
22654	22042	Operator opens the second unaffected SG ADV full open	---	---
25059	18869	Operator increases AFW to 160 gpm in the unaffected SG (gpm)	160	160

Table 6.3-52
Sequence of Events for the SGTRLOP Event

(Page 3 of 3)

Time (sec)		Event	Value	
3876 MW_t	3990 MW_t		3876 MW_t	3990 MW_t
26260	24591	SCS entry conditions reached in the unaffected loop (psia/°F)	> 395/335	> 395/335
28800	28800	Operator activates SCS system (hr)	8	8

Section 6.3.6.3.3.6 Conclusions

For the SGTRLOP event, all acceptance criteria are met. Dose consequences at the EAB and LPZ boundaries are less than a small fraction of the 10 CFR Part 100 exposure guidelines, and are within 10 CFR Part 100 guidelines for the case of a PIS or for the case of one rod held out of the core.

Figure 6.3-209
SGTRLOP Single Failure Event - Core Power vs. Time

(Sheet 1 of 2)

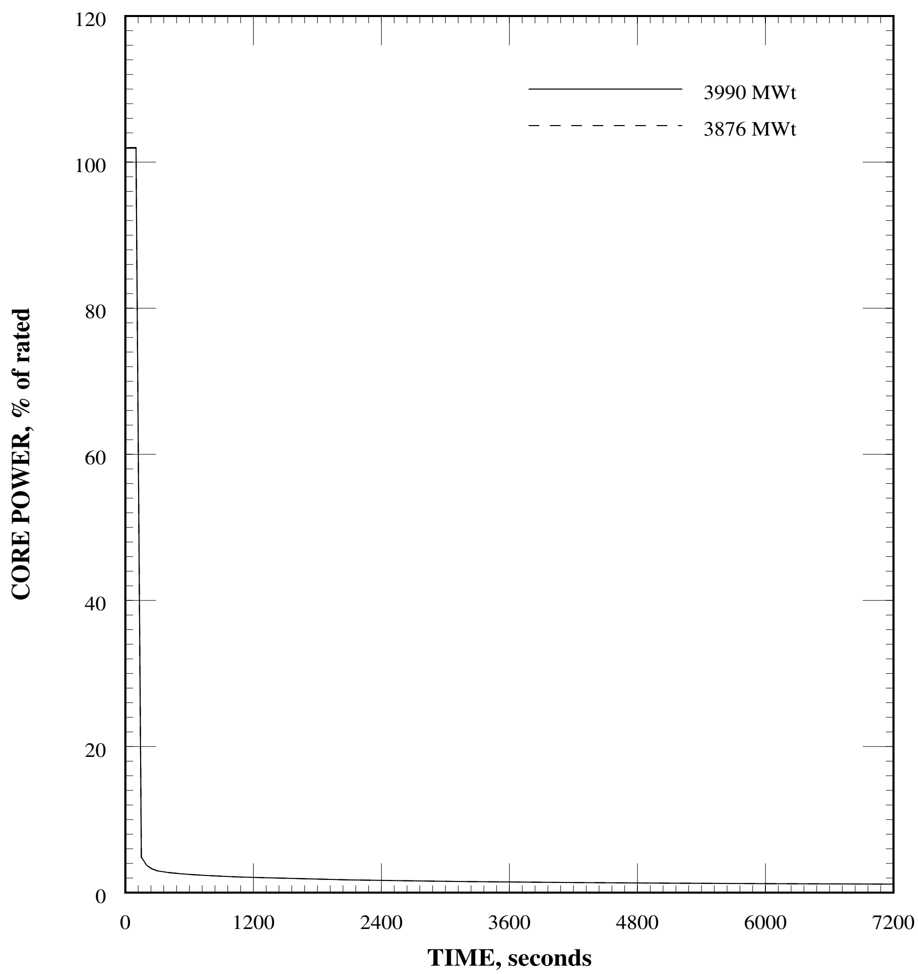


Figure 6.3-209
SGTRLOP Single Failure Event - Core Power vs. Time

(Sheet 2 of 2)

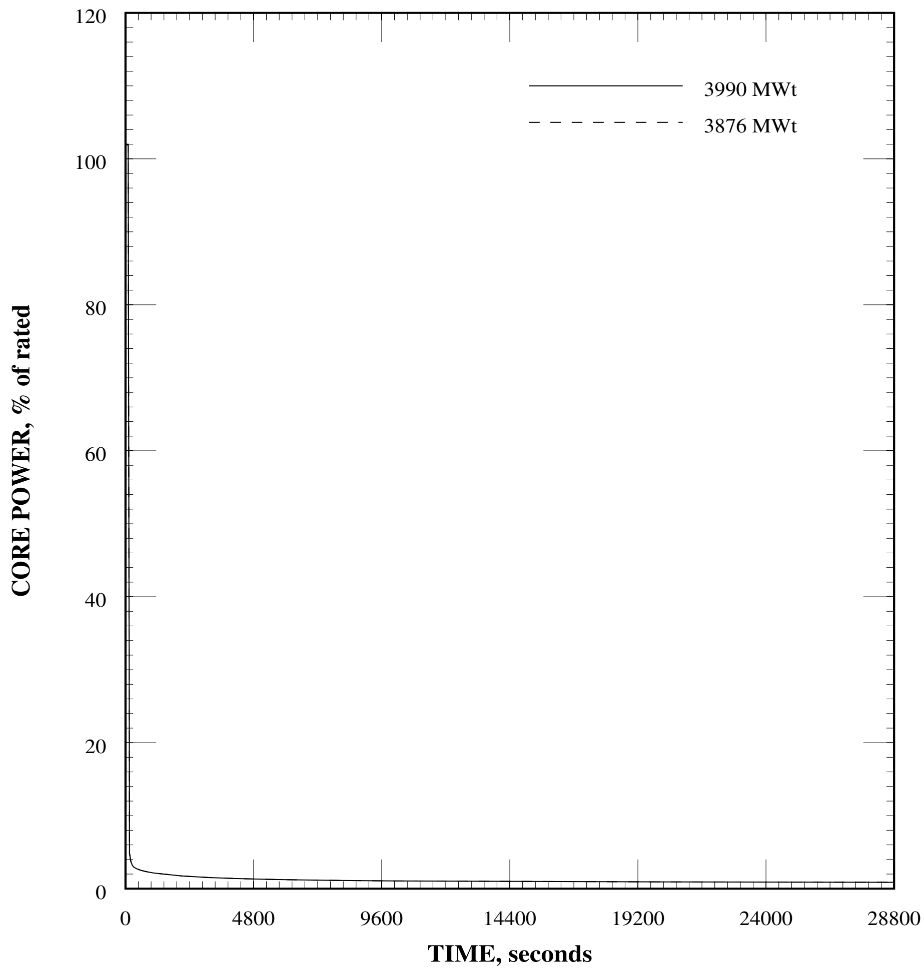


Figure 6.3-210
SGTRLOP Single Failure Event - RCS Pressure vs. Time
(Sheet 1 of 2)

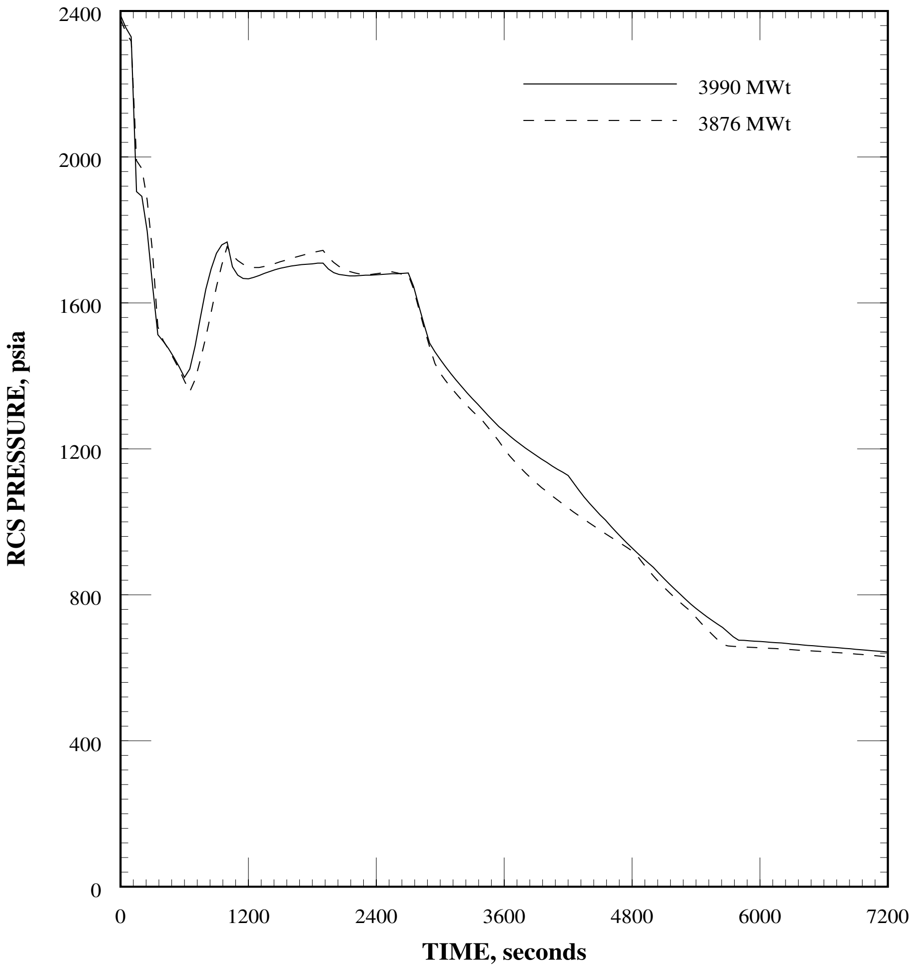


Figure 6.3-210
SGTRLOP Single Failure Event - RCS Pressure vs. Time

(Sheet 2 of 2)

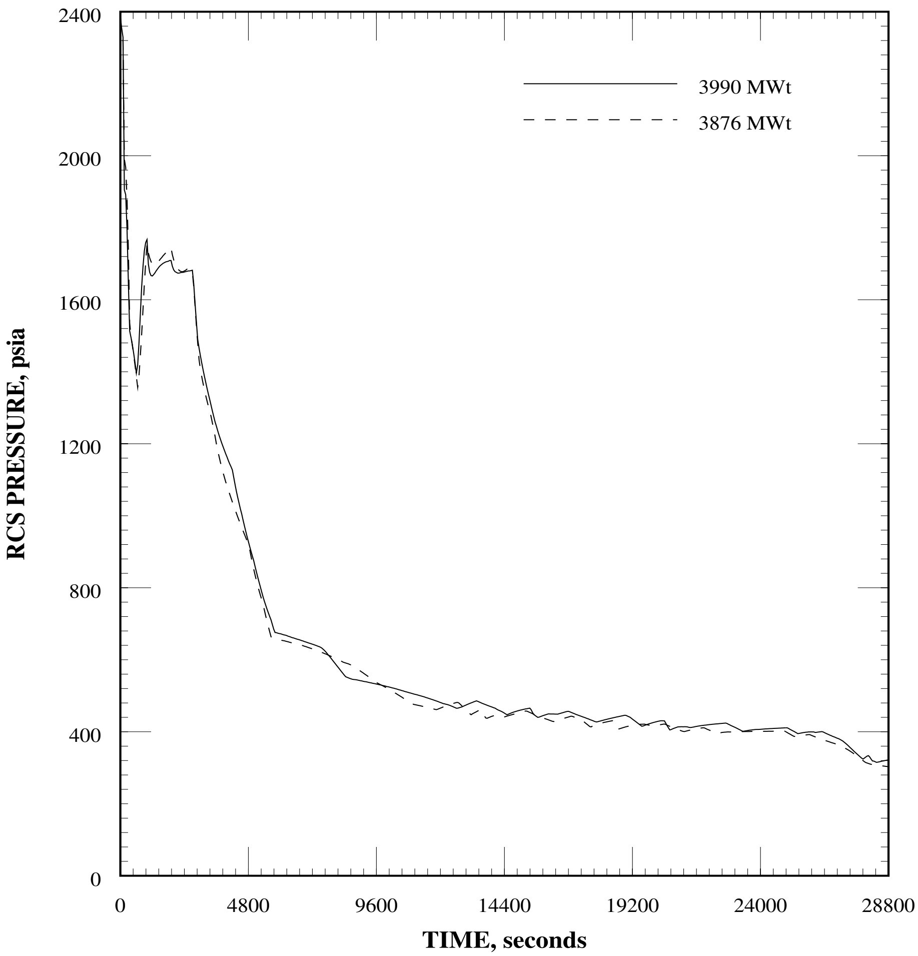


Figure 6.3-211
SGTRLOP Single Failure Event - RCS Temperatures Affected Loop vs. Time

(Sheet 1 of 2)

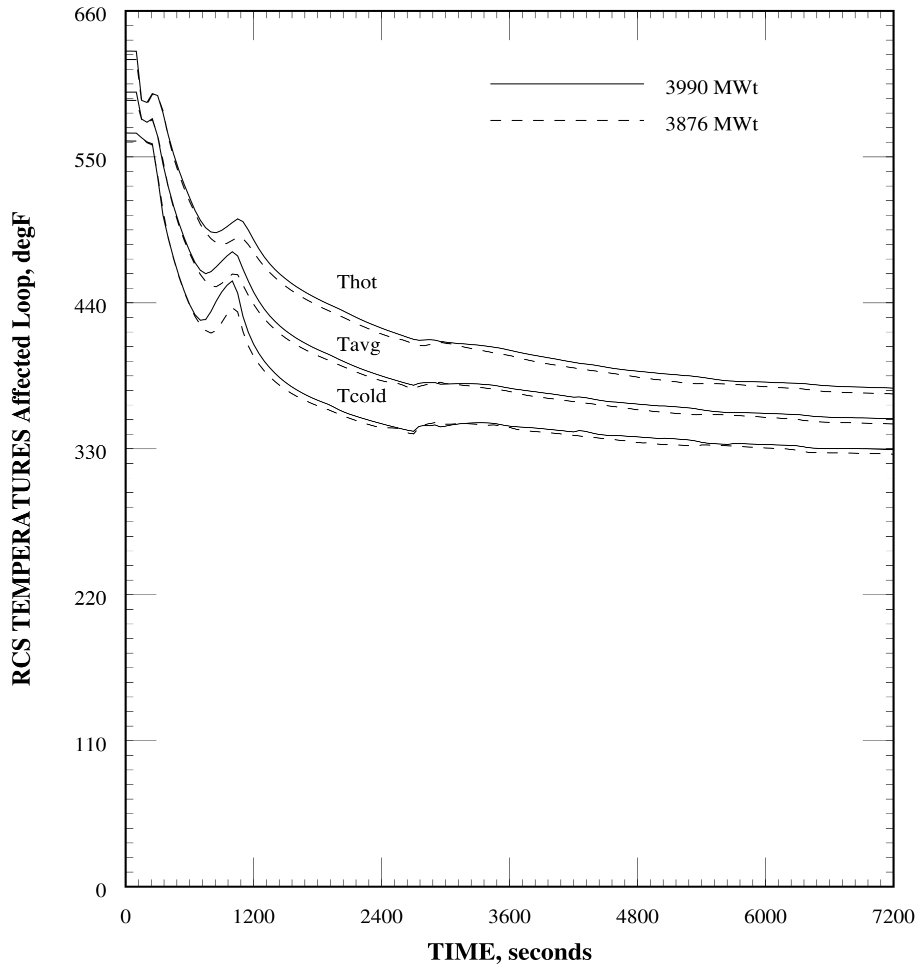


Figure 6.3-211
SGTRLOP Single Failure Event - RCS Temperatures Affected Loop vs. Time

(Sheet 2 of 2)

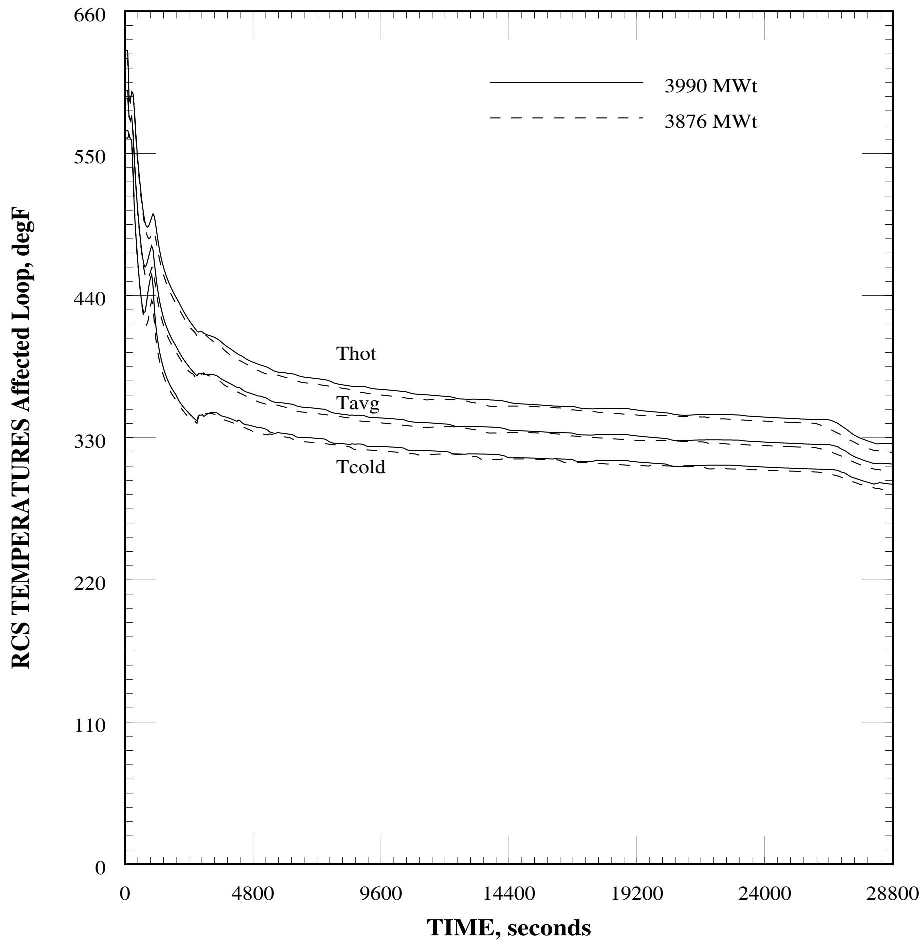


Figure 6.3-212
SGTRLOP Single Failure Event - Upper Head Temperature vs. Time
(Sheet 1 of 2)

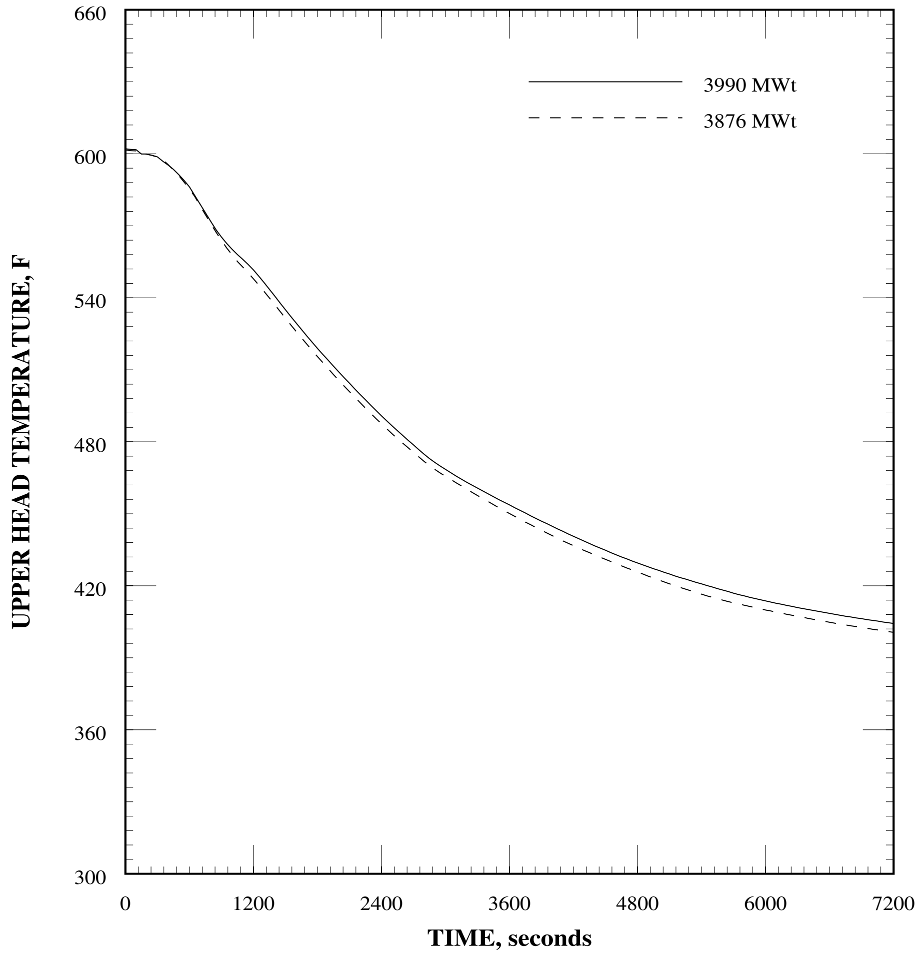


Figure 6.3-212
SGTRLOP Single Failure Event - Upper Head Temperature vs. Time

(Sheet 2 of 2)

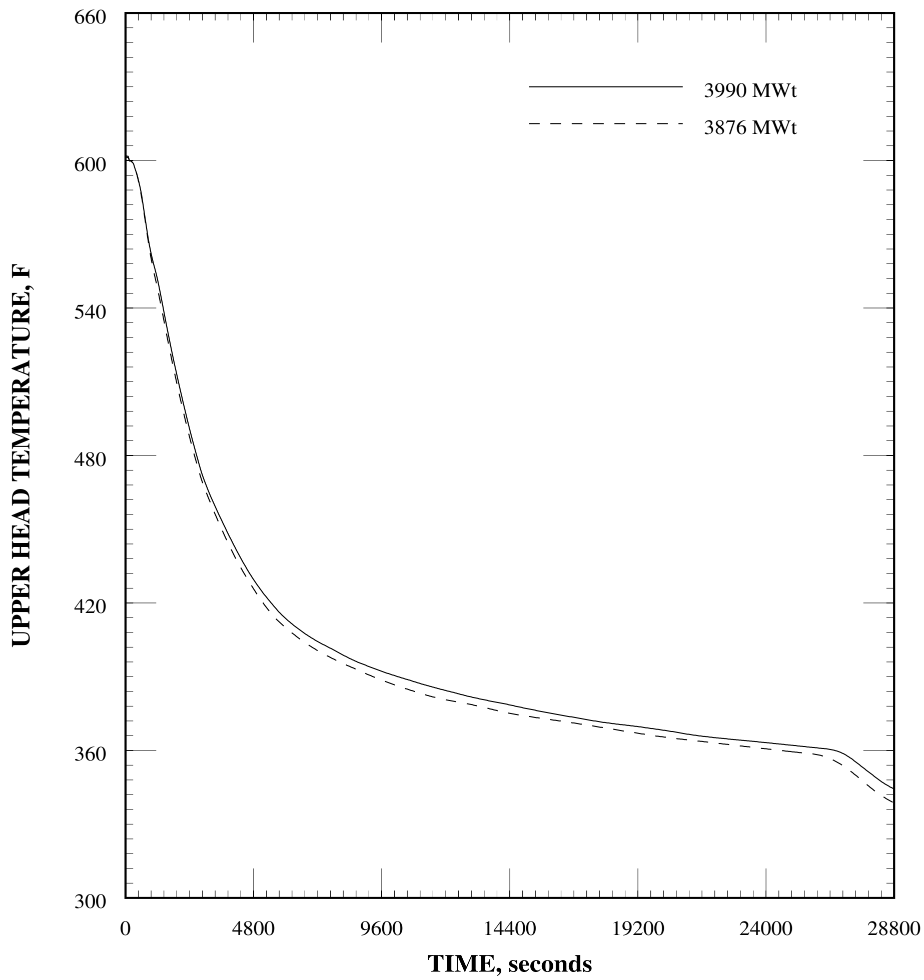


Figure 6.3-213
SGTRLOP Single Failure Event - Pressurizer Liquid Volume vs. Time
(Sheet 1 of 2)

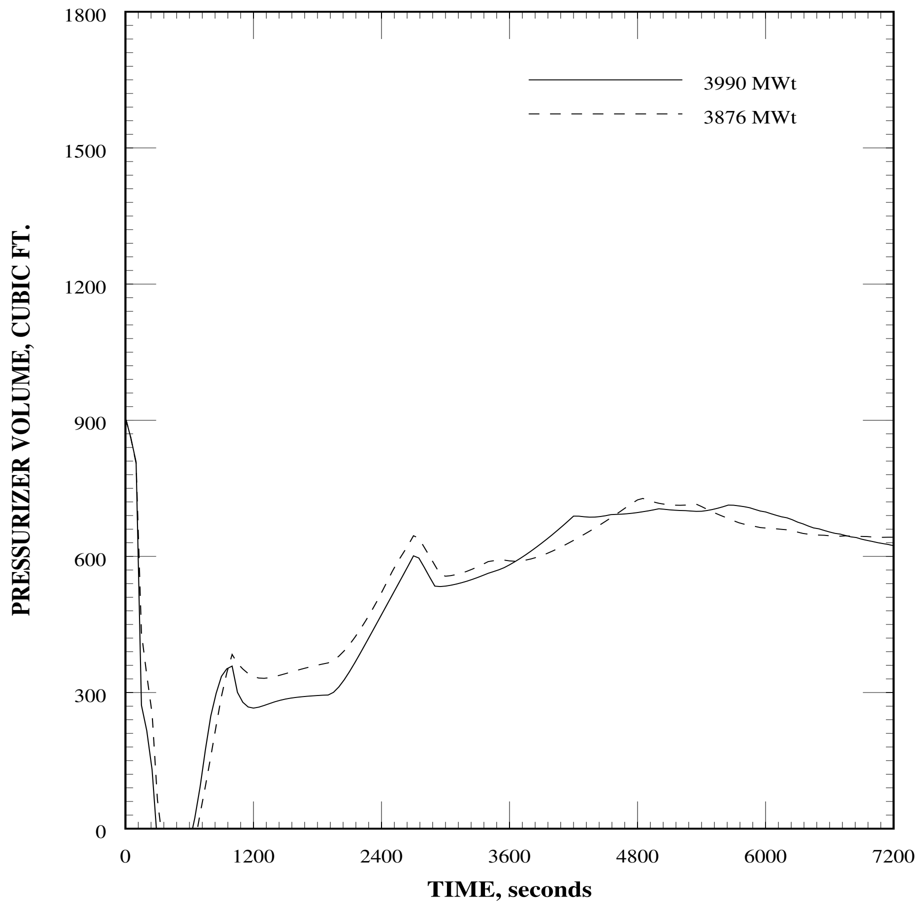


Figure 6.3-213
SGTRLOP Single Failure Event - Pressurizer Liquid Volume vs. Time

(Sheet 2 of 2)

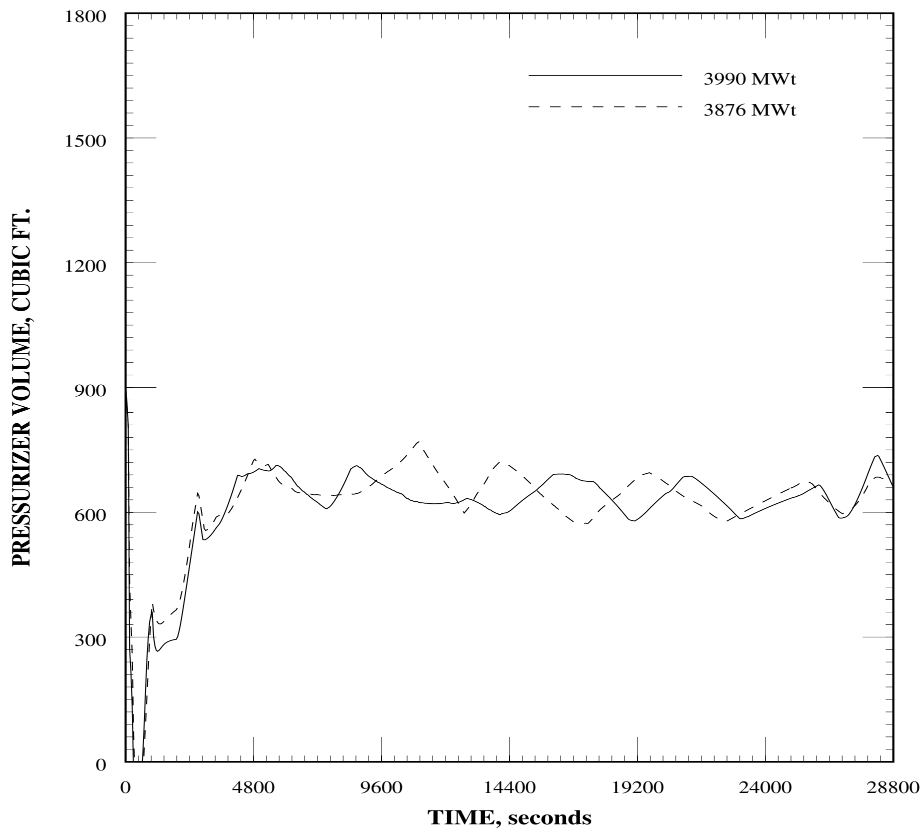


Figure 6.3-214
SGTRLOP Single Failure Event - Upper Head Level vs. Time

(Sheet 1 of 2)

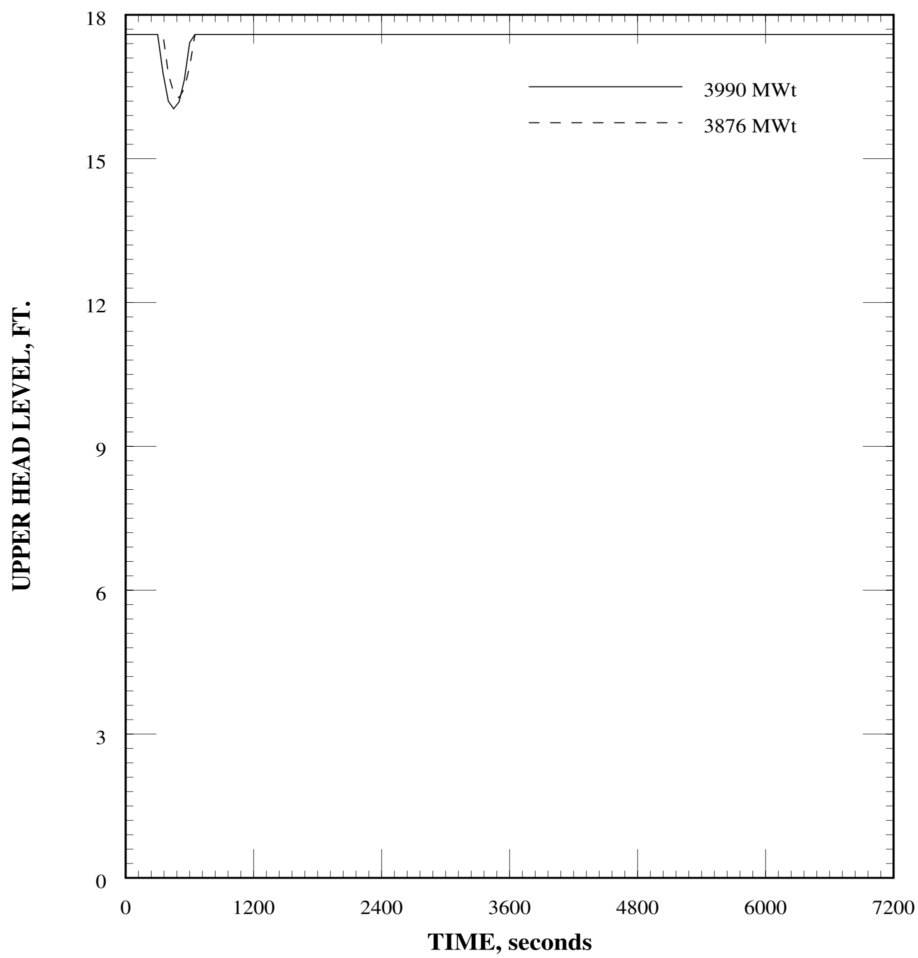


Figure 6.3-214
SGTRLOP Single Failure Event - Upper Head Level vs. Time

(Sheet 2 of 2)

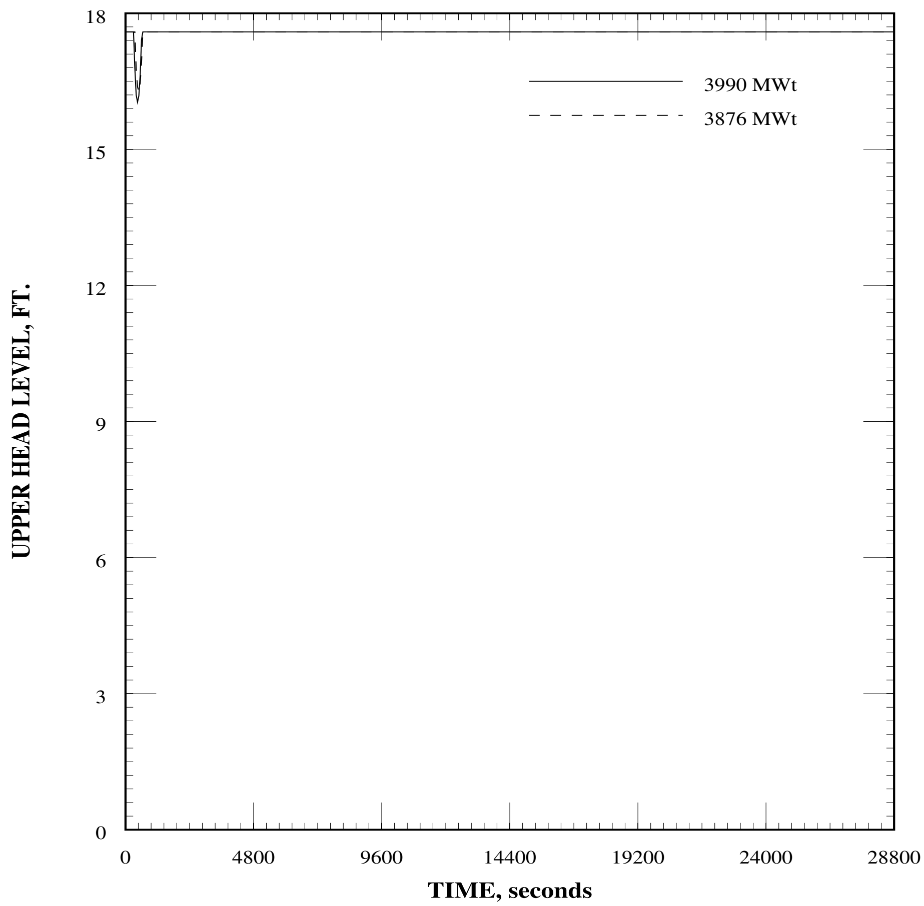


Figure 6.3-215
SGTRLOP Single Failure Event - RCS Liquid Mass vs. Time
(Sheet 1 of 2)

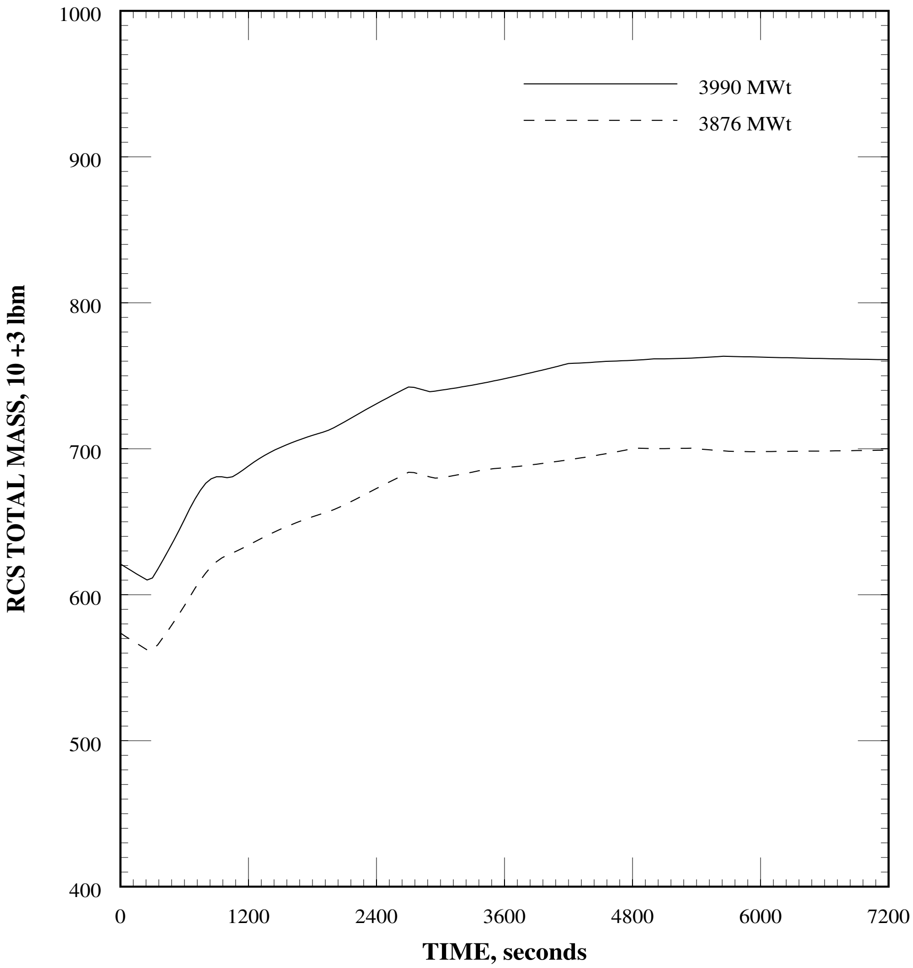


Figure 6.3-215
SGTRLOP Single Failure Event - RCS Liquid Mass vs. Time

(Sheet 2 of 2)

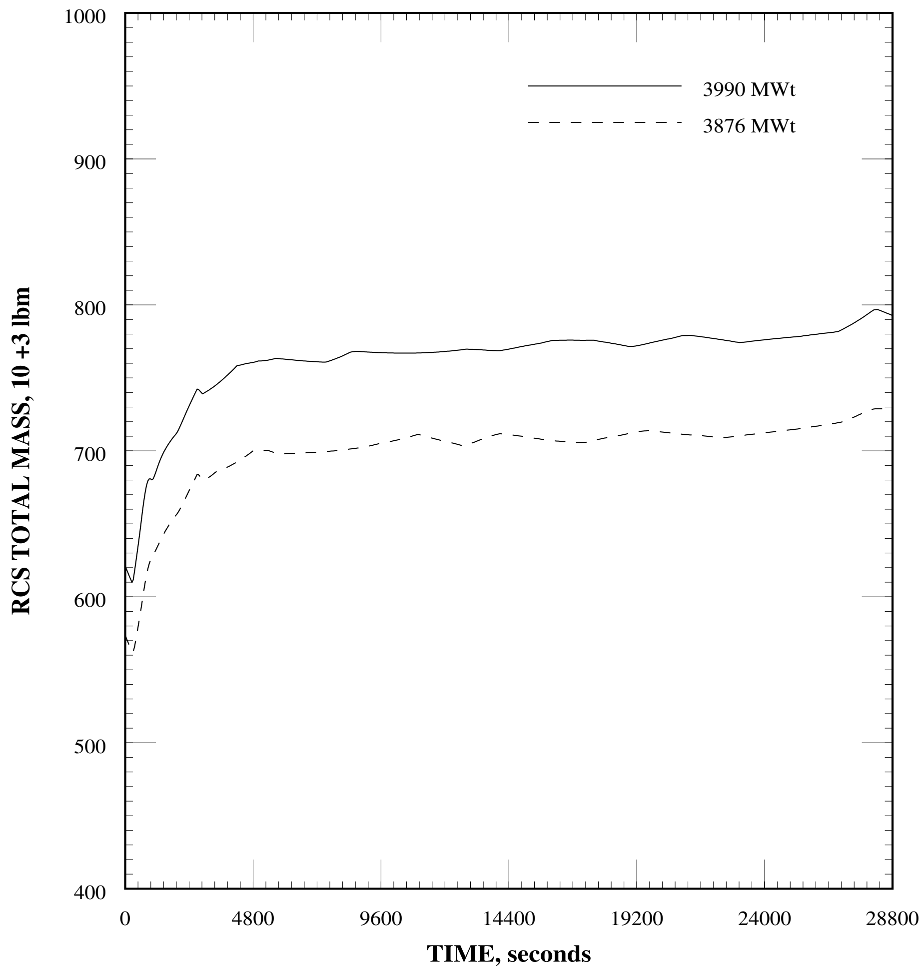


Figure 6.3-216
SGTRLOP Single Failure Event - SG Pressure vs. Time

(Sheet 1 of 2)

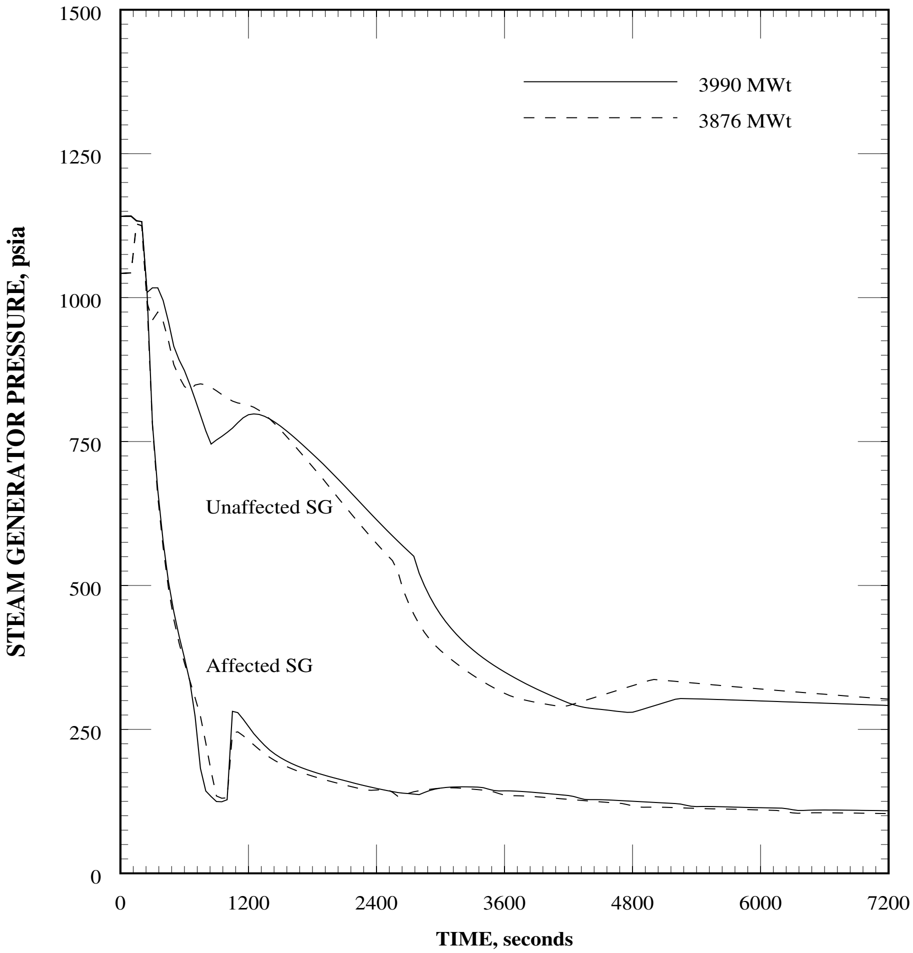


Figure 6.3-216
SGTRLOP Single Failure Event - SG Pressure vs. Time

(Sheet 2 of 2)

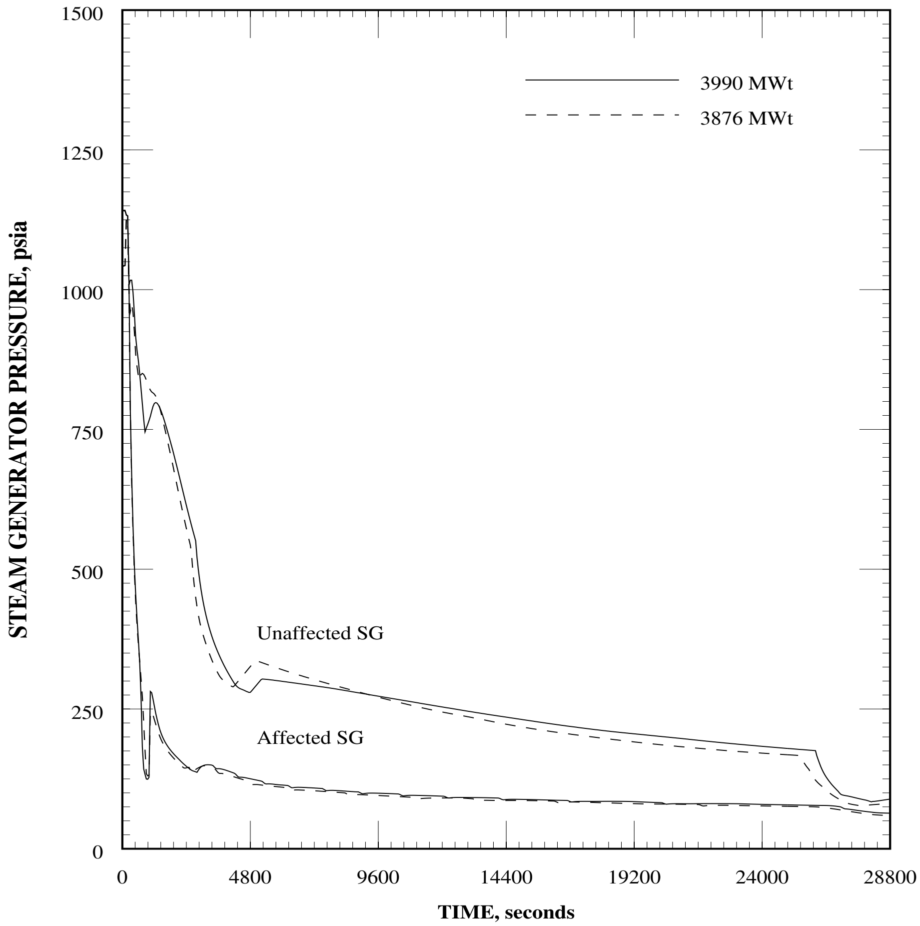


Figure 6.3-217
SGTRLOP Single Failure Event - AFW Integrated Flow vs. Time
(Sheet 1 of 2)

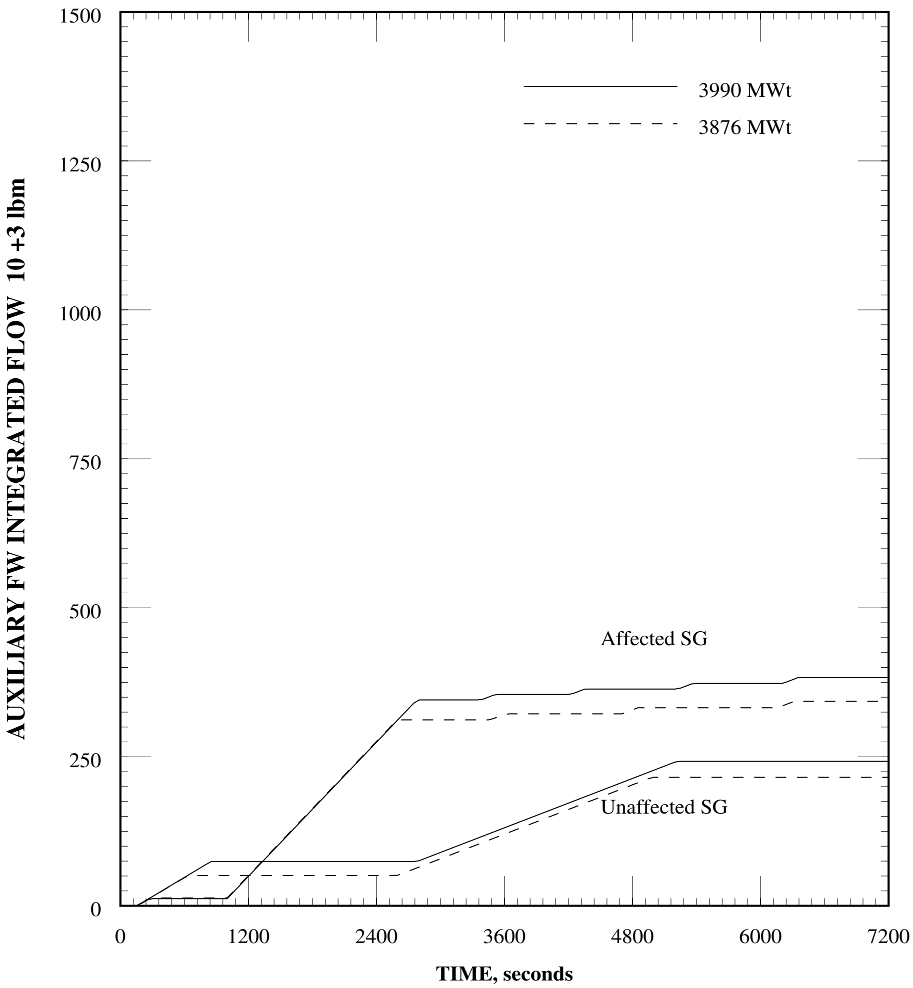


Figure 6.3-217
SGTRLOP Single Failure Event - AFW Integrated Flow vs. Time

(Sheet 2 of 2)

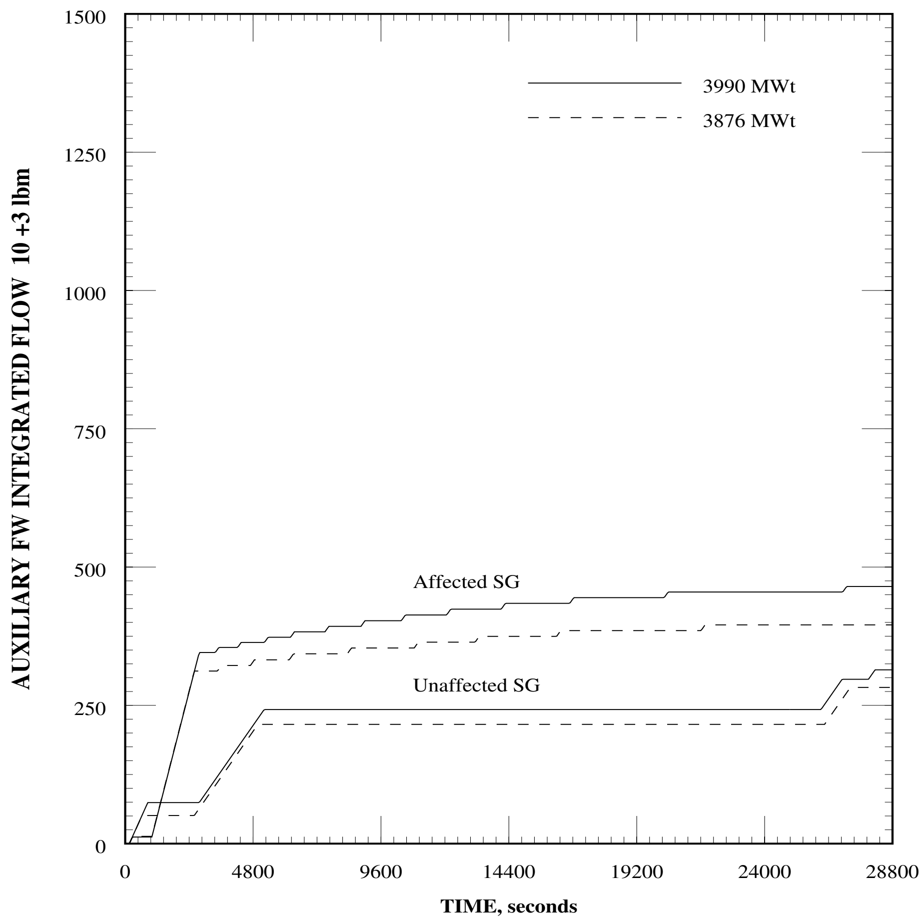


Figure 6.3-218
SGTRLOP Single Failure Event - Tube Leak Rate vs. Time
(Sheet 1 of 2)

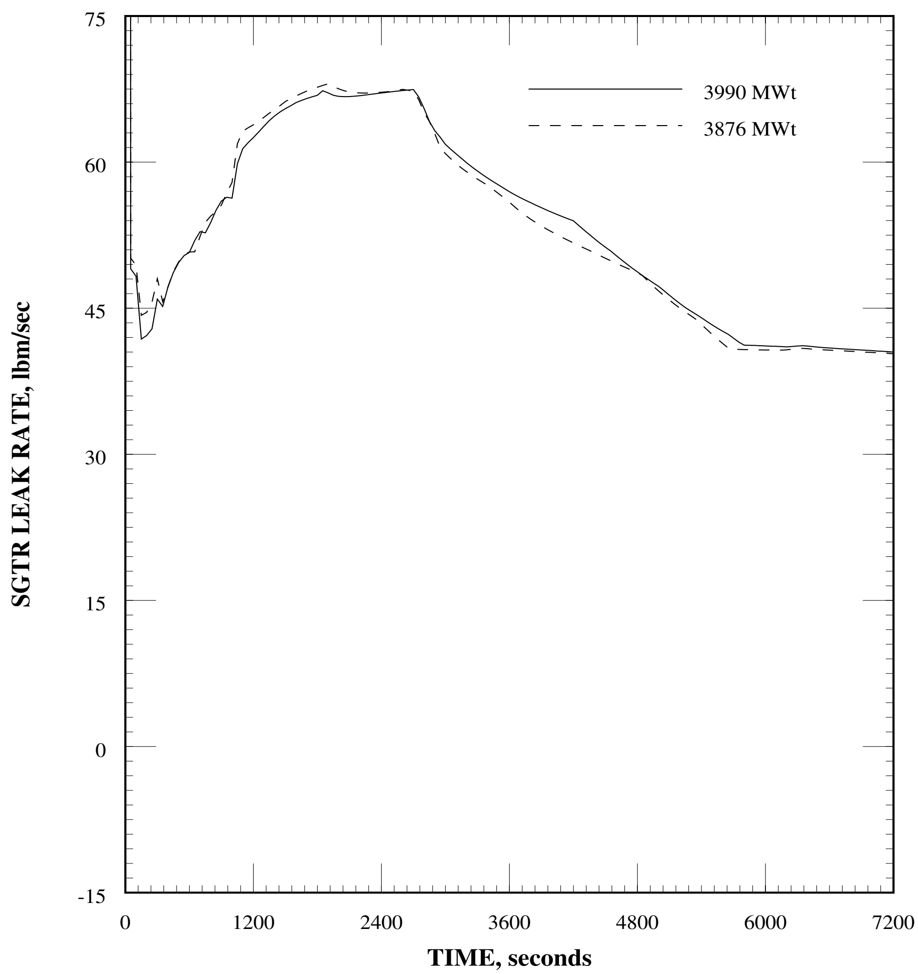


Figure 6.3-218
SGTRLOP Single Failure Event - Tube Leak Rate vs. Time

(Sheet 2 of 2)

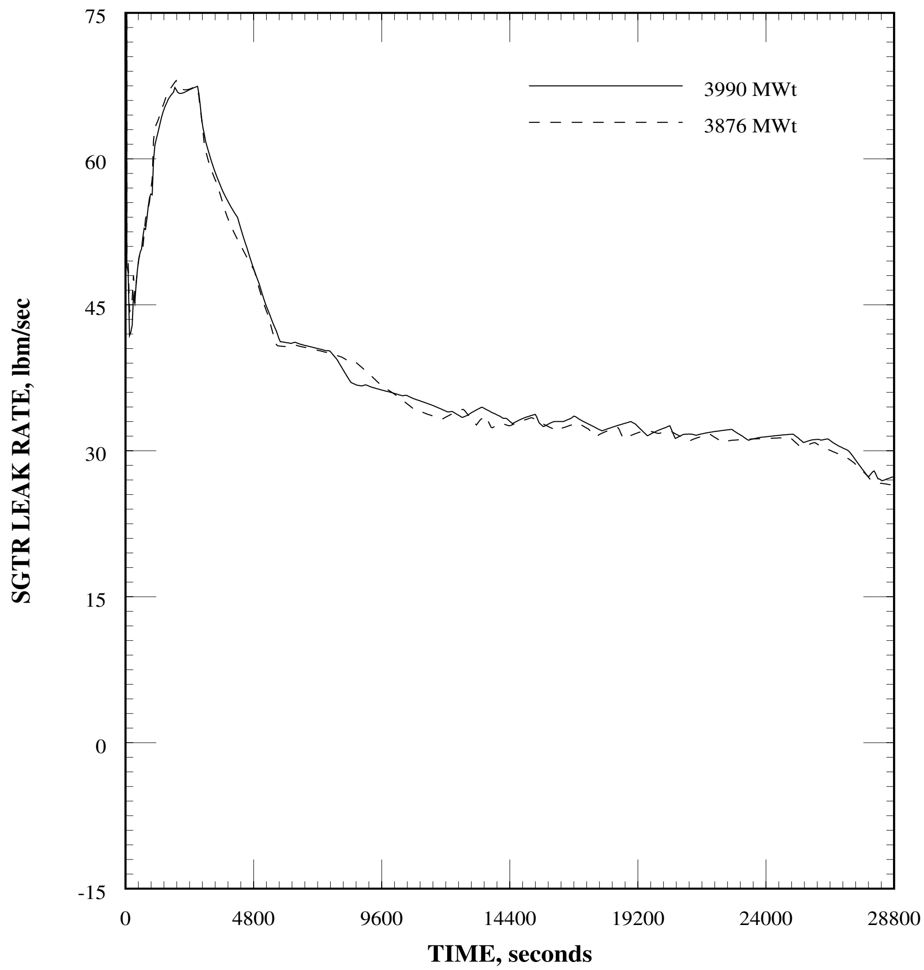


Figure 6.3-219
SGTRLOP Single Failure Event - Integrated Tube Leak vs. Time
(Sheet 1 of 2)

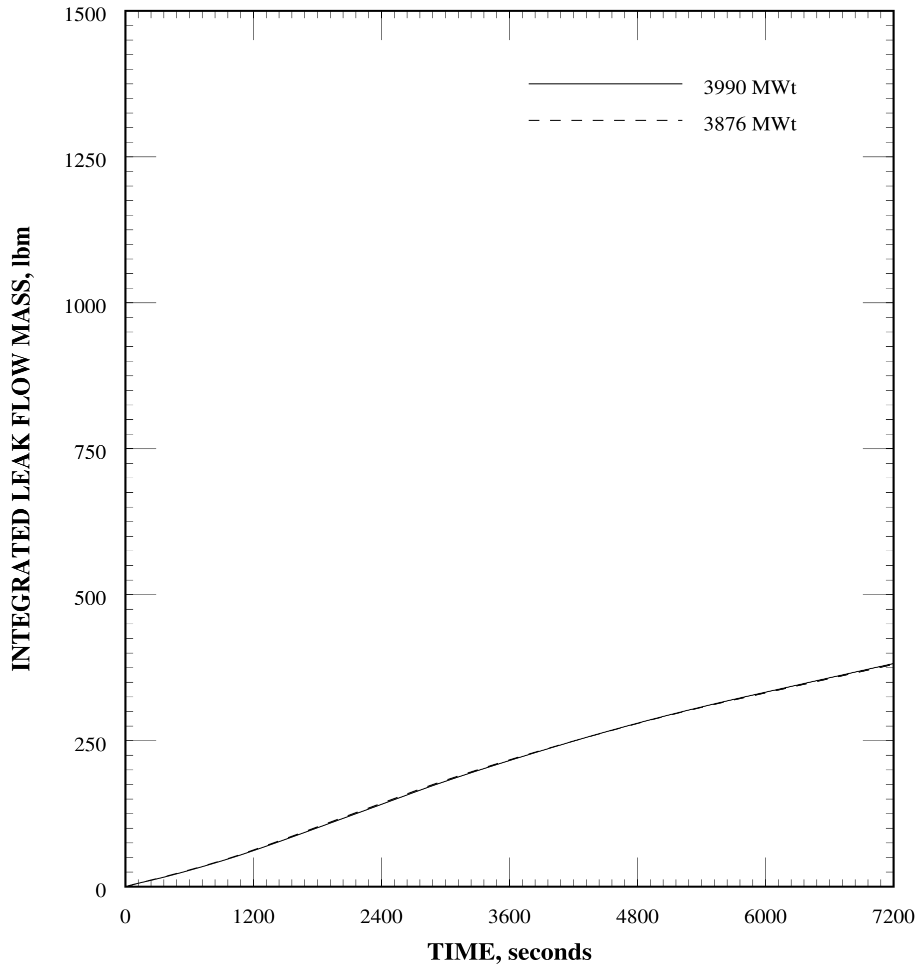


Figure 6.3-219
SGTRLOP Single Failure Event - Integrated Tube Leak vs. Time

(Sheet 2 of 2)

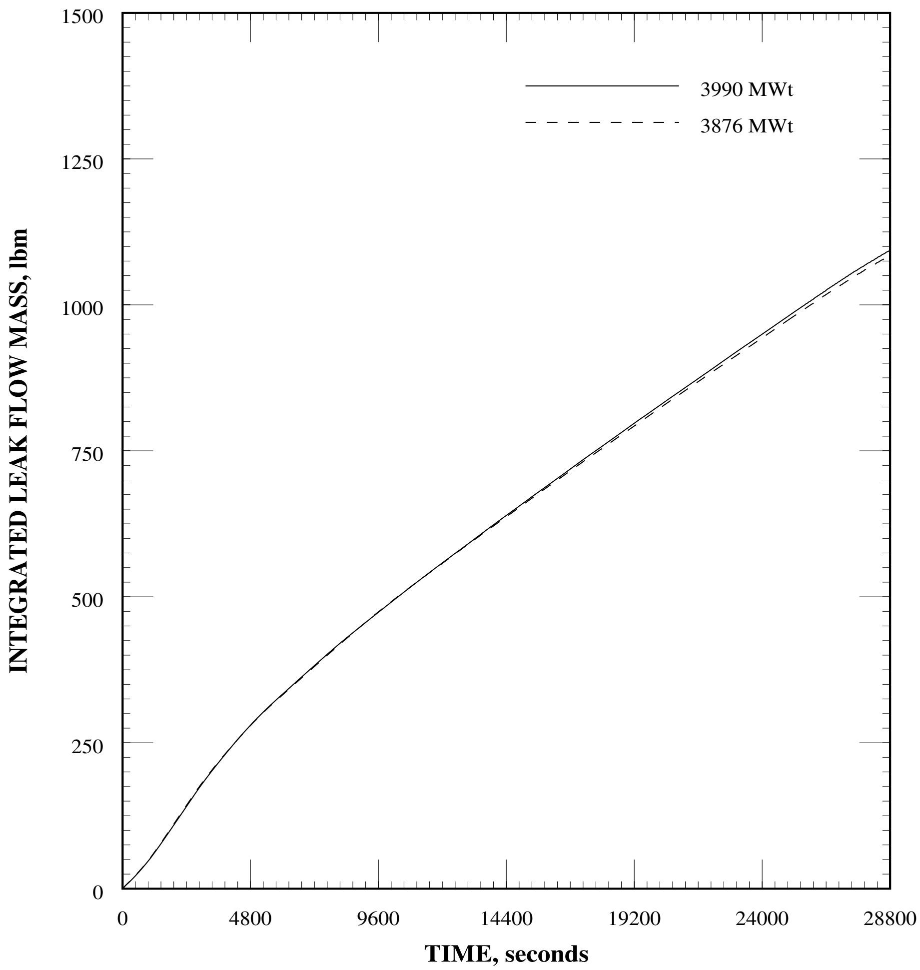


Figure 6.3-220
SGTRLOP Single Failure Event - Leak Flashing Fraction vs. Time
(Sheet 1 of 2)

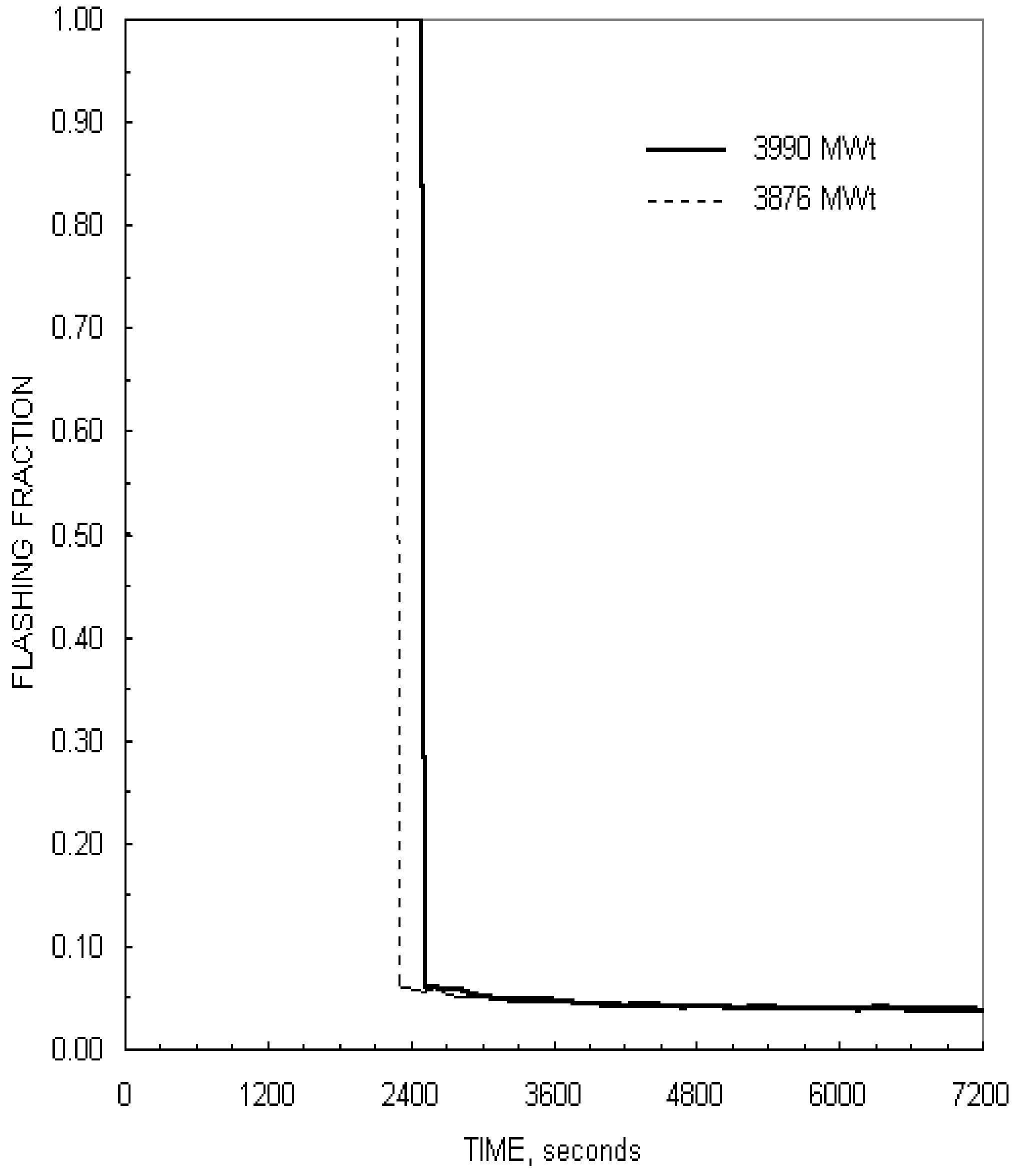


Figure 6.3-220
SGTRLOP Single Failure Event - Leak Flashing Fraction vs. Time

(Sheet 2 of 2)

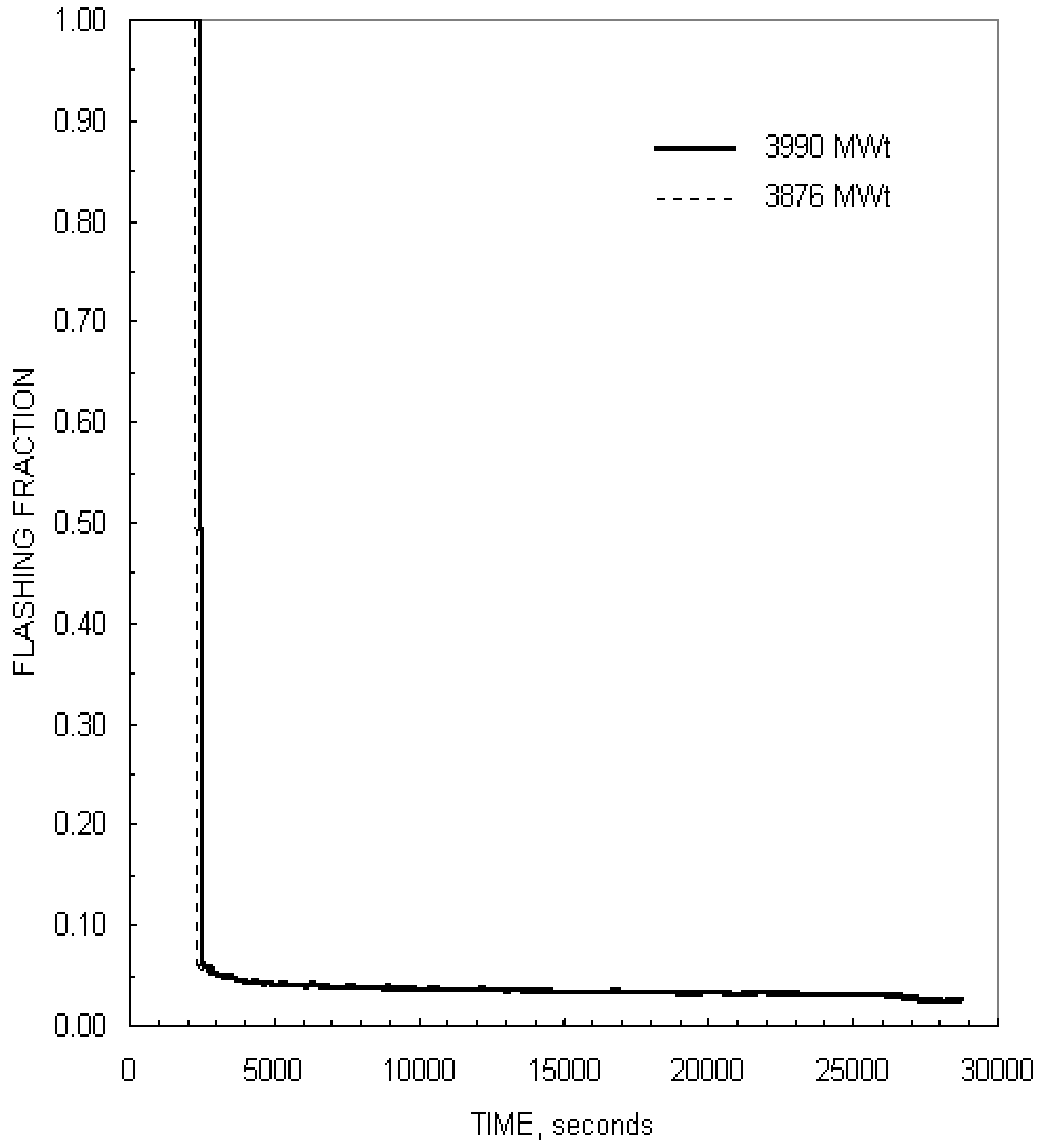


Figure 6.3-221
SGTRLOP Single Failure Event - SG Liquid Inventory vs. Time
(Sheet 1 of 2)

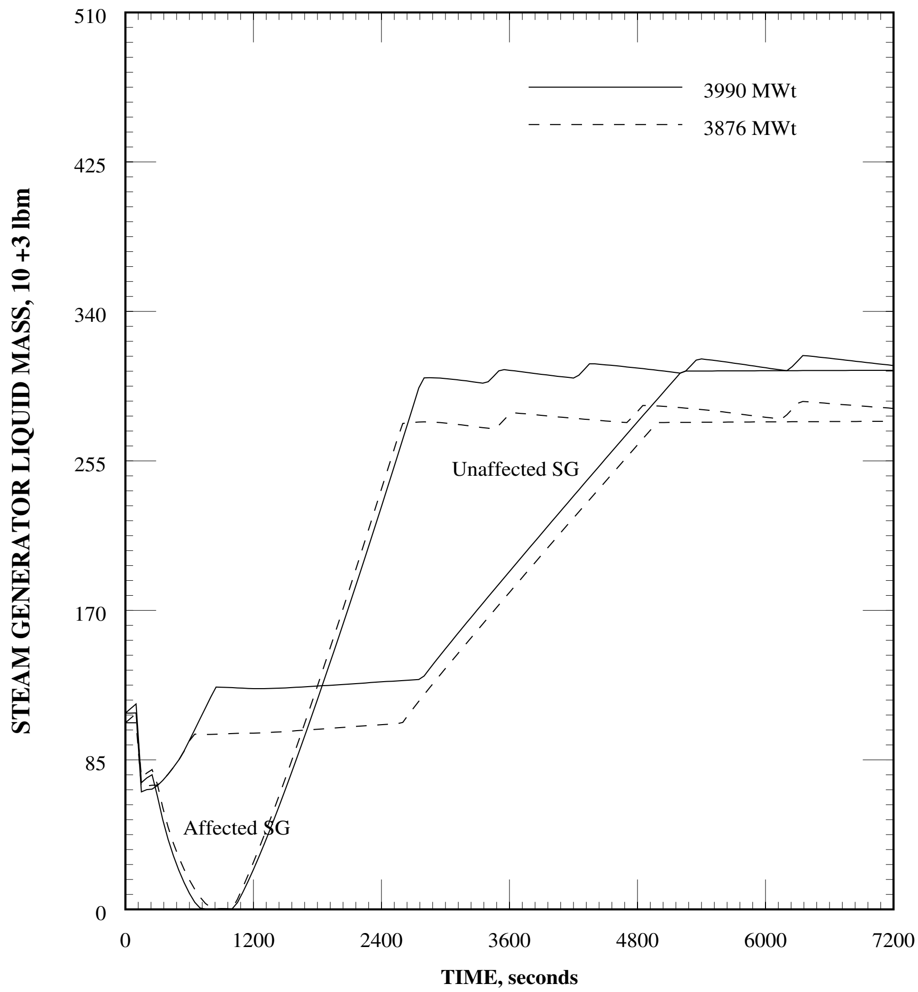


Figure 6.3-221
SGTRLOP Single Failure Event - SG Liquid Inventory vs. Time

(Sheet 2 of 2)

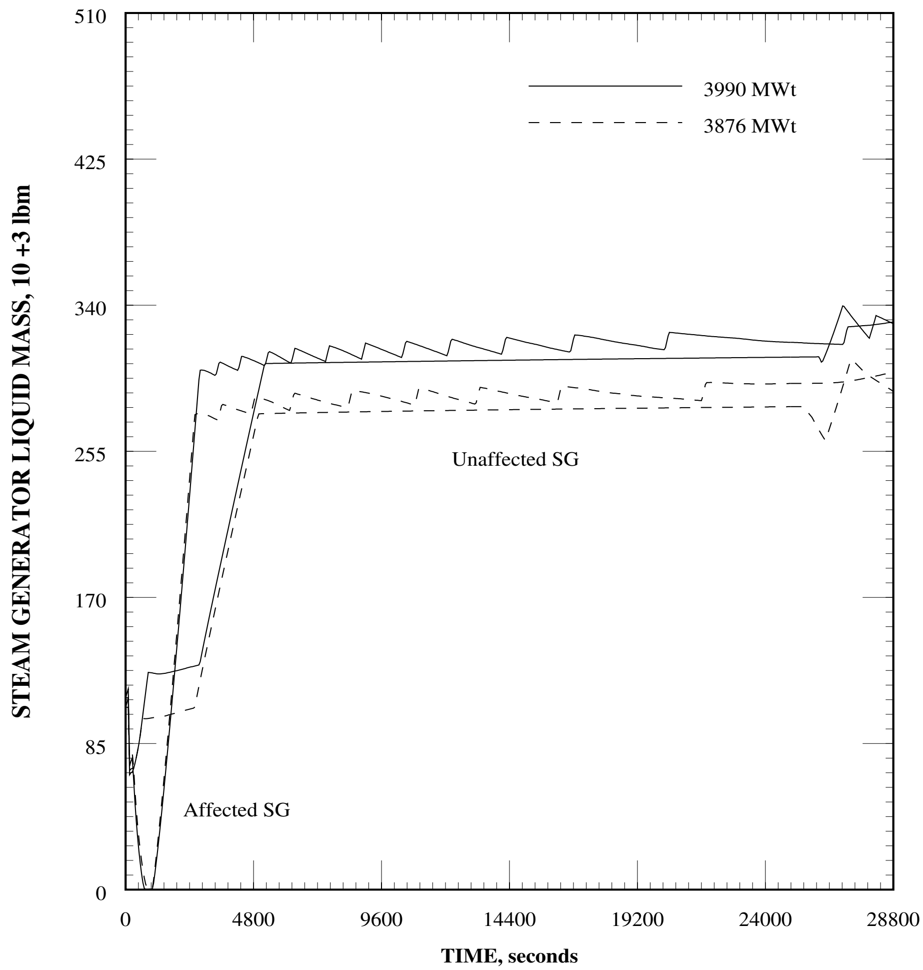


Figure 6.3-222
SGTRLOP Single Failure Event - Integrated ADV Flow vs. Time
(Sheet 1 of 2)

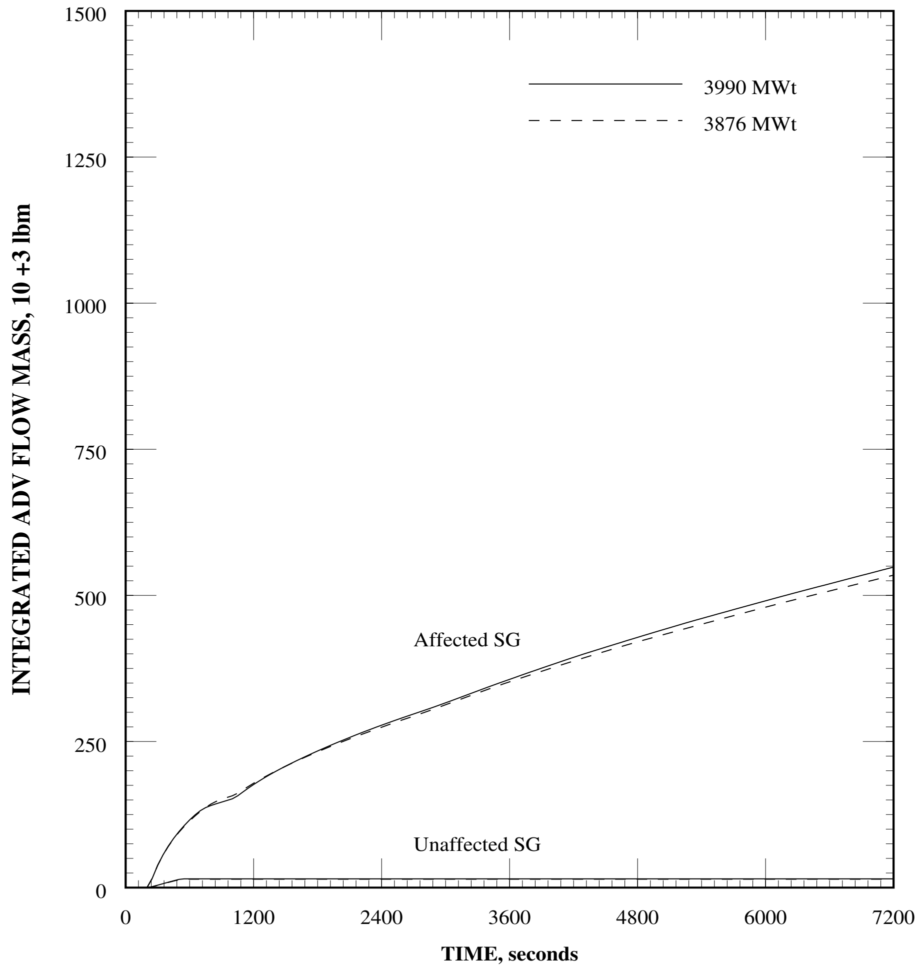


Figure 6.3-222
SGTRLOP Single Failure Event - Integrated ADV Flow vs. Time

(Sheet 2 of 2)

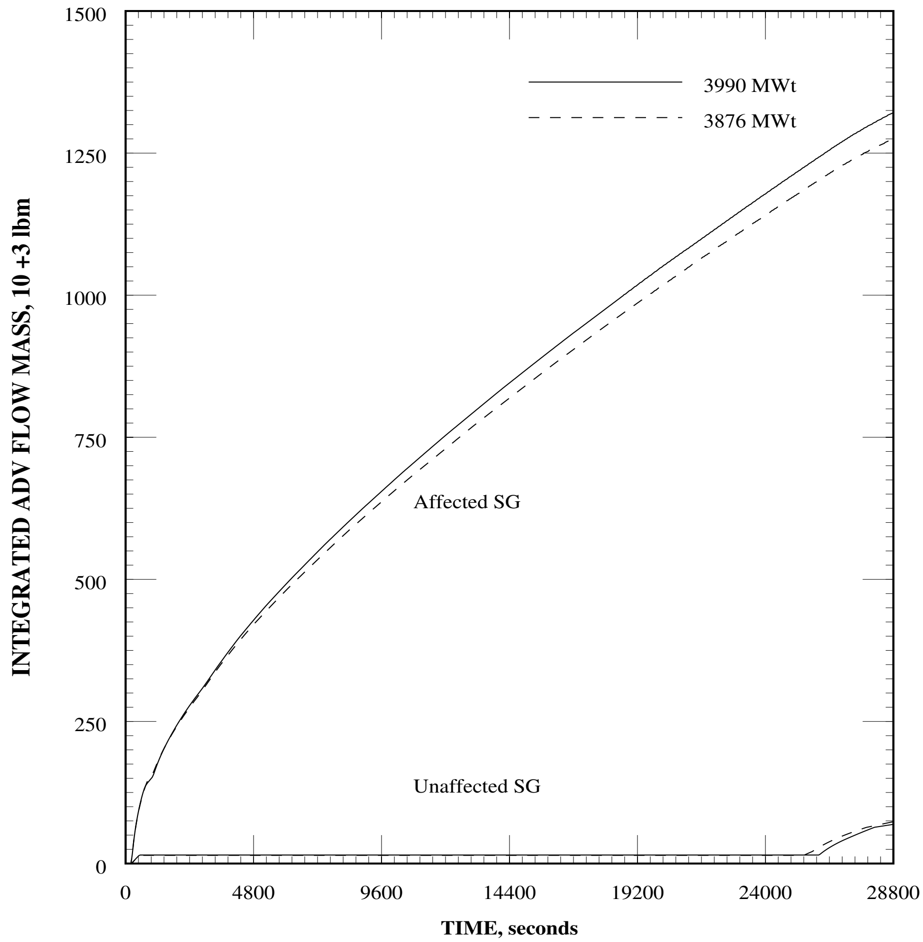


Figure 6.3-223
SGTRLOP Single Failure Event - Subcooled Margin vs. Time
(Sheet 1 of 2)

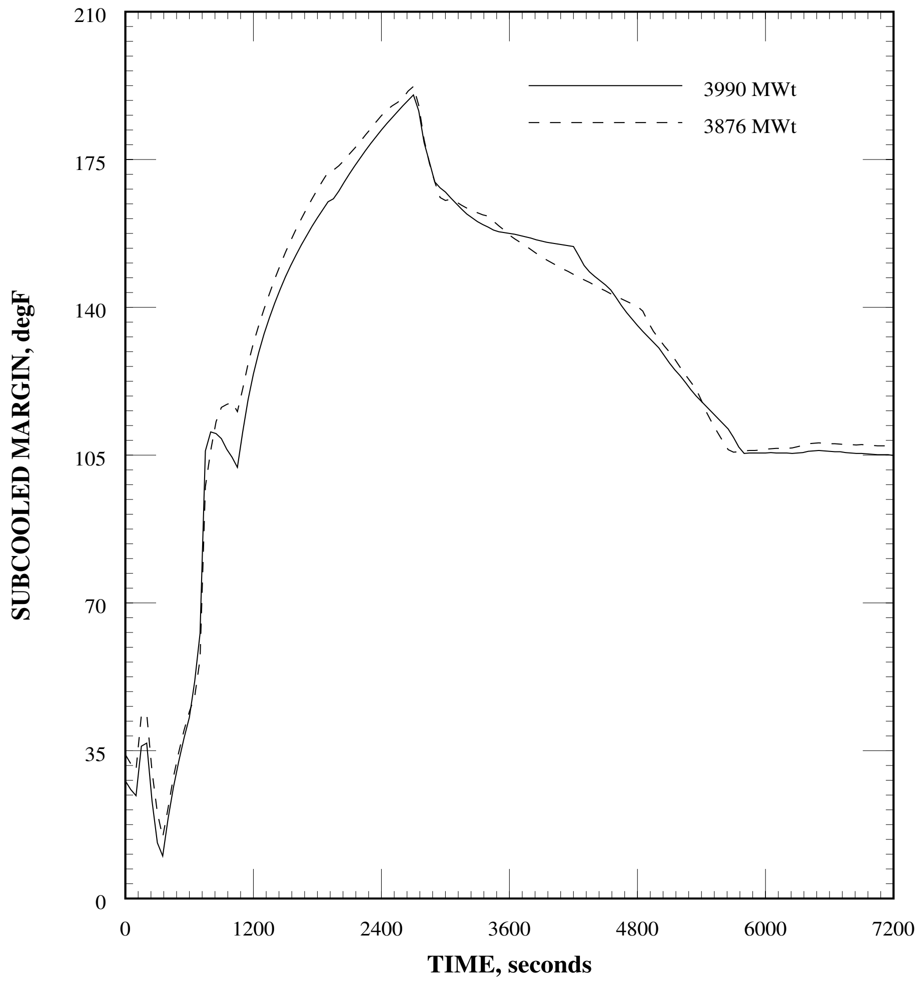


Figure 6.3-223
SGTRLOP Single Failure Event - Subcooled Margin vs. Time

(Sheet 2 of 2)

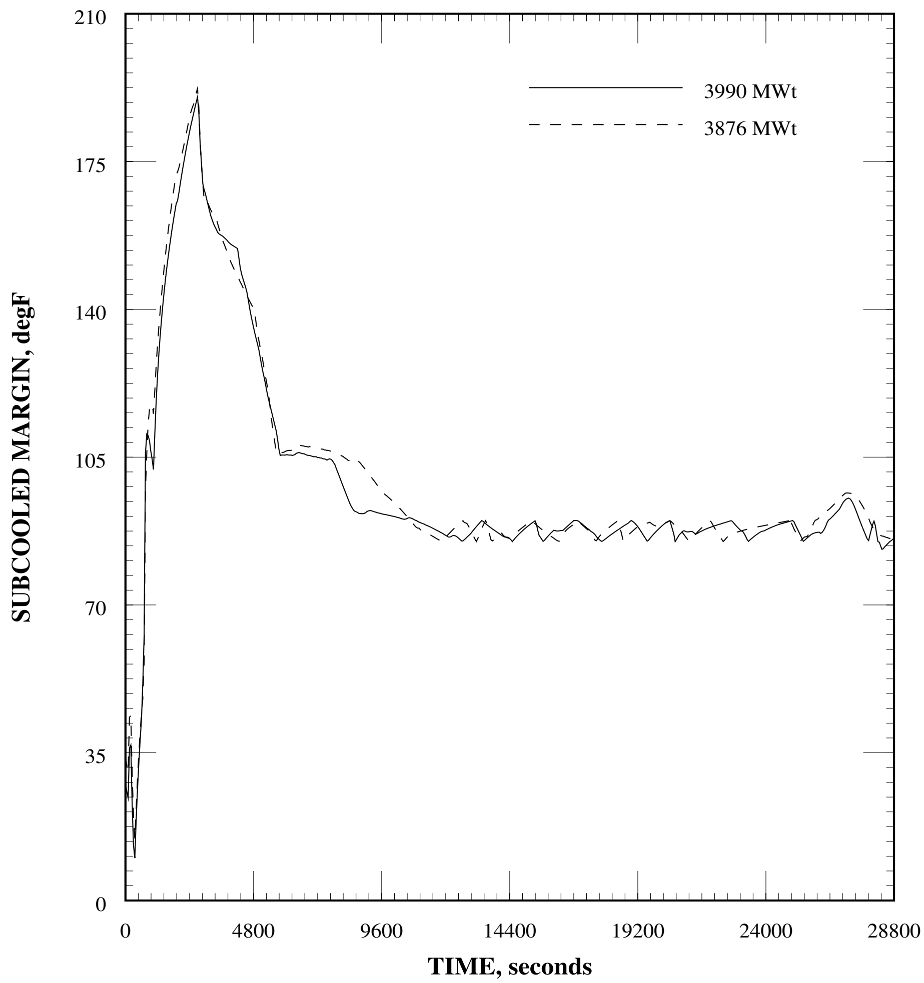


Figure 6.3-224
SGTRLOP Event - Core Power vs. Time
(Sheet 1 of 3)

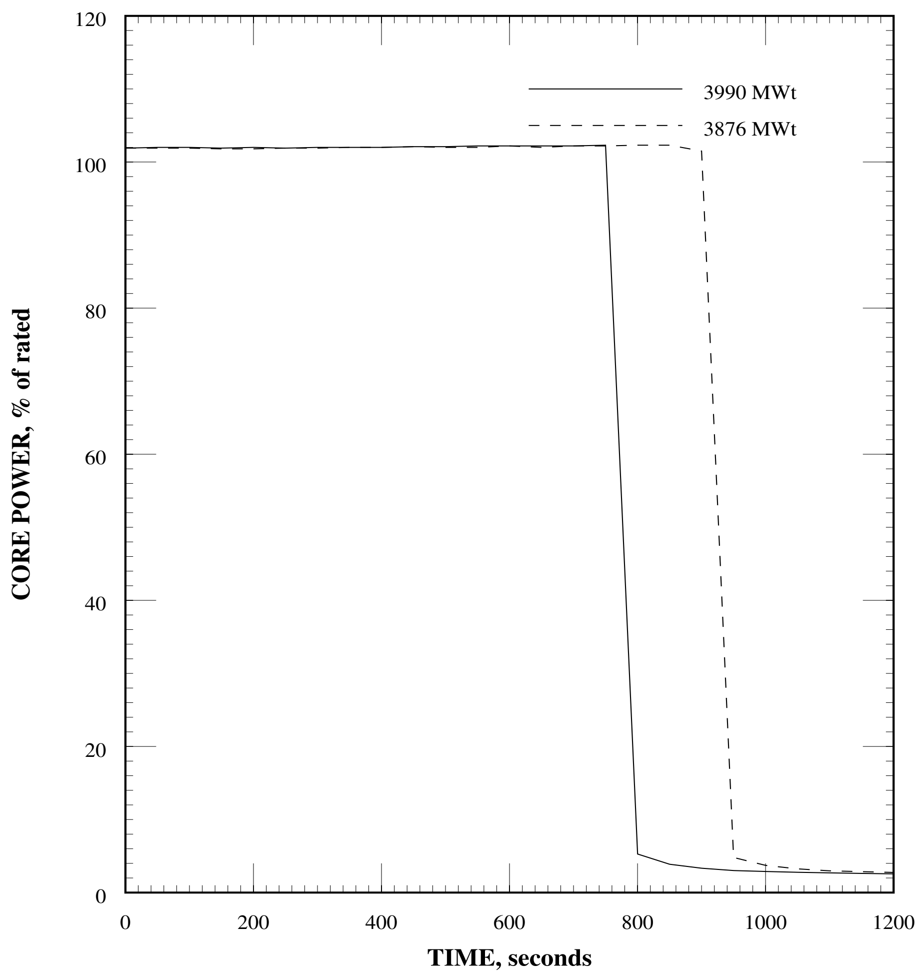


Figure 6.3-224
SGTRLOP Event - Core Power vs. Time

(Sheet 2 of 3)

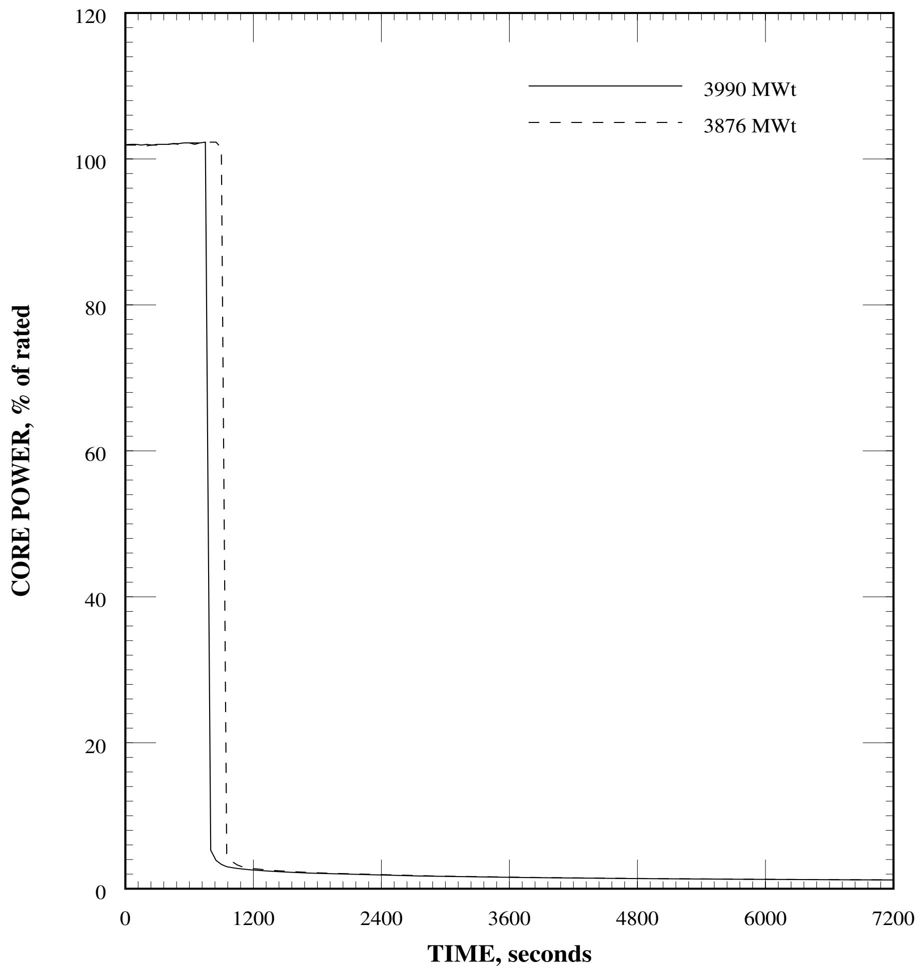


Figure 6.3-224
SGTRLOP Event - Core Power vs. Time

(Sheet 3 of 3)

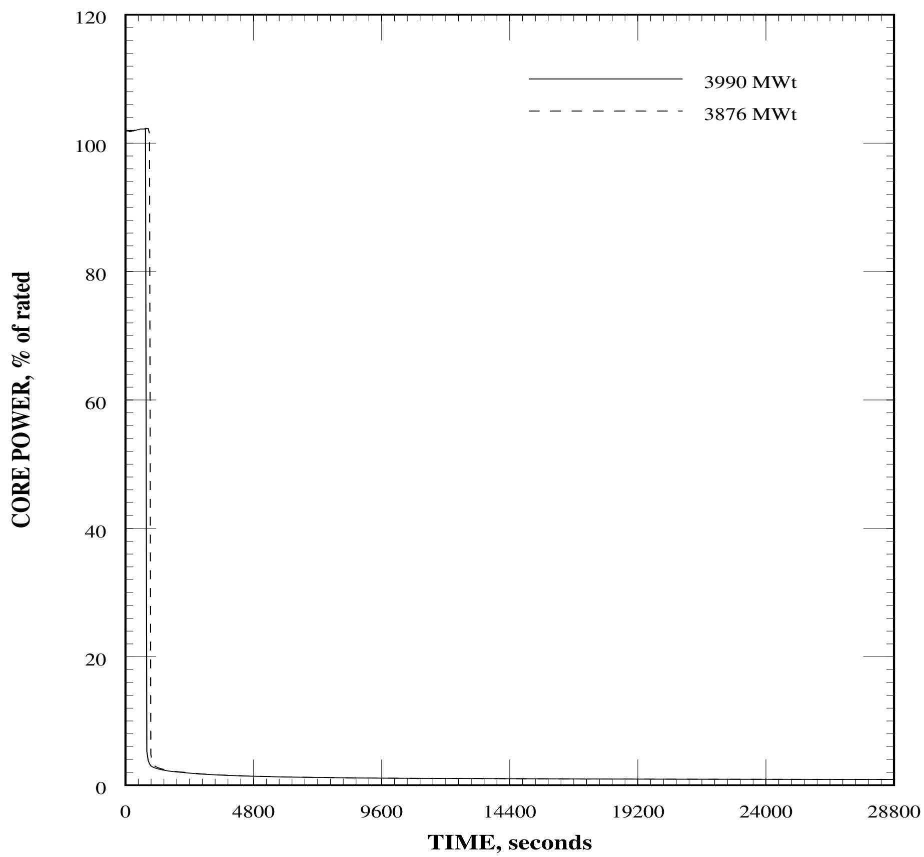


Figure 6.3-225
SGTRLOP Event - RCS Pressure vs. Time

(Sheet 1 of 3)

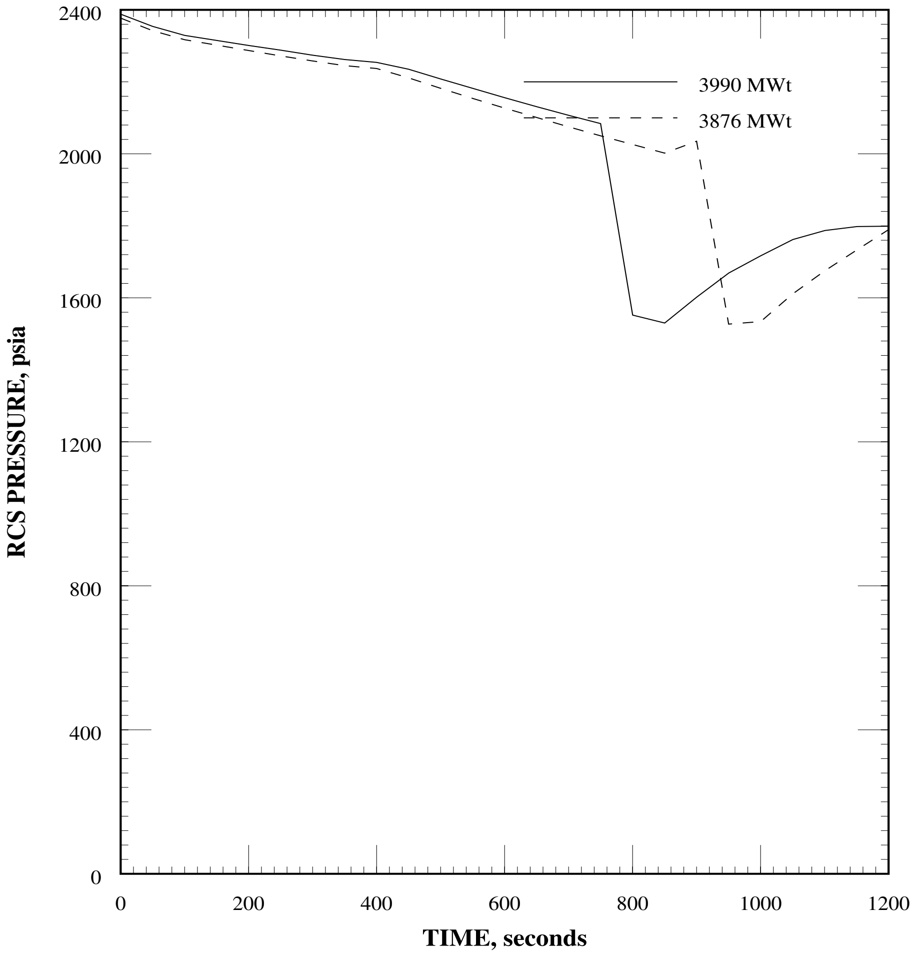


Figure 6.3-225
SGTRLOP Event - RCS Pressure vs. Time

(Sheet 2 of 3)

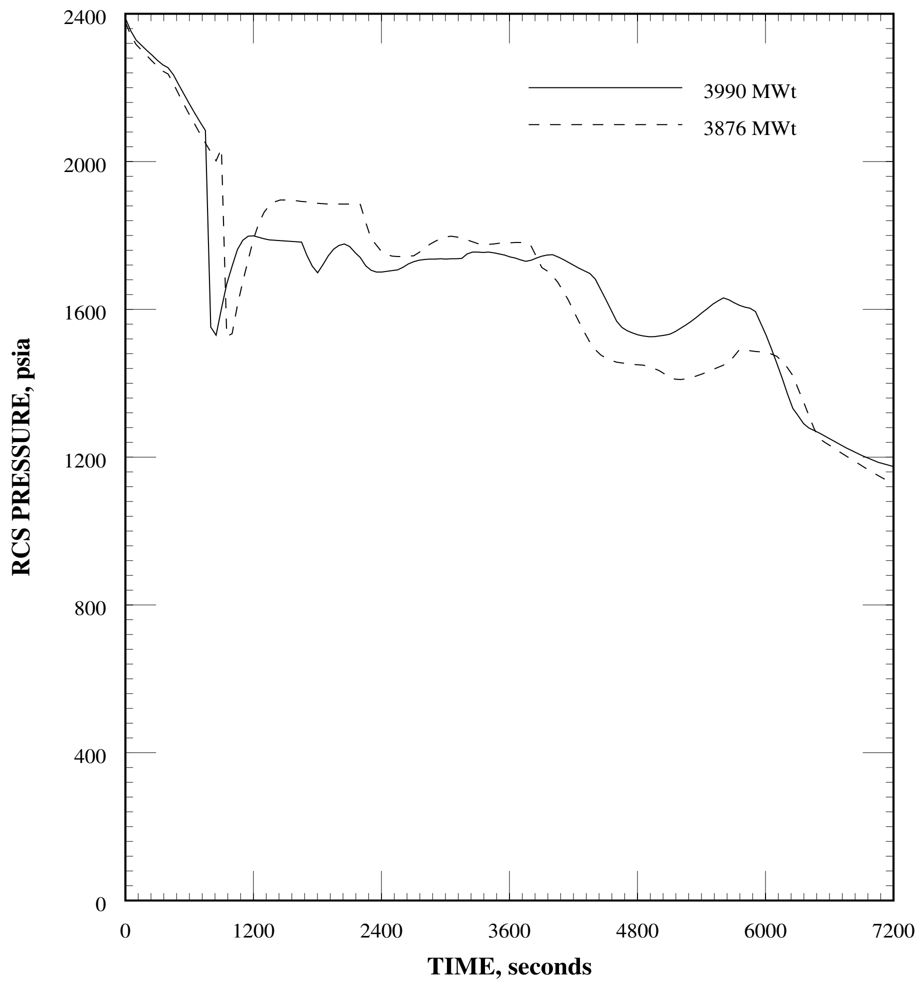


Figure 6.3-225
SGTRLOP Event - RCS Pressure vs. Time

(Sheet 3 of 3)

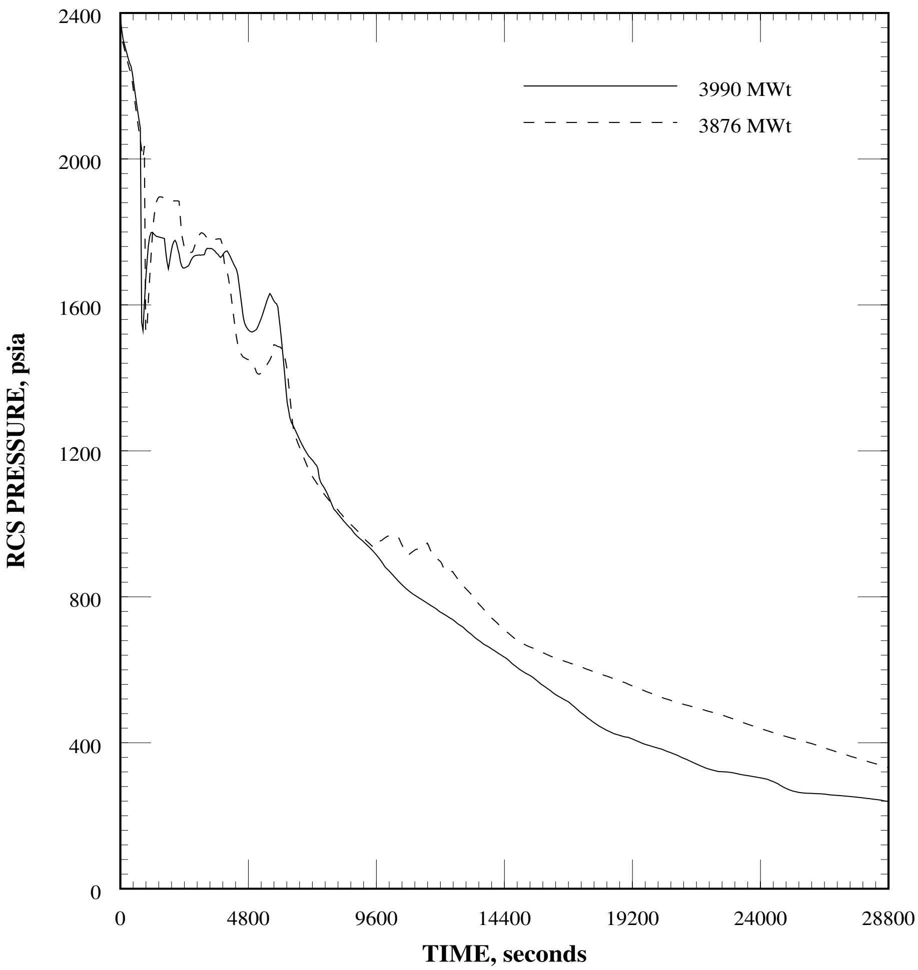


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 1 of 6)

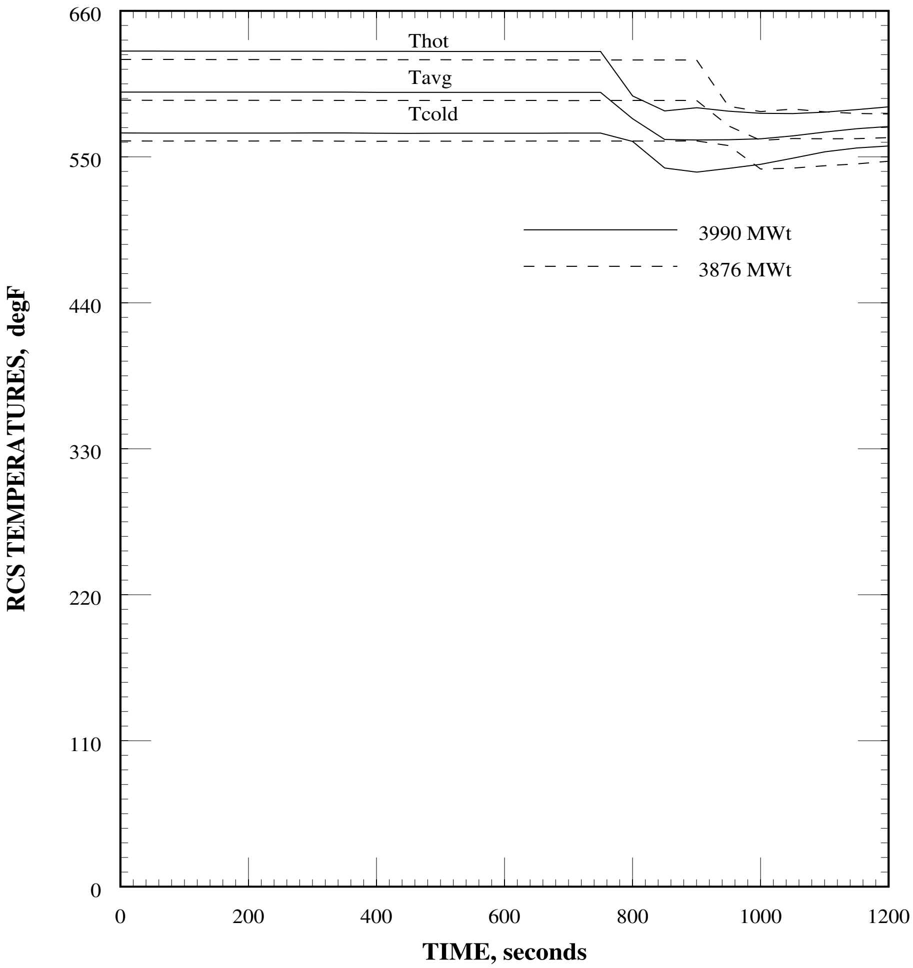


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 2 of 6)

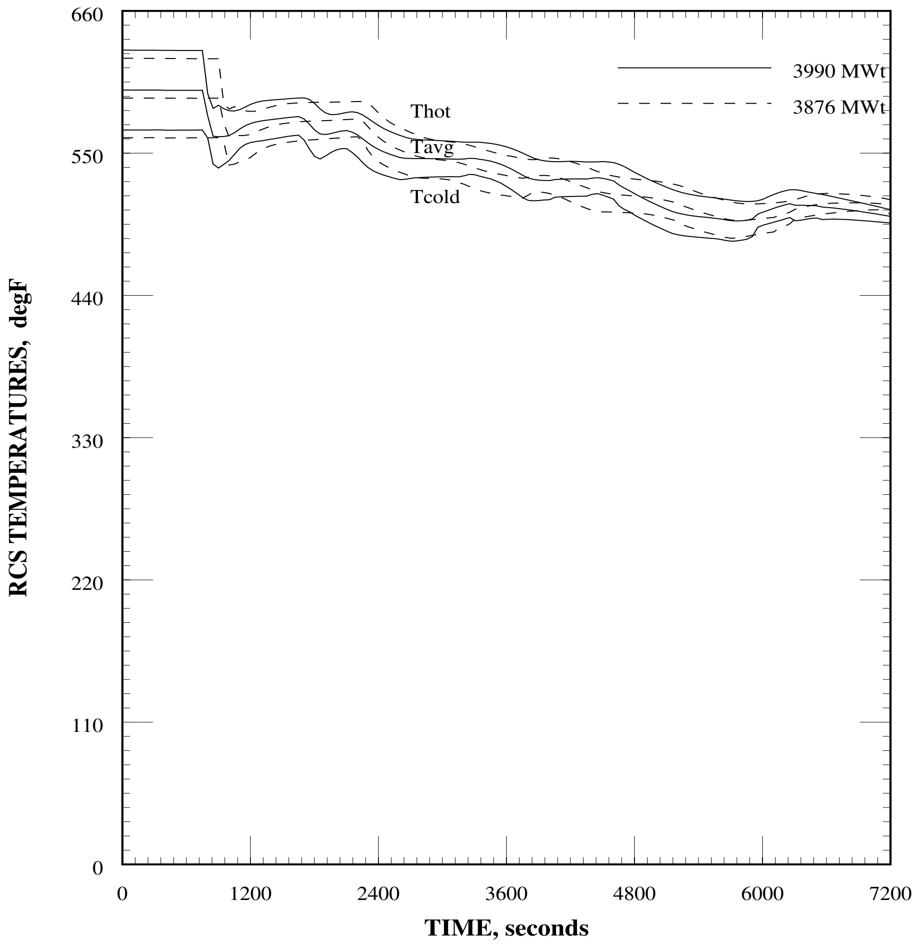


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 3 of 6)

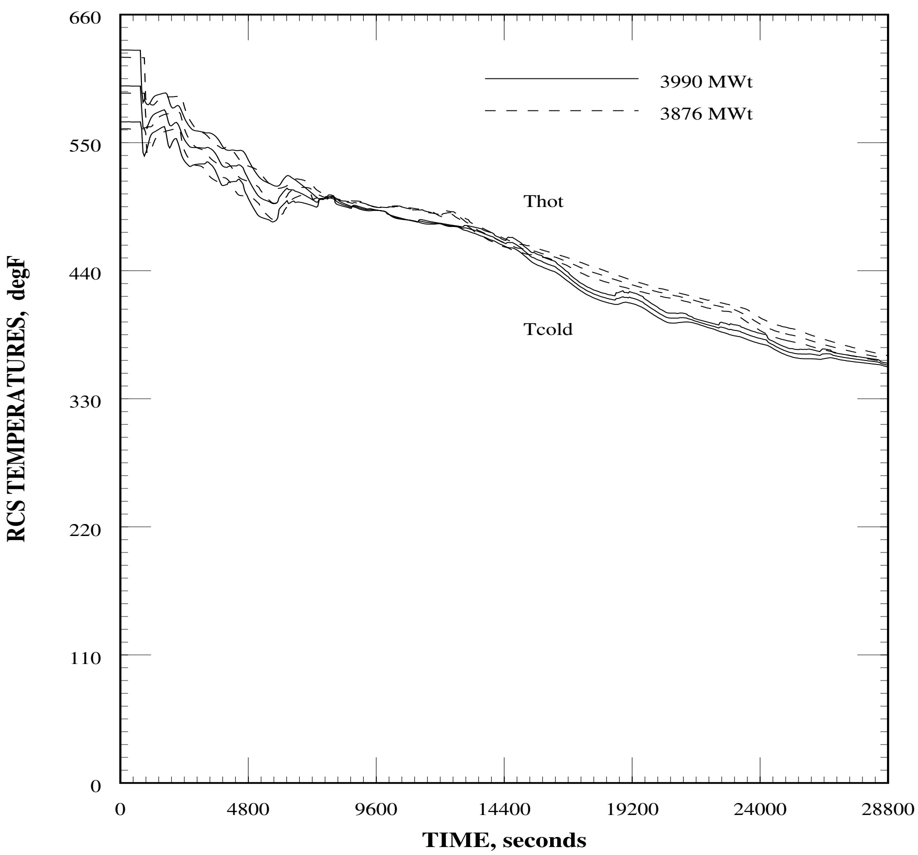


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 4 of 6)

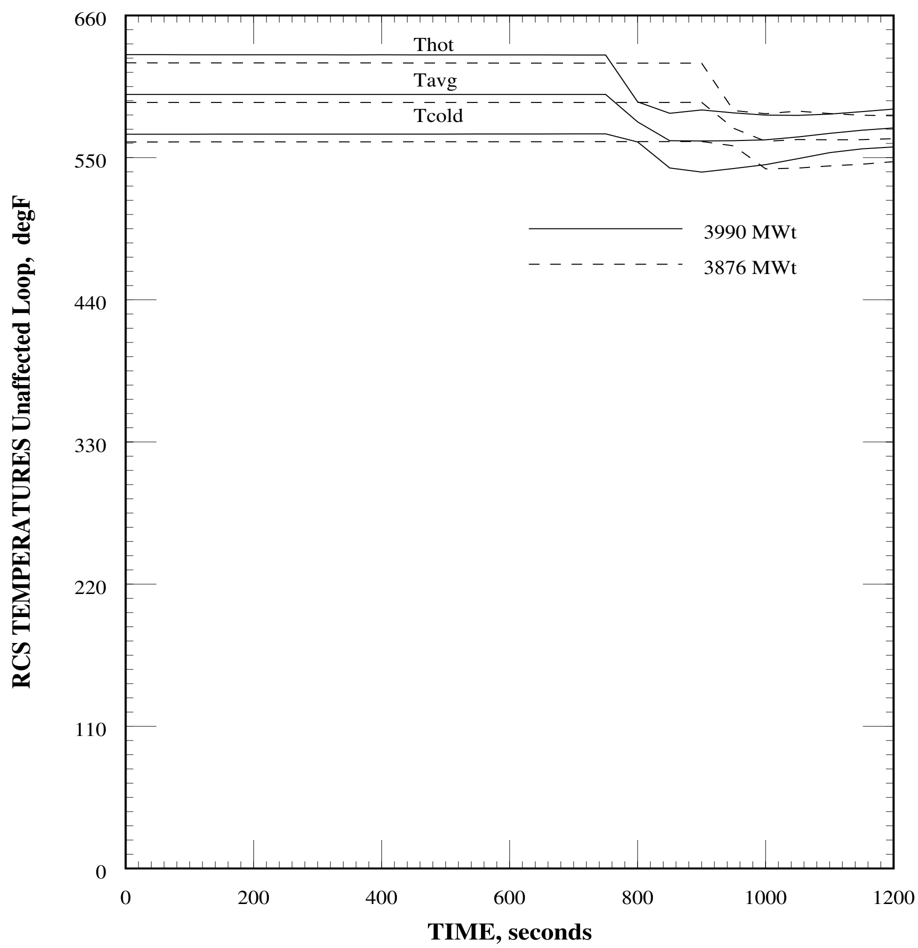


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 5 of 6)

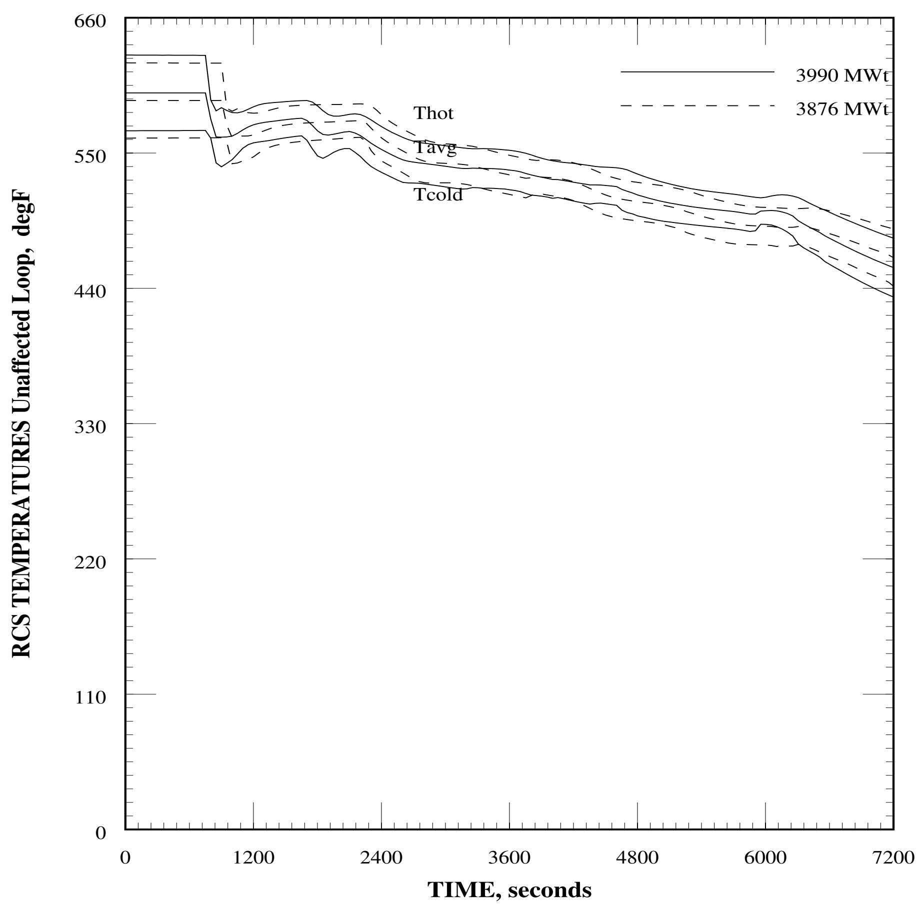


Figure 6.3-226
SGTRLOP Event - Unaffected Loop RCS Temperatures vs. Time

(Sheet 6 of 6)

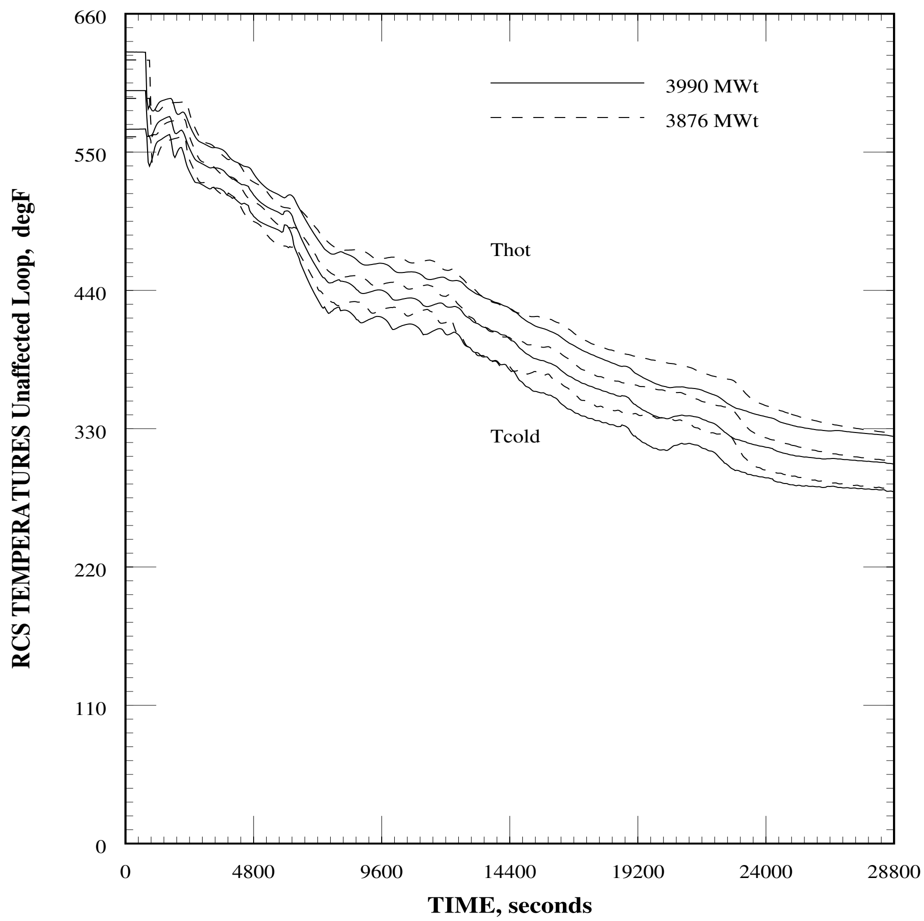


Figure 6.3-227
SGTRLOP Event - Pressurizer Liquid Volume vs. Time
(Sheet 1 of 3)

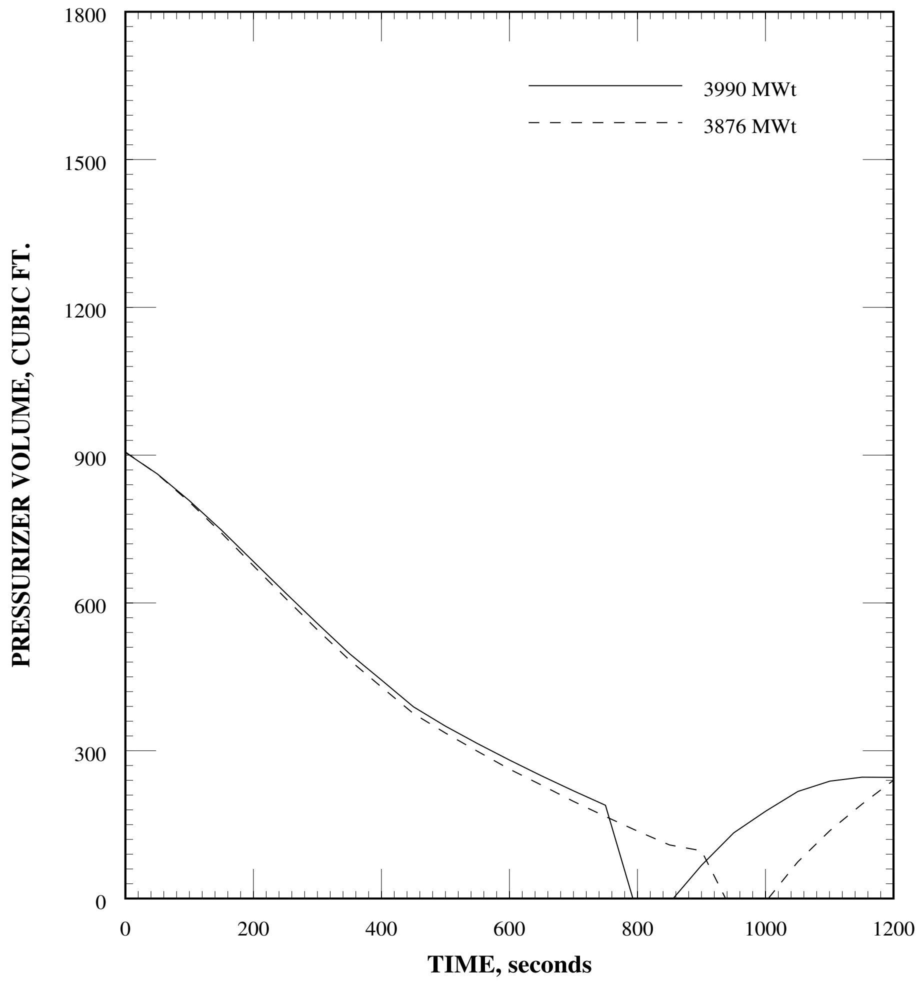


Figure 6.3-227
SGTRLOP Event - Pressurizer Liquid Volume vs. Time

(Sheet 2 of 3)

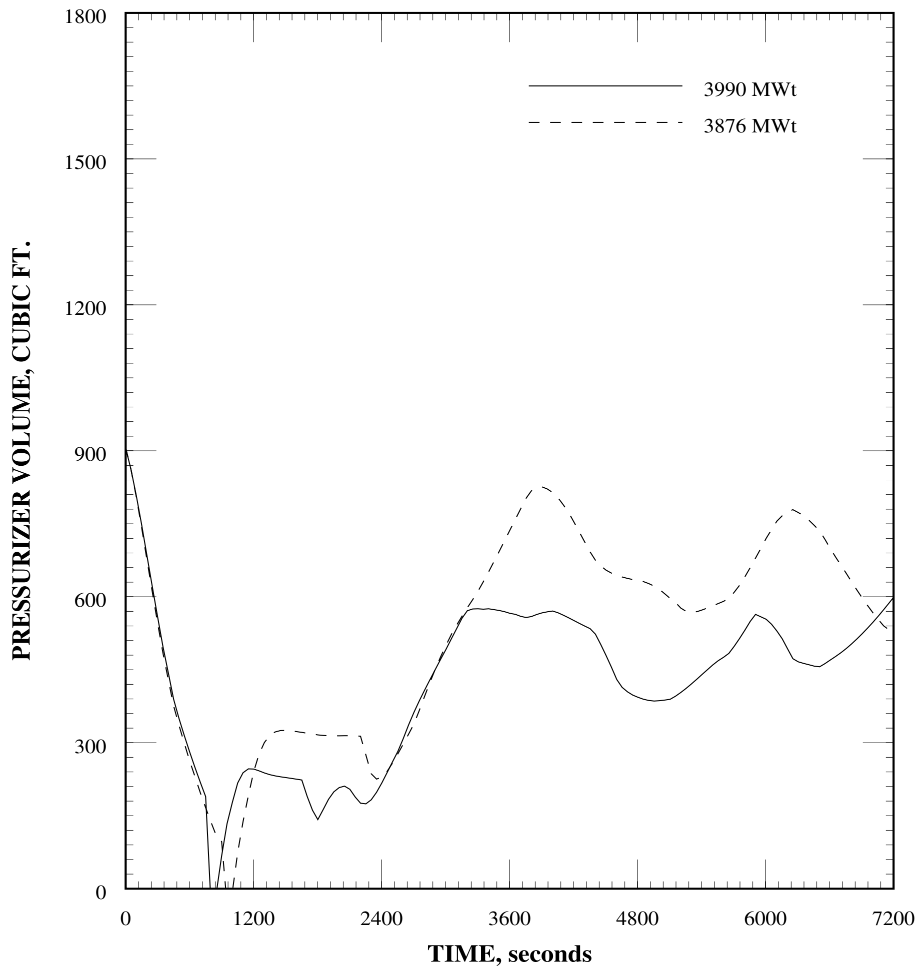


Figure 6.3-227
SGTRLOP Event - Pressurizer Liquid Volume vs. Time

(Sheet 3 of 3)

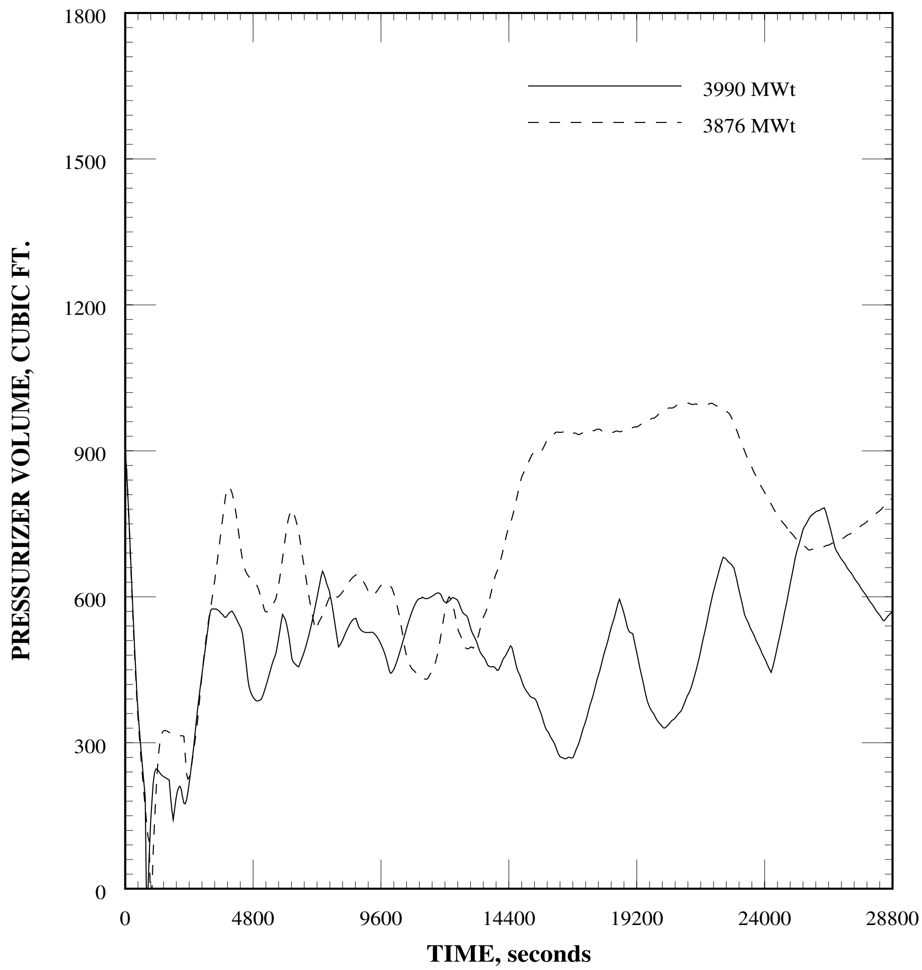


Figure 6.3-228
SGTRLOP Event - Upper Head Void Fraction vs. Time

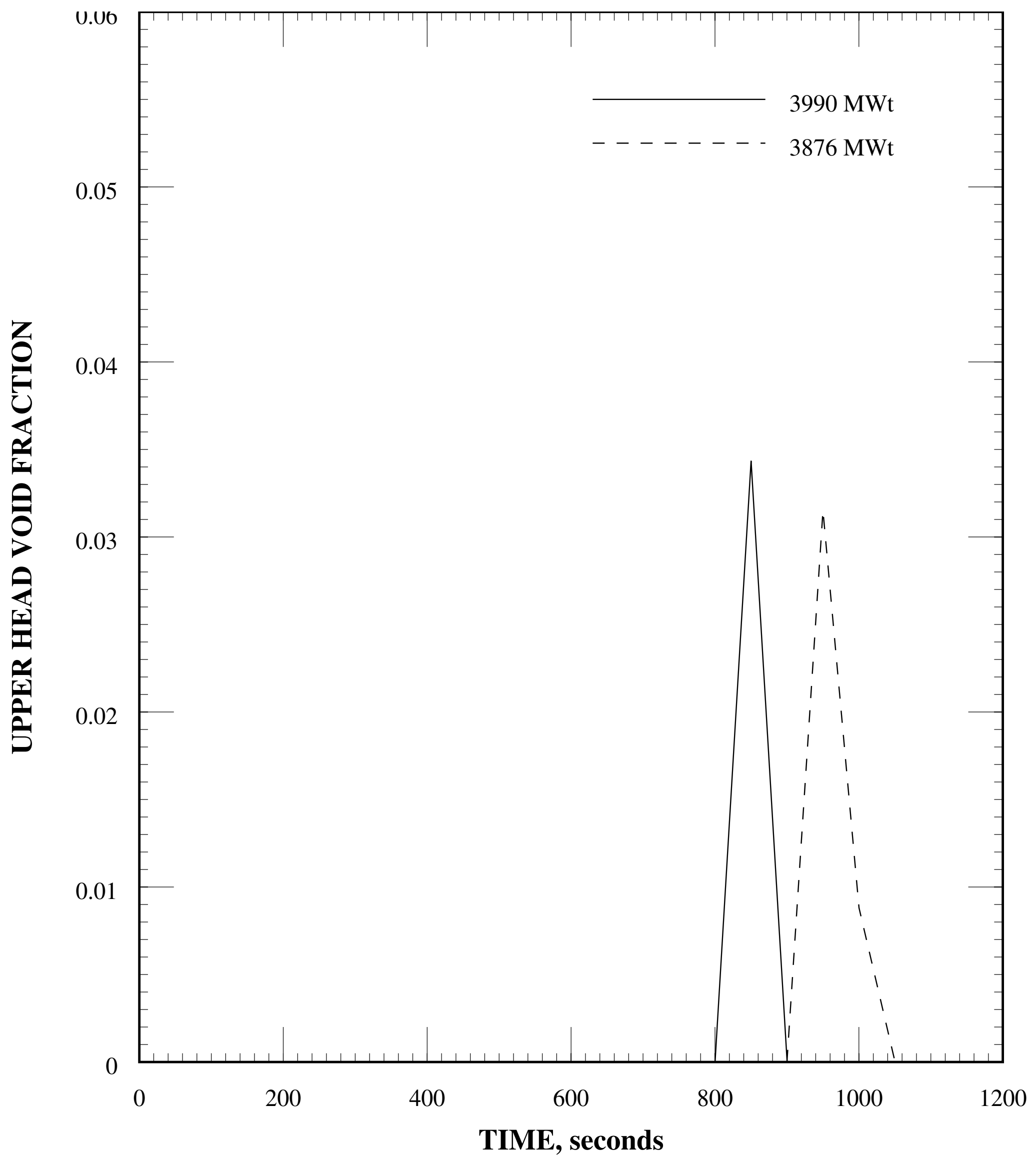


Figure 6.3-229
SGTRLOP Event - RCS Liquid Mass vs. Time

(Sheet 1 of 2)

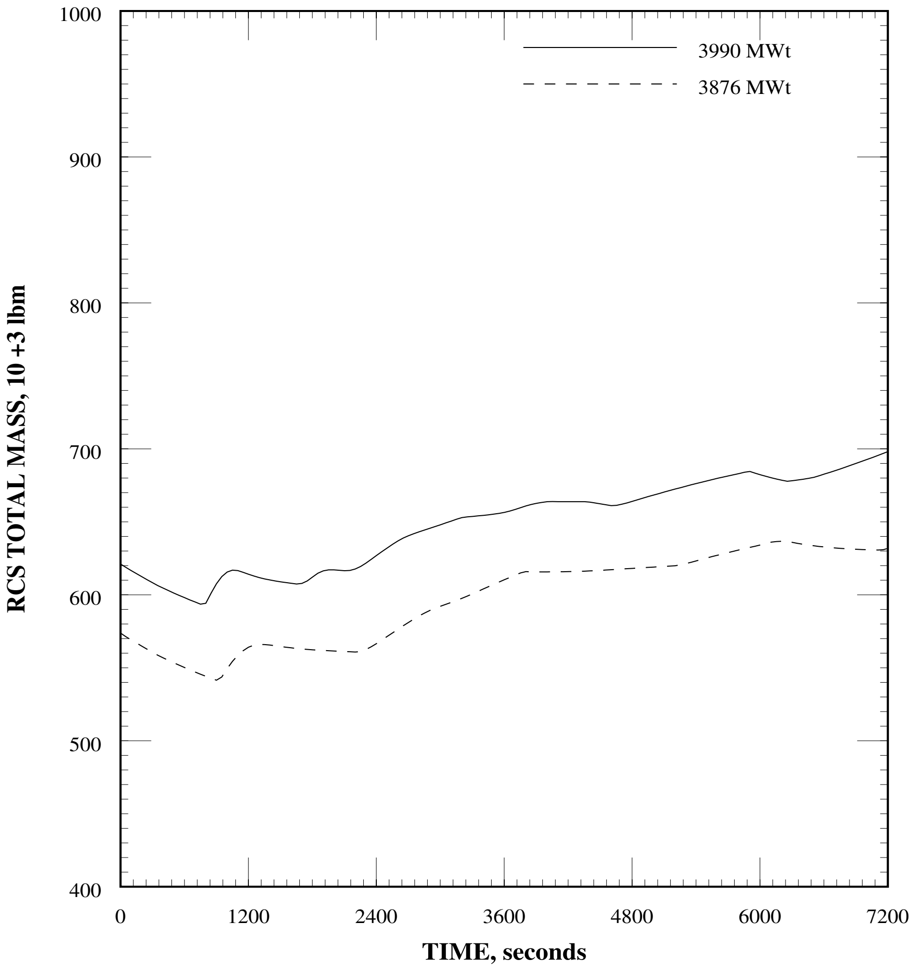


Figure 6.3-229
SGTRLOP Event - RCS Liquid Mass vs. Time

(Sheet 2 of 2)

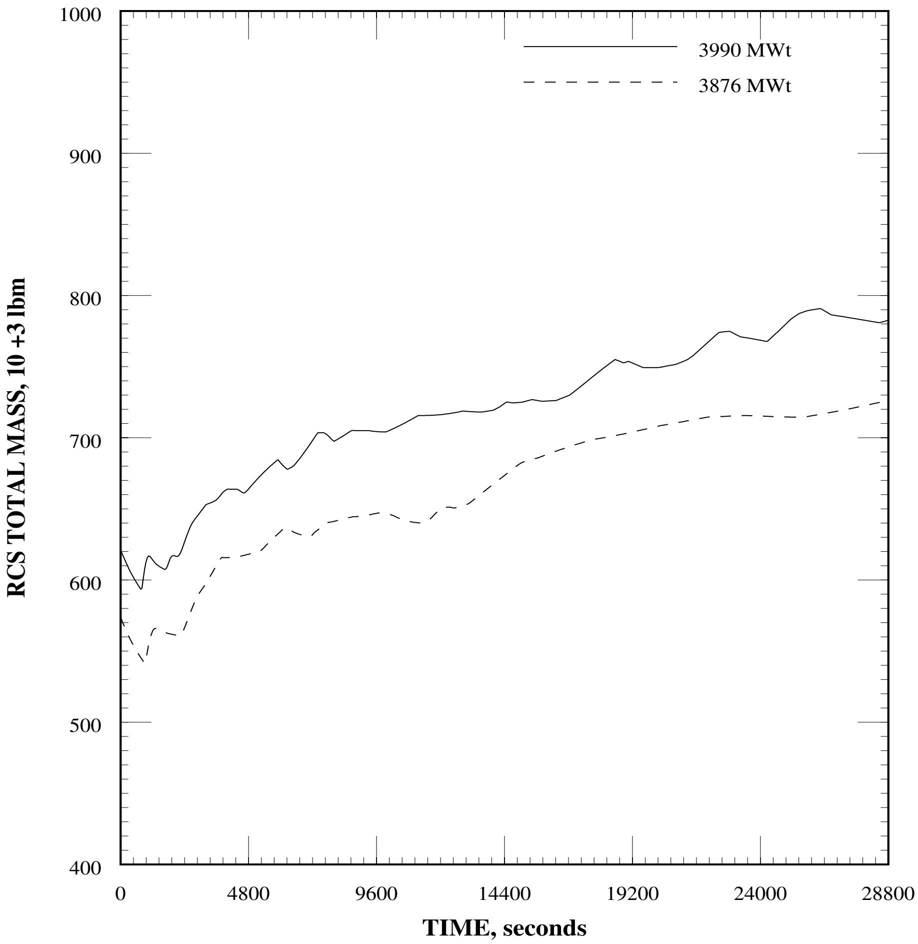


Figure 6.3-230
SGTRLOP Event - SG Pressure vs. Time

(Sheet 1 of 3)

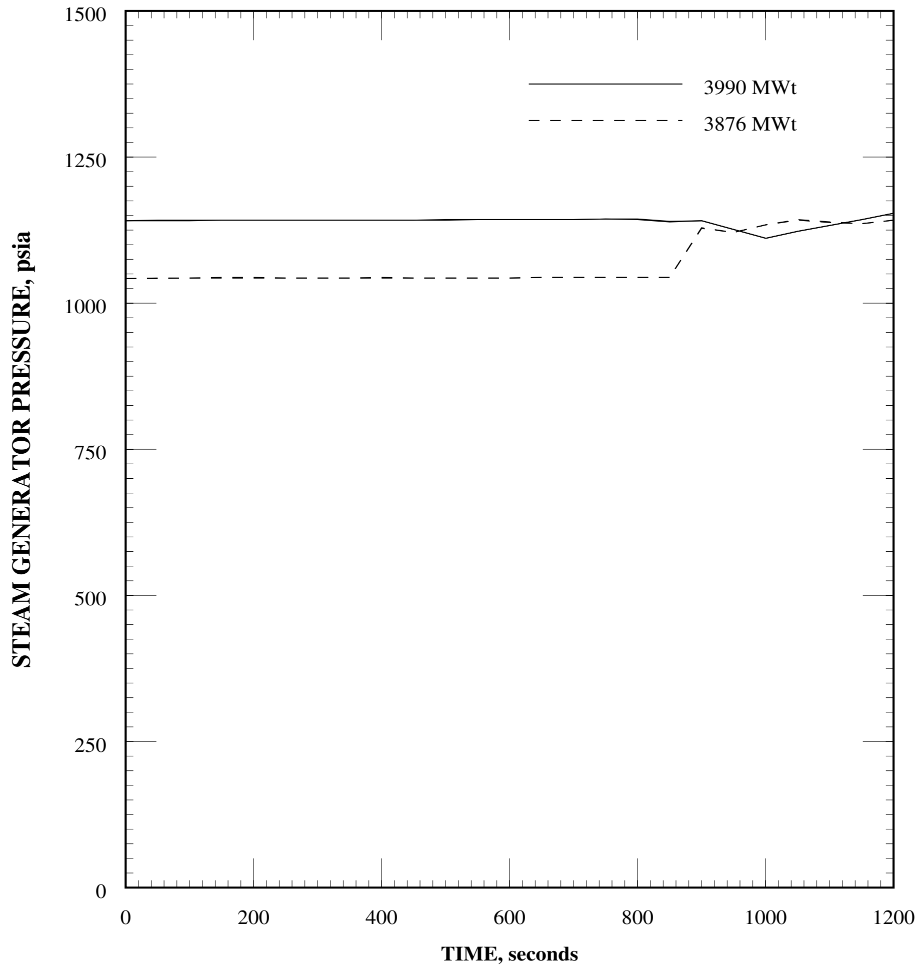


Figure 6.3-230
SGTRLOP Event - SG Pressure vs. Time

(Sheet 2 of 3)

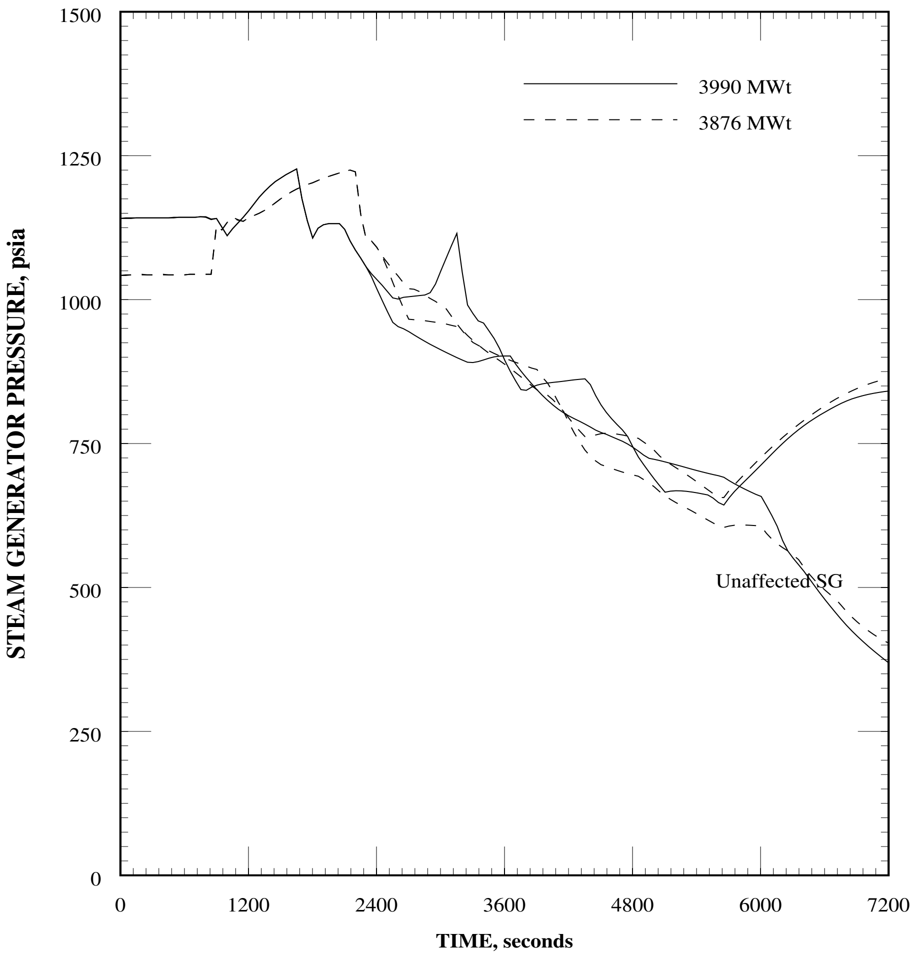


Figure 6.3-230
SGTRLOP Event - SG Pressure vs. Time

(Sheet 3 of 3)

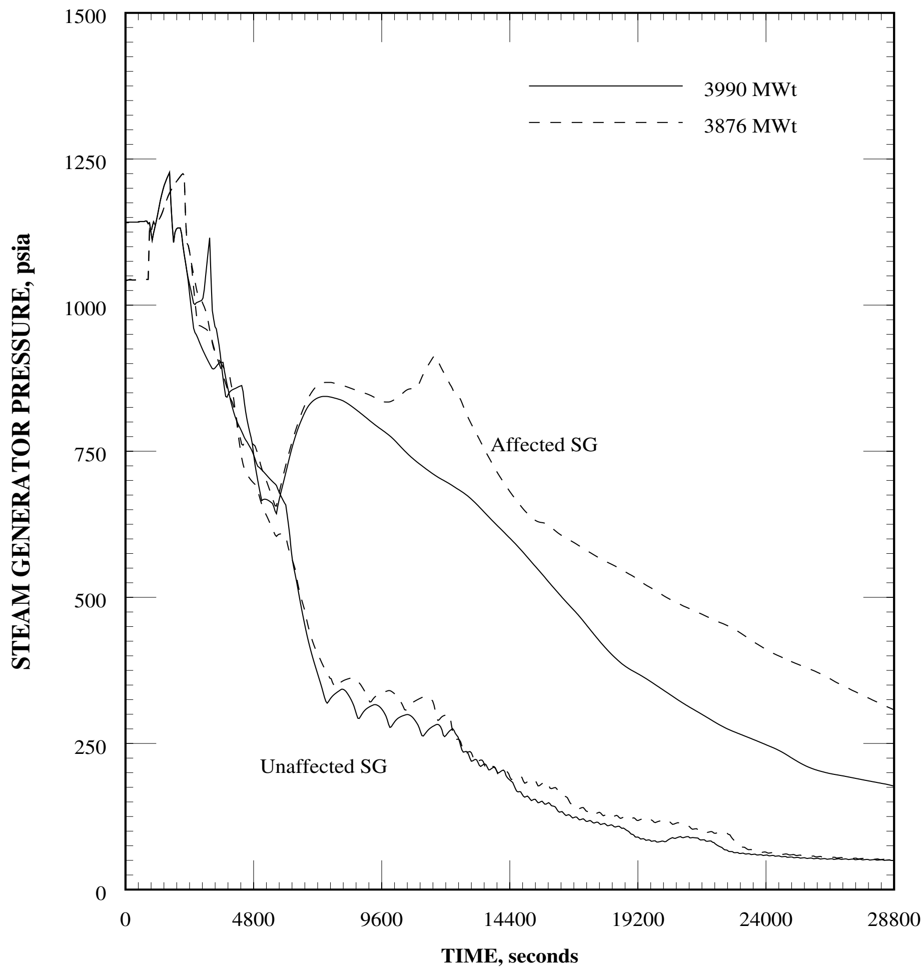


Figure 6.3-231
SGTRLOP Event - Tube Leak Rate vs. Time
(Sheet 1 of 3)

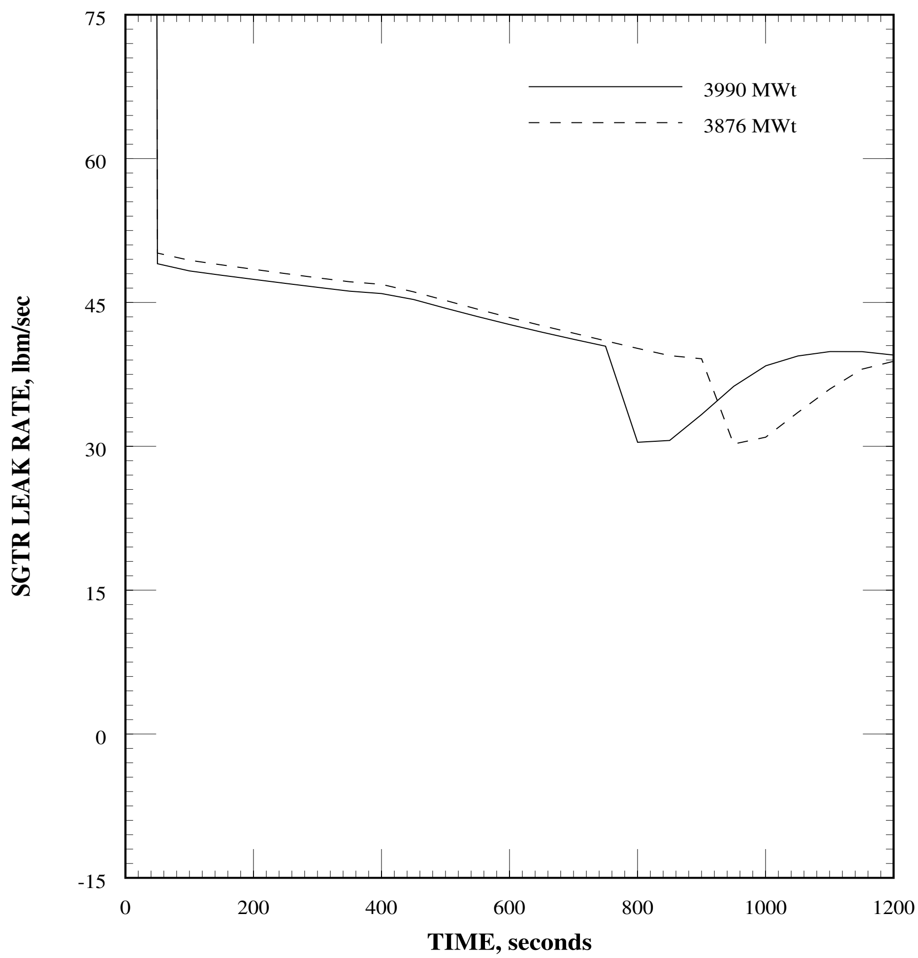


Figure 6.3-231
SGTRLOP Event - Tube Leak Rate vs. Time

(Sheet 2 of 3)

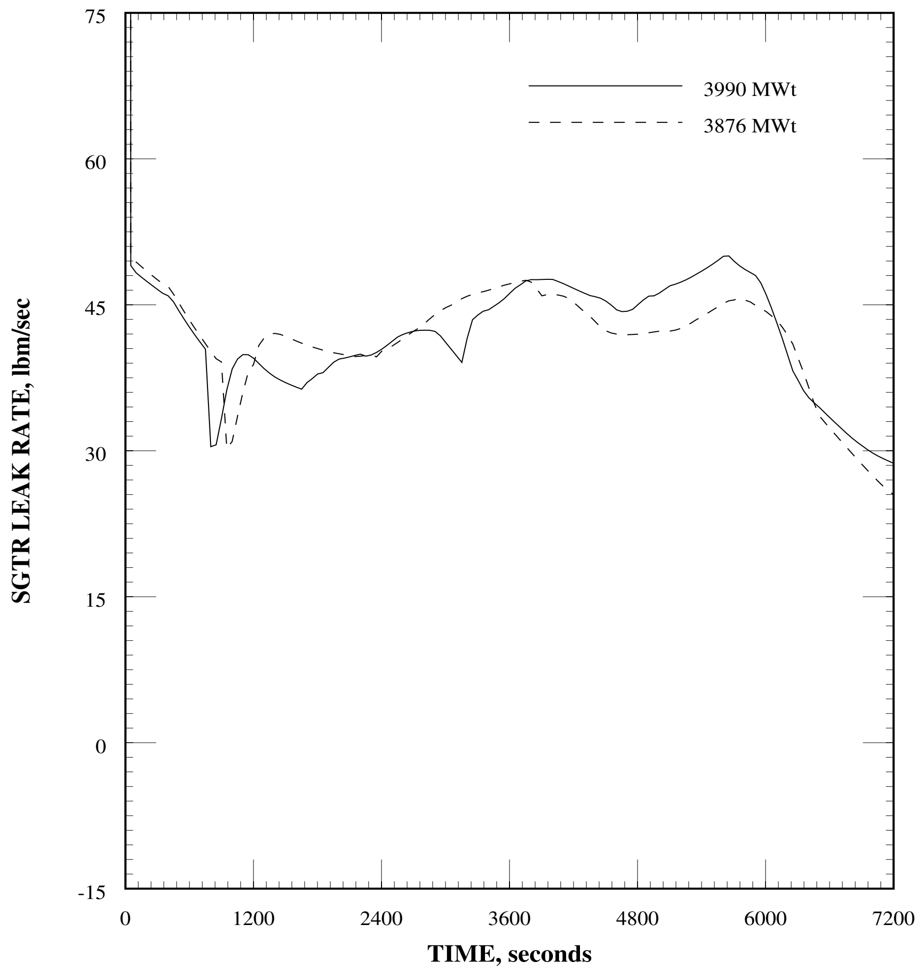


Figure 6.3-231
SGTRLOP Event - Tube Leak Rate vs. Time

(Sheet 3 of 3)

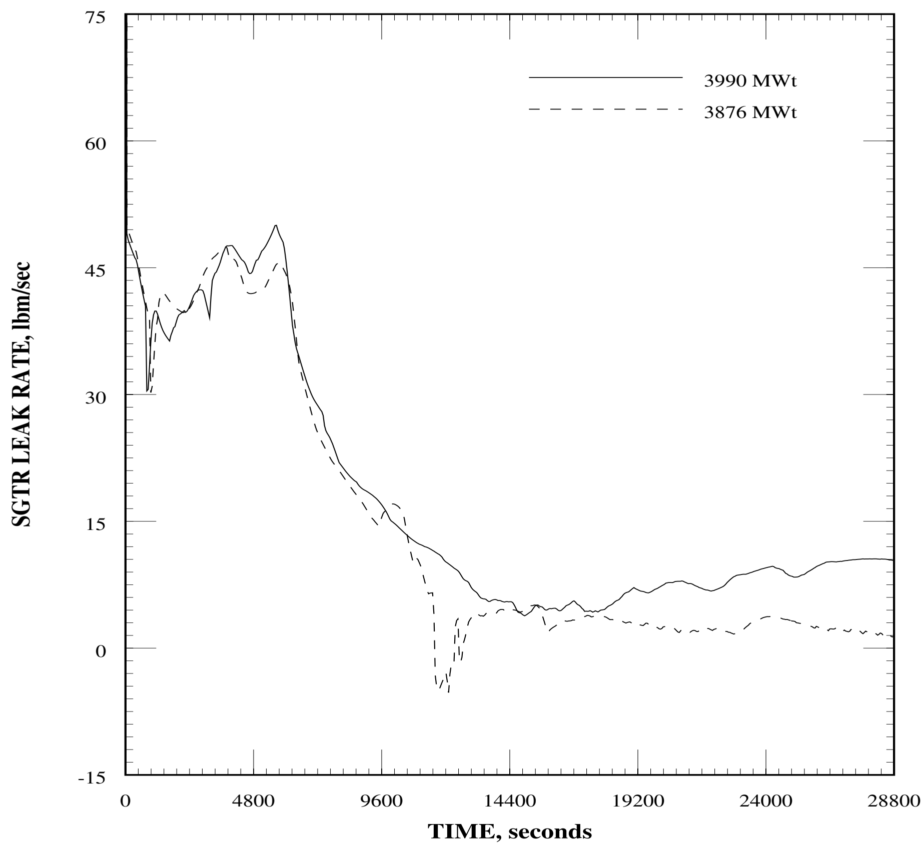


Figure 6.3-232
SGTRLOP Event - Integrated Tube Leak vs. Time
(Sheet 1 of 3)

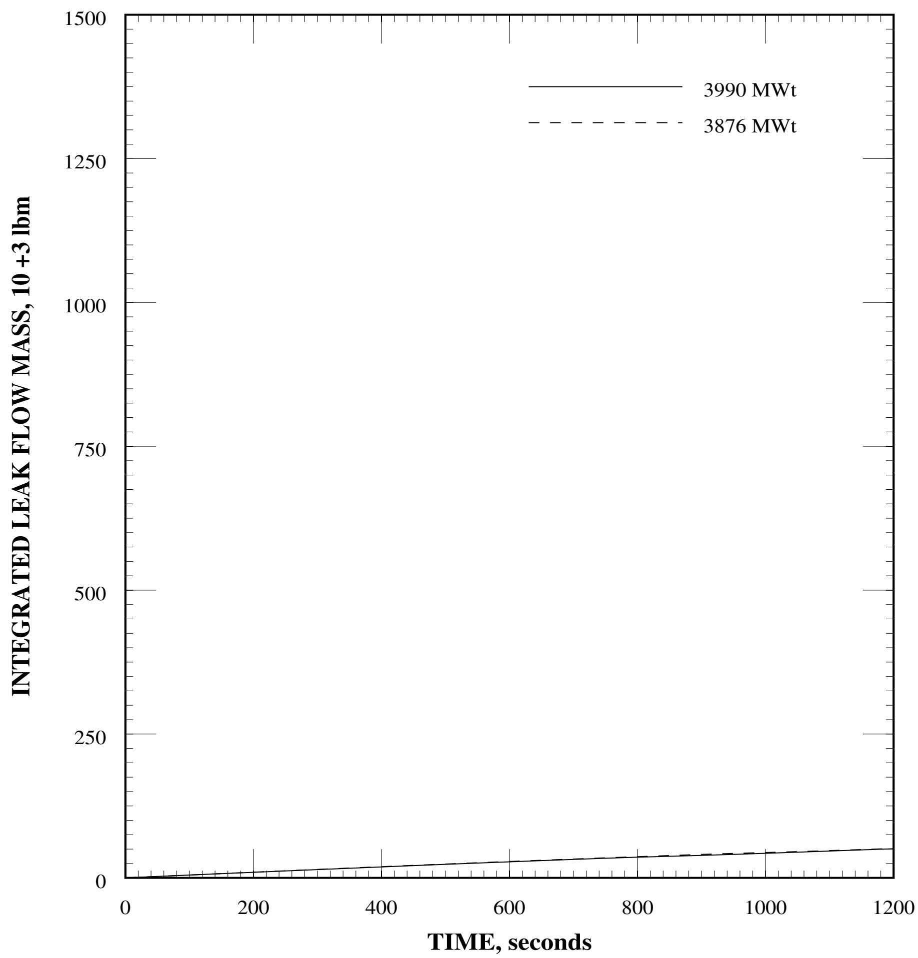


Figure 6.3-232
SGTRLOP Event - Integrated Tube Leak vs. Time

(Sheet 2 of 3)

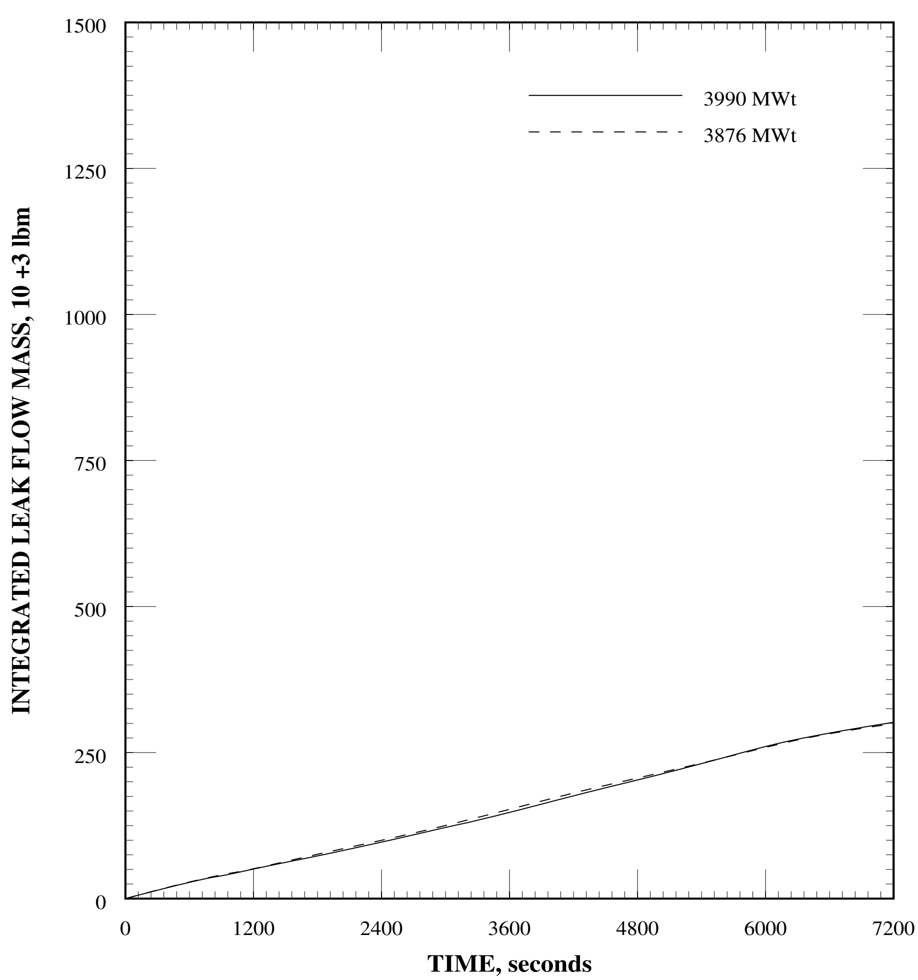


Figure 6.3-232
SGTRLOP Event - Integrated Tube Leak vs. Time

(Sheet 3 of 3)

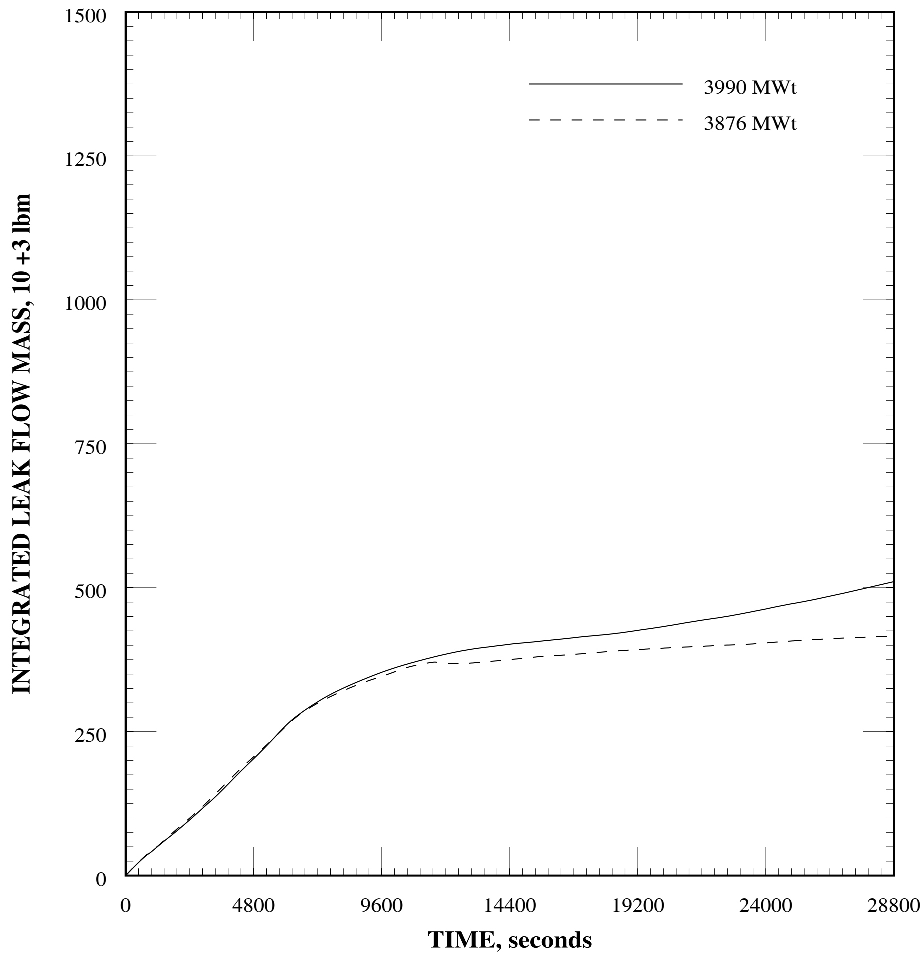


Figure 6.3-233
SGTRLOP Event - Leak Flash Fraction vs. Time
(Sheet 1 of 2)

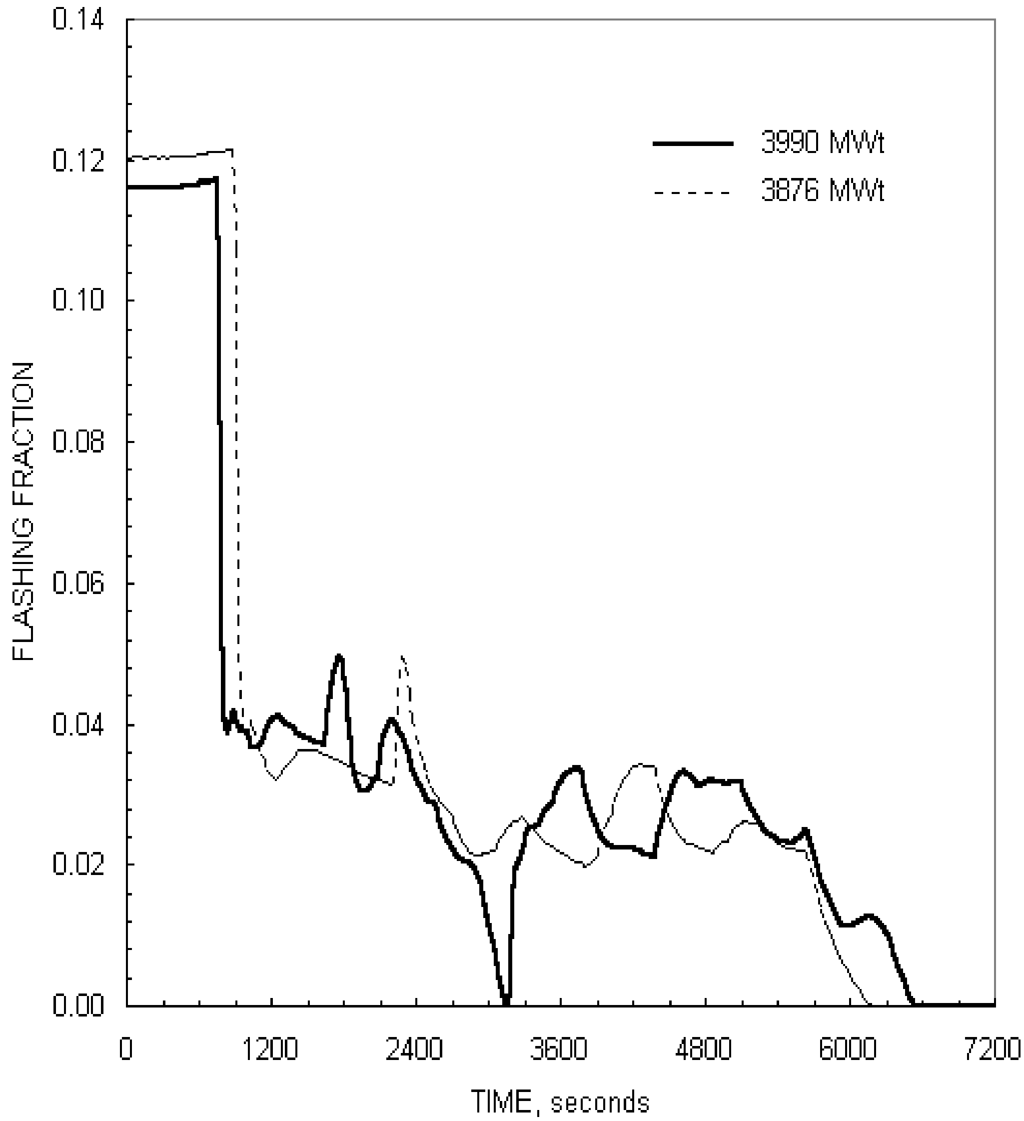


Figure 6.3-233
SGTRLOP Event - Leak Flash Fraction vs. Time

(Sheet 2 of 2)

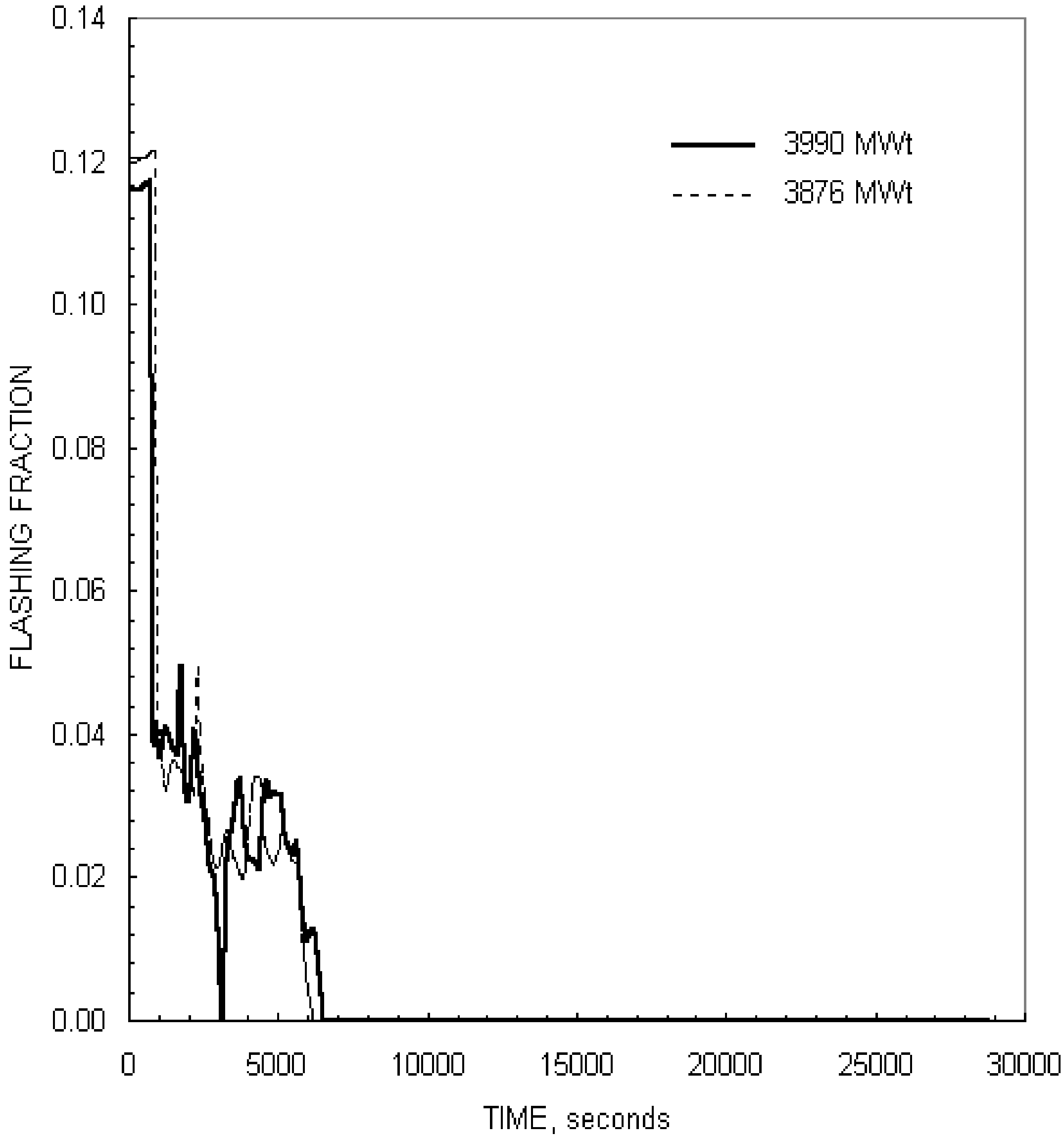


Figure 6.3-234
SGTRLOP Event - SG Liquid Inventory vs. Time
(Sheet 1 of 3)

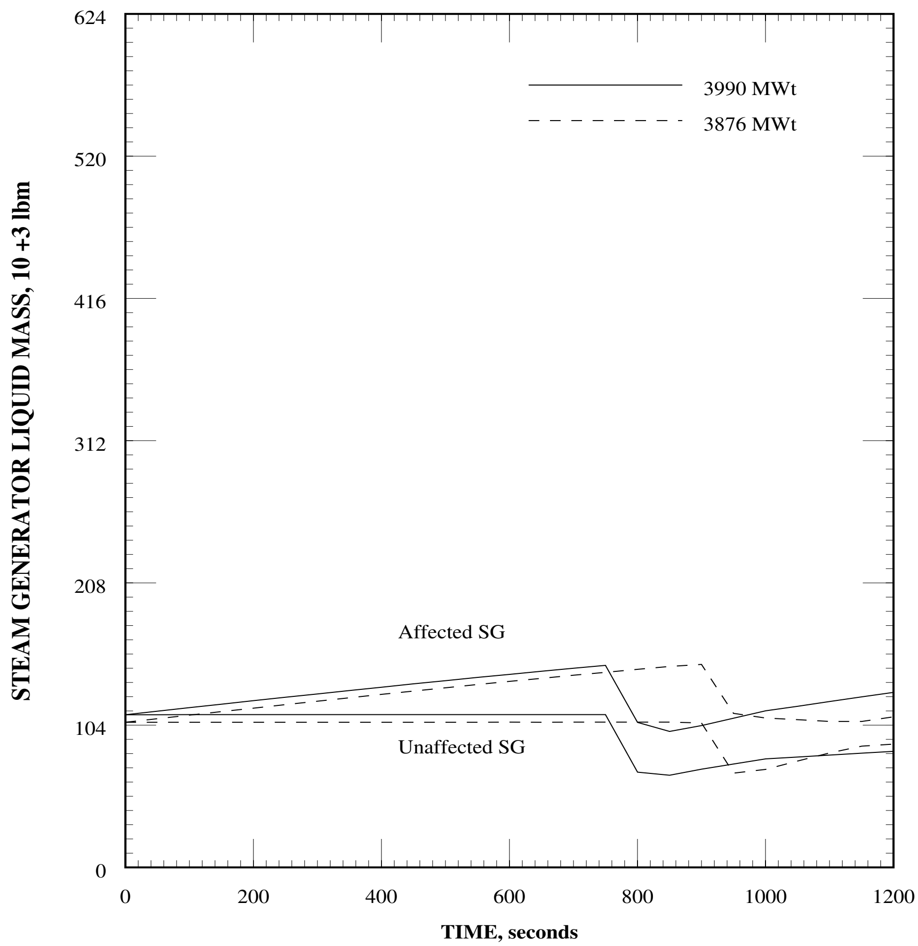


Figure 6.3-234
SGTRLOP Event - SG Liquid Inventory vs. Time

(Sheet 2 of 3)

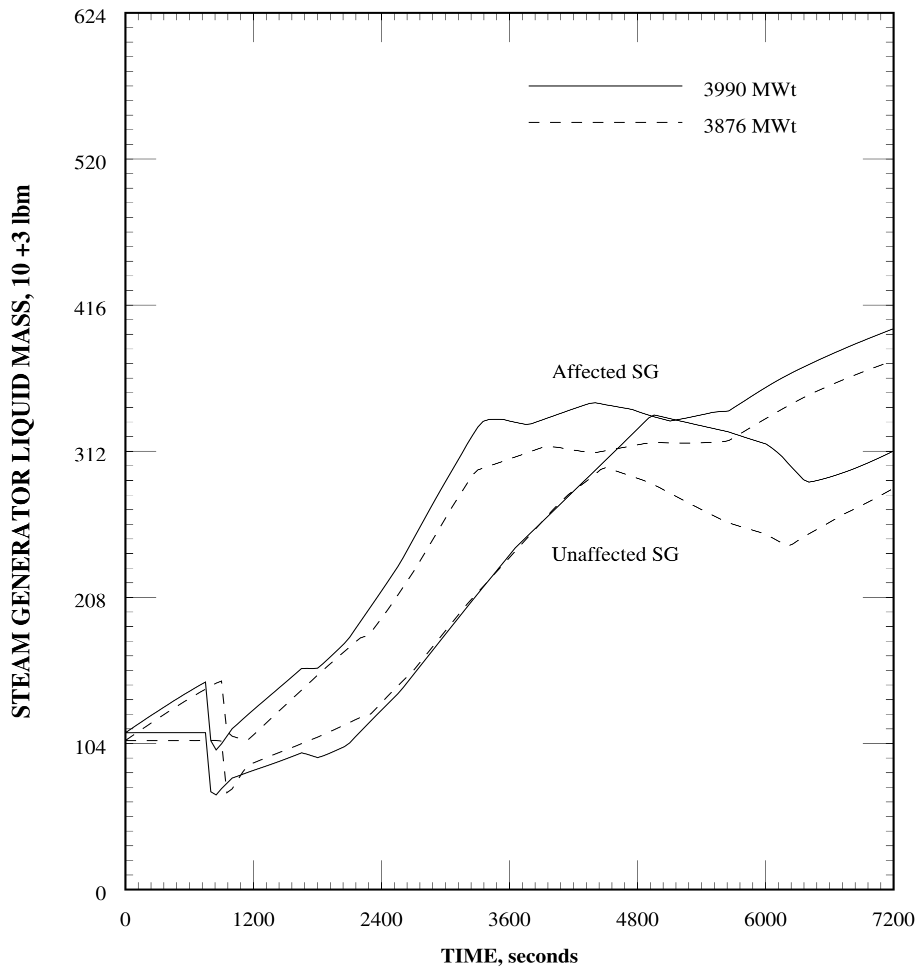


Figure 6.3-234
SGTRLOP Event - SG Liquid Inventory vs. Time

(Sheet 3 of 3)

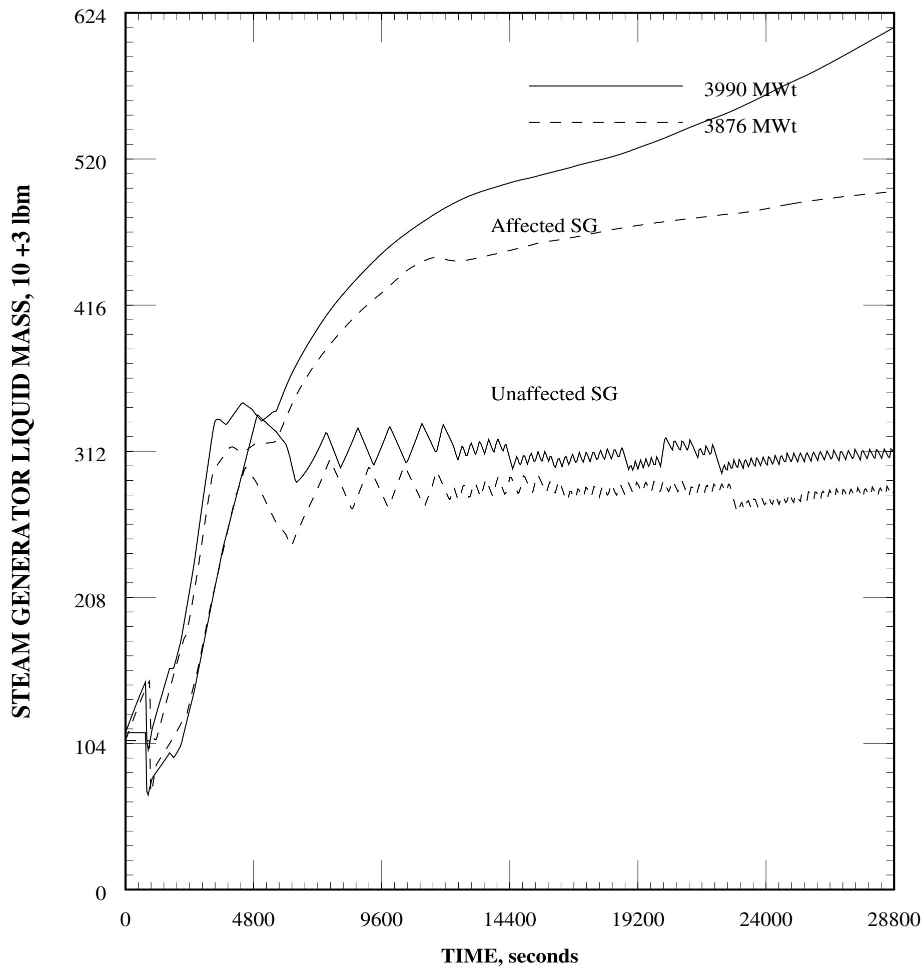


Figure 6.3-235
SGTRLOP Event - Integrated SI Flow vs. Time

(Sheet 1 of 2)

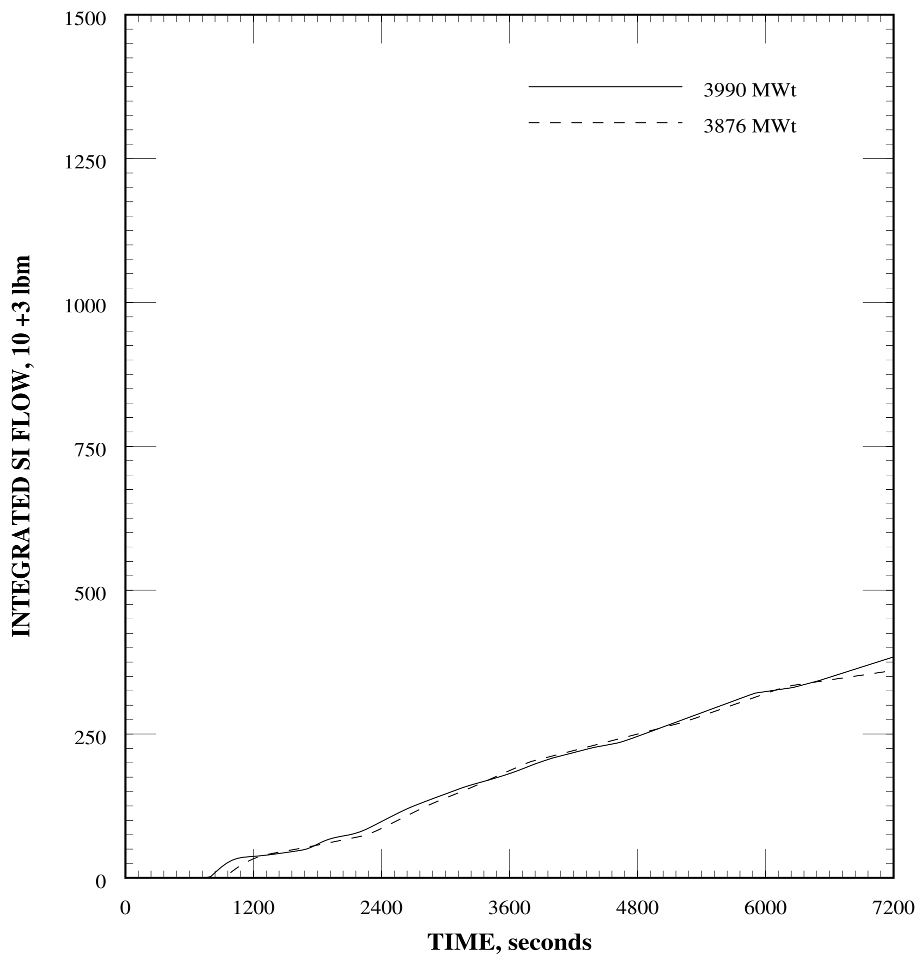


Figure 6.3-235
SGTRLOP Event - Integrated SI Flow vs. Time

(Sheet 2 of 2)

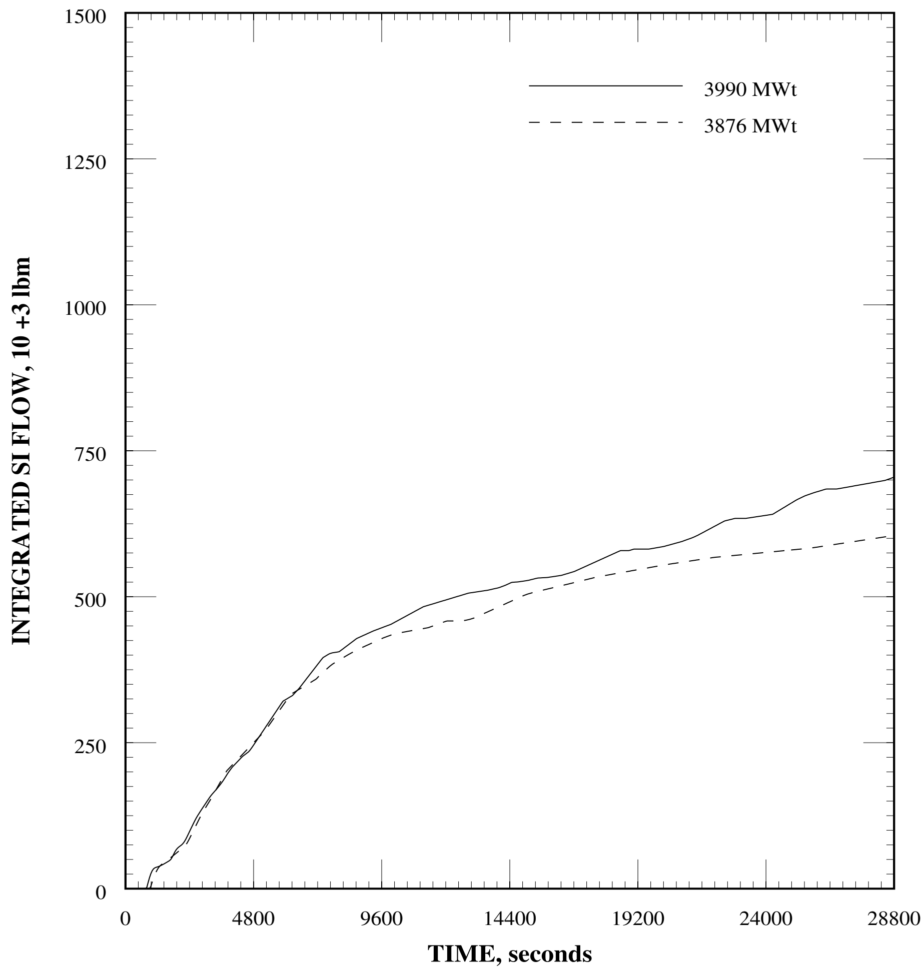


Figure 6.3-236
SGTRLOP Event - AFW Flow vs. Time

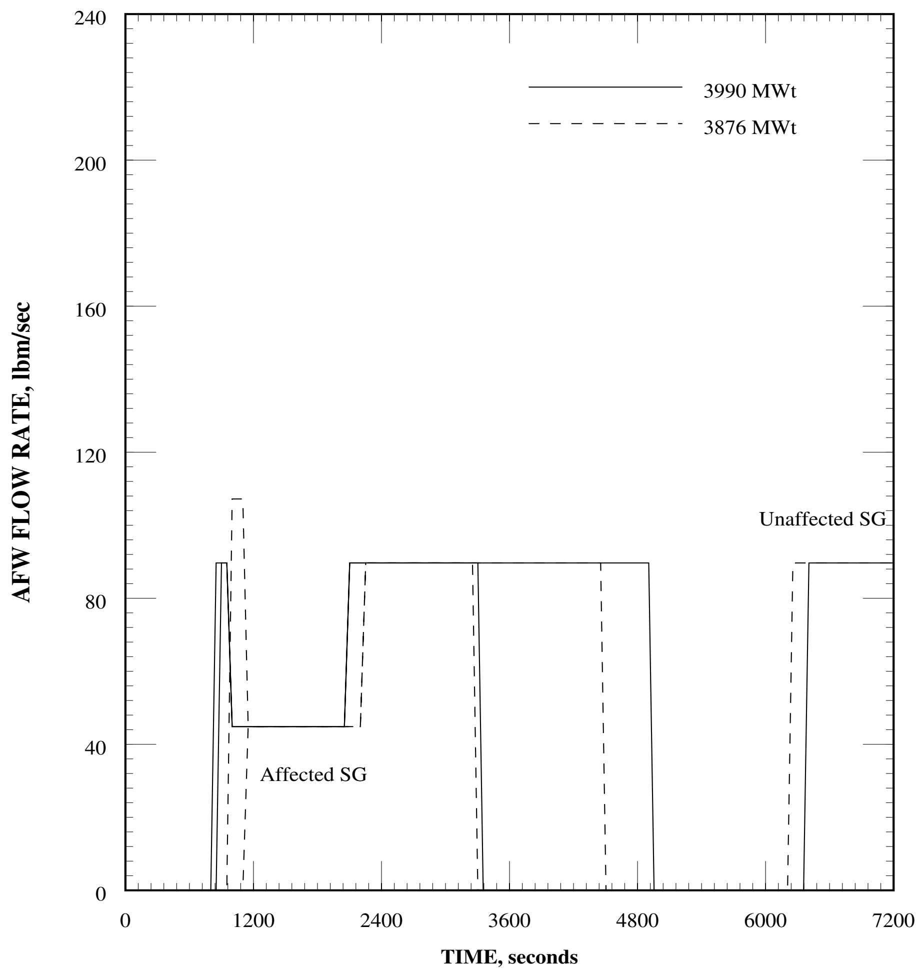


Figure 6.3-237
SGTRLOP Event - SG Safety Flow vs. Time

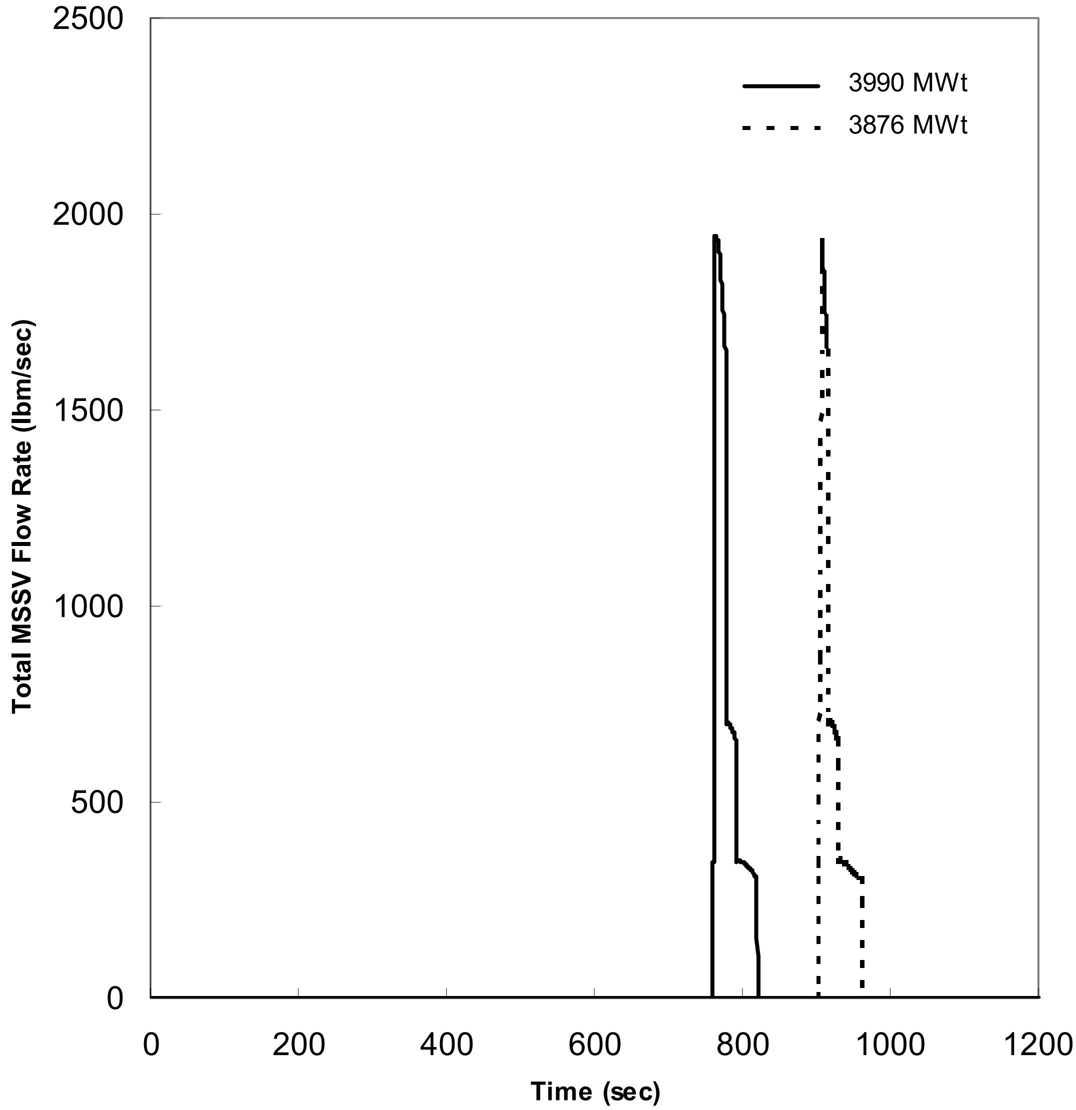


Figure 6.3-238
SGTRLOP Event - Integrated ADV Flow vs. Time
(Sheet 1 of 2)

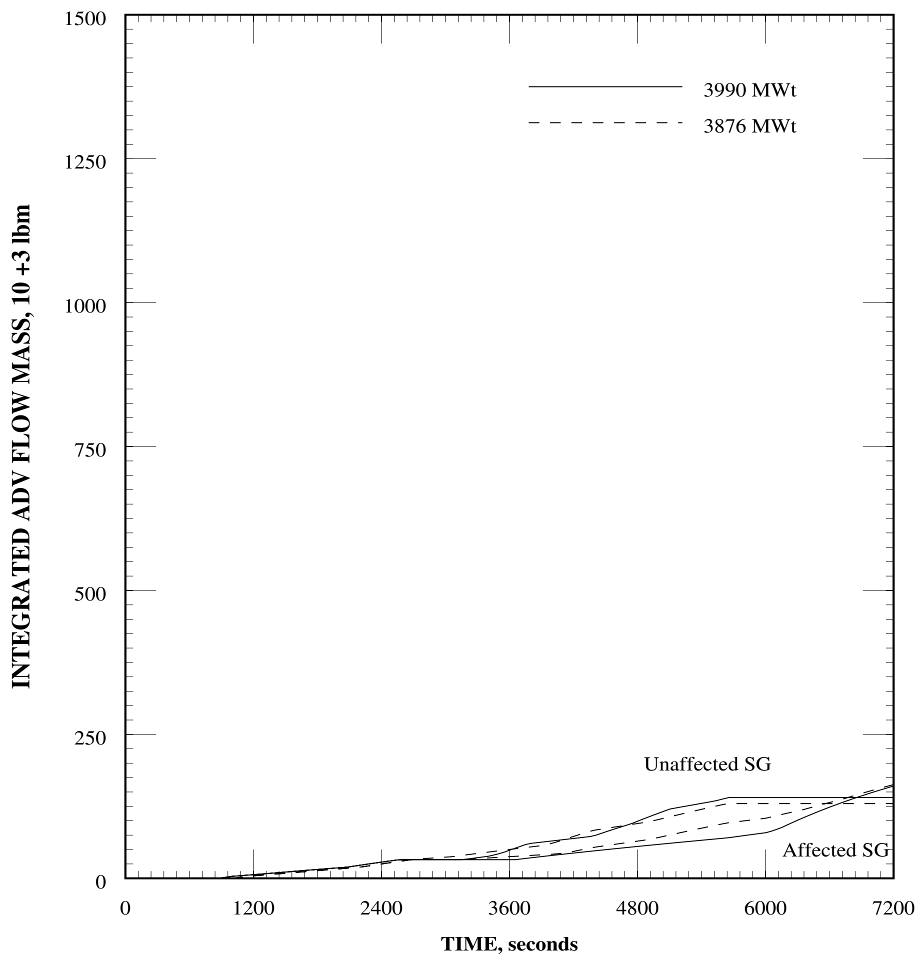


Figure 6.3-238
SGTRLOP Event - Integrated ADV Flow vs. Time

(Sheet 2 of 2)

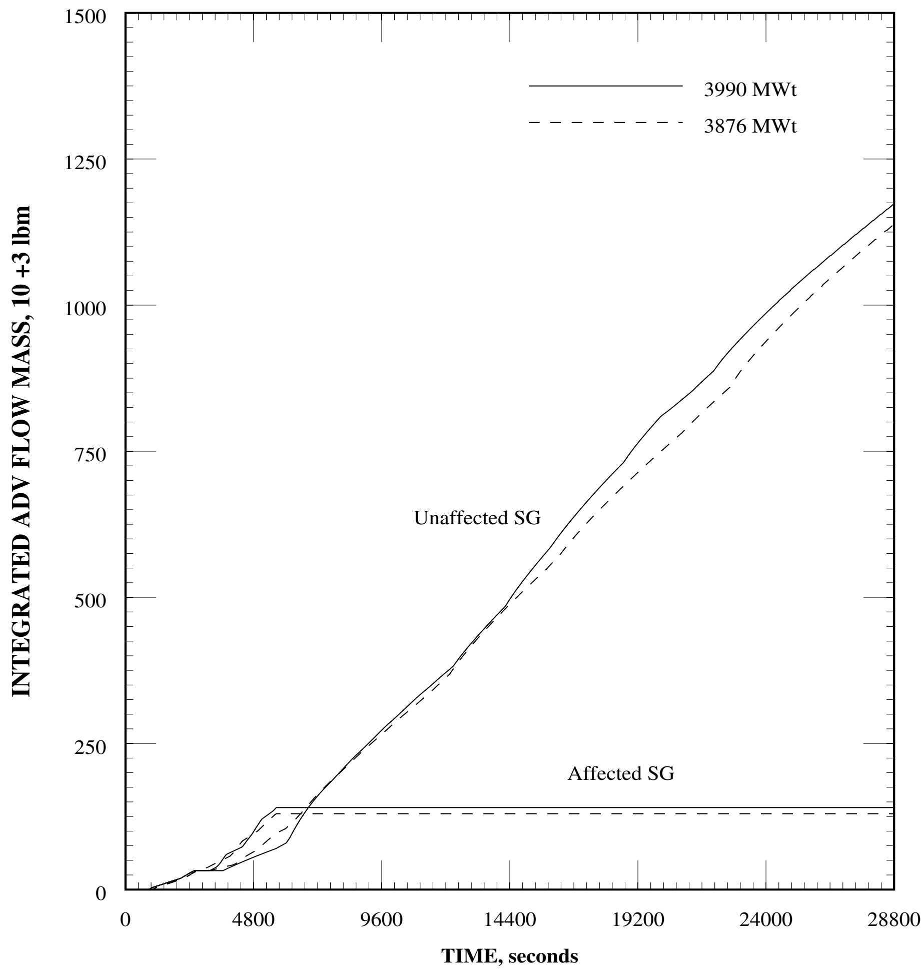


Figure 6.3-239
SGTRLOP Event - Subcooled Margin vs. Time

(Sheet 1 of 2)

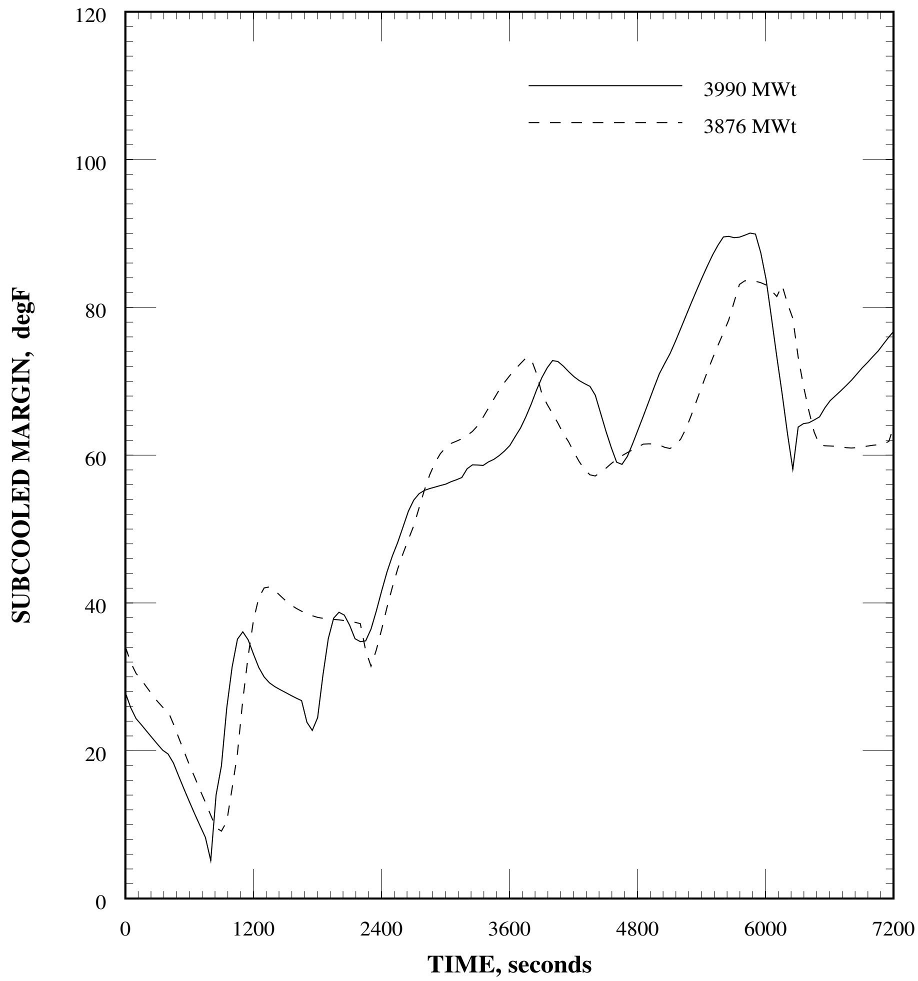


Figure 6.3-239
SGTRLOP Event - Subcooled Margin vs. Time

(Sheet 2 of 2)

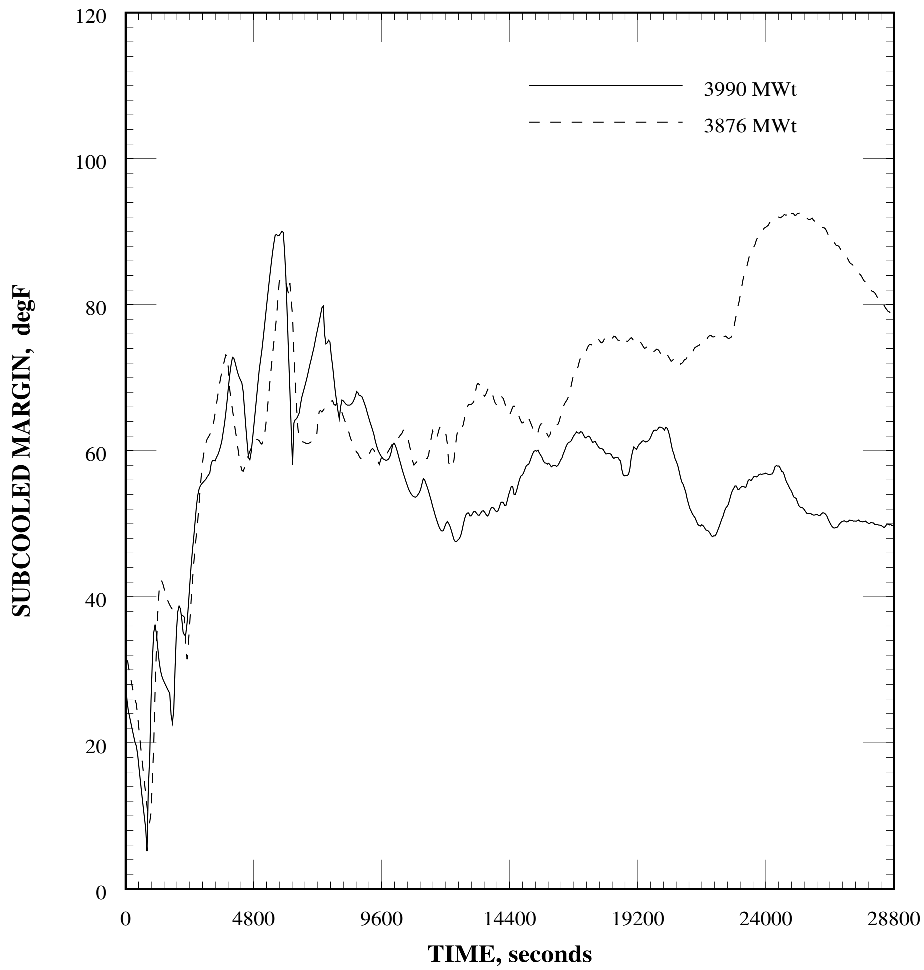
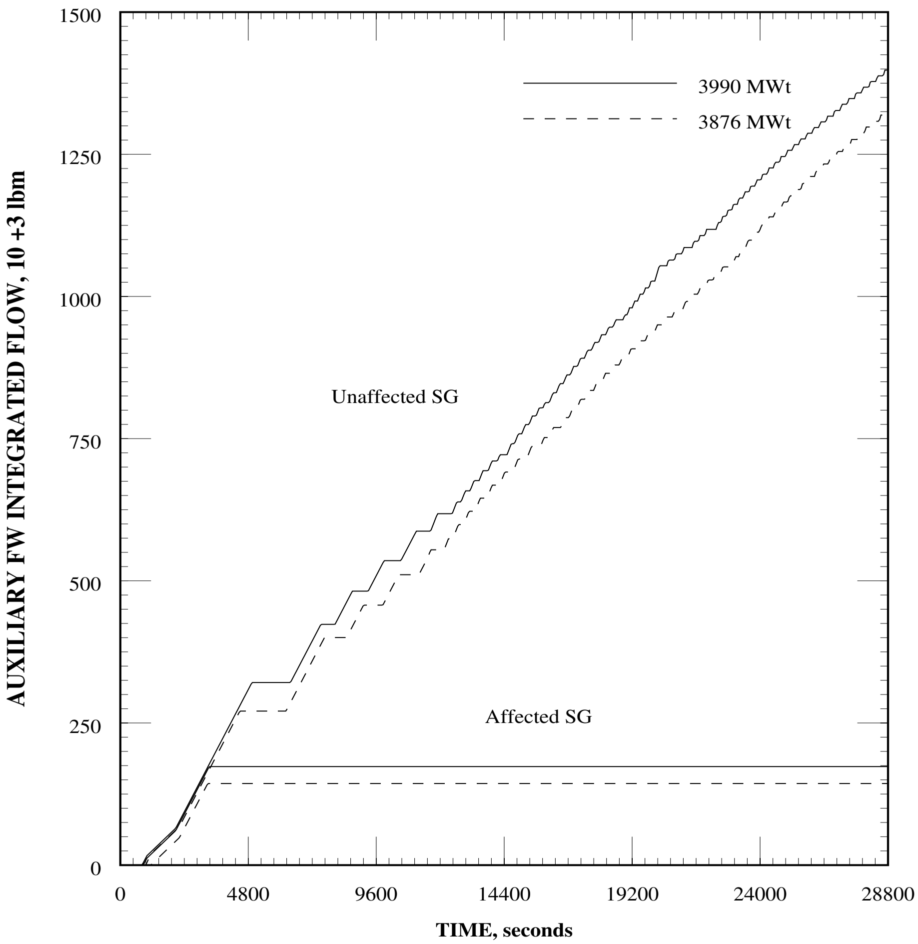


Figure 6.3-240
SGTRLOP Event - Integrated AFW Flow vs. Time



Section 6.3.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

As described in UFSAR Section 15.6.4, this event applicable to BWRs only.

Section 6.3.6.5 Loss-of-Coolant Accidents

ECCS performance and LOCA are discussed in Section 6.1.

Radiological consequences for this event are presented in Section 6.4.6.3.

Section 6.3.7 Radioactive Material Release from a Subsystem or Component

UFSAR Section 15.7 describes radioactive material release from a subsystem or component. Specifically:

- UFSAR Section 15.7.1 Waste Gas System Failure,
- UFSAR Section 15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere),
- UFSAR Section 15.7.3 Postulated Radioactive Releases due to Liquid-Containing Tank Failures, and
- UFSAR Section 15.7.4 Radiological Consequences of Fuel Handling Accidents.

These transients are included in this submittal as part of LOCA/non-LOCA events and are not impacted by PUR. These transients are not impacted because of the non-mechanistic or random determination of failures is not changed by PUR.

In the case of fuel handling accidents, the fuel structural integrity is discussed in Section 7.3. UFSAR Section 15.7.4 indicates that the mechanism for fuel failure does not change for the increase in power level.

Radiological consequences for these events are presented in Section 6.4.7.

Section 6.3.8 Limiting Infrequent Events

Section 6.3.8.1 Anticipated Operational Occurrences in Combination with a Single Active Failure

As an analytical simplification, a composite event was created to bound the DNBR degradation for all infrequent events - including AOOs in combination with a single active failure. When determining the actual limiting infrequent event, all combinations of initiating events and single active failures need to be evaluated. To avoid evaluating all of the potential initiating AOOs, the composite event assumes that an unspecified initiating event degrades all the thermal margins preserved by COLSS and brings core conditions to the DNBR SAFDL. This assumption is conservative since the AOOs are specifically analyzed to ensure that SAFDLs are not violated and the necessary thermal margin is preserved by the LCOs. The most limiting single active failure for DNBR degradation is a LOP, resulting in the coastdown of all four RCPs. Therefore, the composite event is defined as a LOF from SAFDL conditions.

The limiting AOOs for peak linear heat rate are the bank CEAW events presented in Section 6.3.4.1. There are no single active failures nor postulated operator errors that could occur with these events that would produce more severe consequences.

Section 6.3.8.1.1 Acceptance Criteria

As defined in the SRP Section 15.1.1, the specific acceptance criterion is:

- An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Offsite radiological consequences must be limited to a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body).

Section 6.3.8.1.2 Description of Analysis

A set of initial conditions corresponding to the DNBR SAFDL was calculated with the CETOP-D code. The core average and hot channel response to the LOF event from these initial conditions was simulated using the 1D HERMITE code. The transient DNBR values were calculated using the CETOP-D code that uses the CE-1 CHF correlation. At the time of minimum DNBR, a more accurate prediction of the DNBR was calculated using the more detailed TORC code.

Both the existing configuration and PUR were analyzed. Input parameters and initial conditions were selected to maximize the DNBR degradation.

Section 6.3.8.1.2.1 Transient Simulation

The limiting AOO with single failure event is modeled as a LOF from SAFDL conditions. An undefined “Limiting AOO” is assumed to degrade all available COLSS margin and forces the hot channel DNBR to the SAFDL. At this point the limiting single failure, LOP, occurs and further degrades DNB. The SAFDL conditions include an assumed, pre-existing power of 116%, due to the undefined limiting AOO.

Although a LOP would not occur for at least three seconds following a turbine trip, this evaluation conservatively assumes a coincident turbine trip and LOP. The RCP coastdown leads to a CPC DNBR reactor trip. RCS flow coastdown degrades DNBR below the initial SAFDL conditions. DNBR degradation is terminated when the mitigating effects of the scram CEA insertion dominate the flow coastdown.

Section 6.3.8.1.3 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-53 contains the initial conditions used for the LOF from SAFDL event.

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 including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. There is no operator action for the first 30 minutes of the event.

Table 6.3-53
Parameters Used for the Limiting Infrequent Event

PARAMETER	Value 3990 MW_t
Initial core power (% of rated)	116
Initial core inlet temp (°F)	548
Initial pressurizer pressure (psia)	2325
Initial RCS flow (% of design)	98
Initial pressurizer level (ft)	N/A
Initial SG level (ft)	N/A
MTC ($\Delta\rho$/°F)	0.0E-04
FTC	least negative
Kinetics	maximum β
CEA worth at trip - WRSO (%$\Delta\rho$)	-8.0
Fuel rod gap conductance (Btu/hr-ft^2-°F)	660
Plugged SG tubes (% of tubes/SG)	N/A
Single failure	none
LOP	yes

Section 6.3.8.1.4 Results

The limiting infrequent event (i.e., AOO with single failure) is simply modeled as a LOF from SAFDL conditions. This conservative modeling approach is intended to bound the DNBR degradation of all Infrequent AOOs. Since this approach only models the core average and hot channel in HERMITE, a sequence of events table and plots depicting the NSSS response are not provided.

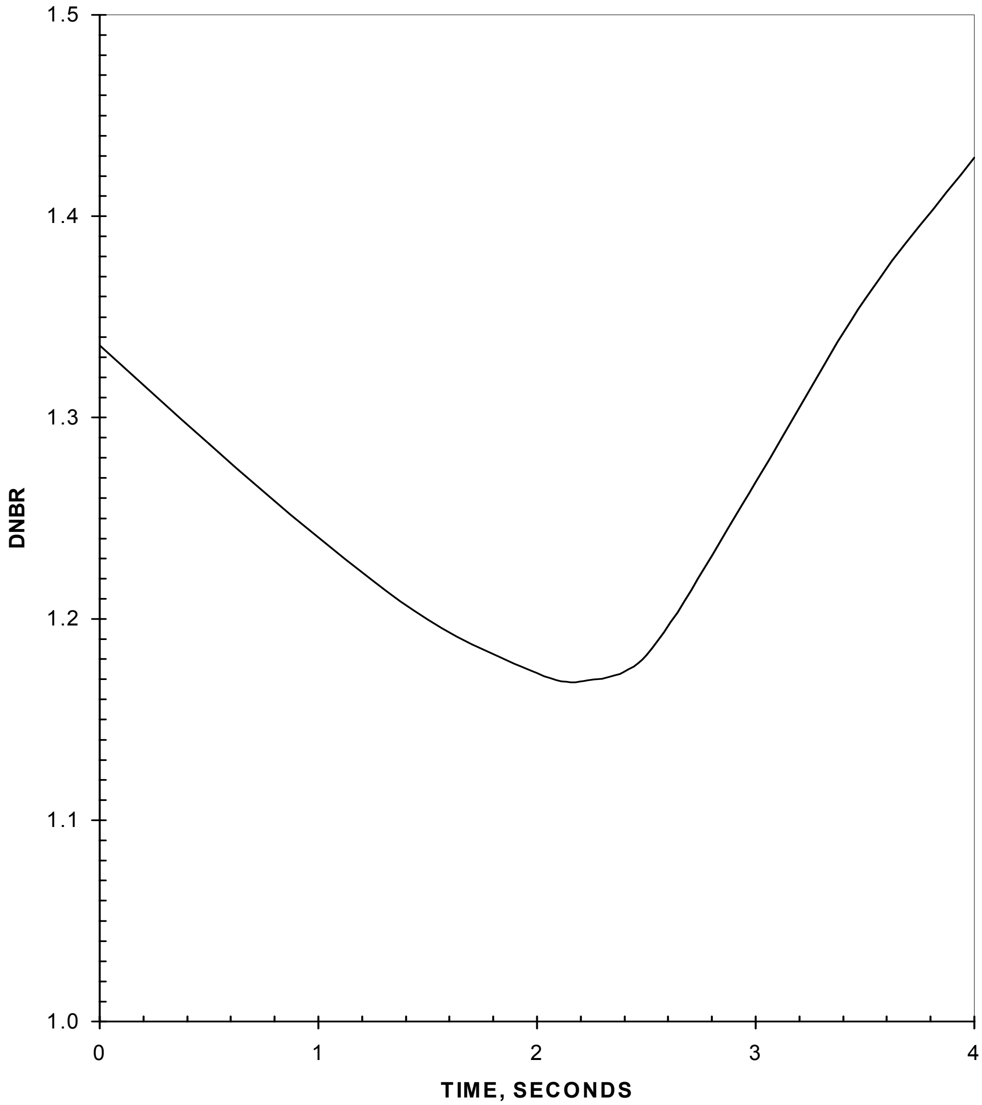
Starting from SAFDL conditions, the LOP results in an RCP coastdown that leads to a CPC DNBR reactor trip. The reduction in RCS flow degrades DNBR below the initial SAFDL conditions. Within 3 seconds of reactor trip, local and average core heat flux has decreased enough such that no pins remain in DNB. Hence, DNB propagation is not predicted to occur. Figure 6.3-241 provides the transient DNBR response for the event.

Radiological consequences for this event are presented in Section 6.4.8.

Section 6.3.8.1.5 Conclusions

The limiting infrequent event (i.e., AOO with single failure) results in a limited number of fuel pins predicted to be in DNB for a few seconds. DNB propagation is not predicted to occur. Offsite doses remained below the acceptance criteria for this category of event. Specifically, a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body).

Figure 6.3-241
Limiting AOO with Single Failure Event - DNBR vs. Time at 3990 MW_t



Section 6.3.8.2 Anticipated Transient Without Scram

10 CFR Part 50.62 requires that PWRs manufactured by Combustion Engineering have systems diverse from the reactor trip system to scram the reactor, trip the turbine, and initiate AFW under conditions of an Anticipated Transient Without Scram (ATWS). The Diverse Auxiliary Feedwater System (DAFAS), Supplemental Protection System (SPS), and Diverse Turbine Trip (DTT) response satisfies the ATWS Rule requirements.

Per the UFSAR, the SPS will provide the required reactor trip (on high pressurizer pressure) and subsequent turbine trip and DAFAS will provide an AFAS (on low SG level) during an ATWS. The SPS and DAFAS setpoints were selected to initiate the required action, but not to interfere with the PPS and AFAS.

The increase in rated thermal power from 3876 MW_t to 3990 MW_t and the characteristics of the SGs (i.e., larger RCS and secondary side inventory, increased heat transfer area, etc.) will not require a change to the existing PPS HPPT setpoint, PSV setpoint, and AFAS setpoint. The changes will not diminish the effectiveness of the SPS and DAFAS response or require changes to their setpoints. Therefore, the SPS, DTT, and DAFAS systems will continue to be within the ATWS Rule requirements.

Section 6.3.8.3 Station Blackout

The Station Blackout (SBO) requirements of Regulatory Guide 1.155 (Reference 6-30) are for a 4-hour coping plant. In response to the SBO rule, two Gas Turbine Generators (GTGs) and their associated equipment act as the alternate AC power source. The alternate AC source was designed to be available within 1 hour of the onset of a SBO and would power the equipment necessary to cope with an SBO for the remaining 3 hours of the coping time duration. The GTGs are described in UFSAR Section 8.3.1.1.10.

The increase in rated thermal power from 3876 MW_t to 3990 MW_t will not challenge the ability to cope during a complete loss of AC power, as described below.

Section 6.3.8.3.1 Auxiliary Feedwater and Steam Release

Following a reactor trip, the steam-turbine-driven AFW pump and the pneumatically operated ADVs are used to remove decay heat. The AFW steam flow control valves are dc-powered Motor Operated Valves (MOVs) and would be available for the SBO coping time duration. The dc batteries have been sized with sufficient capacity to independently supply the loads required for SBO coping. The ADVs that are air operated during normal power operations, are equipped with air accumulators and back-up nitrogen gas to facilitate remote/manual operation after a loss of instrument air. The capacity of the back-up nitrogen supply to the ADVs is sufficient for decay heat removal following PUR.

The AFW and ADV systems do not require increased electrical demand as a function of decay heat for their operation. Therefore, the ability to remove decay heat during an SBO remains acceptable following the PUR implementation.

Section 6.3.8.3.2 Loss of Ventilation

Areas containing equipment required to mitigate the effects of the SBO event were evaluated to predict peak temperatures. The AFW steam flow control valves are located in the steam turbine pump room, while the ADVs are located in the top level of the MSSS. The calculated peak temperatures for the AFW pump room and MSSS were found to be within the acceptable limits for equipment operability described in Regulatory Guide 1.155 and NUMARC 87-00 (Reference 6-30 and Reference 6-31).

The impacts of PUR on the total containment heat loads were evaluated in Section 6.2. Containment temperatures during an SBO remain below previously reviewed values for the existing plant configuration.

Section 6.3.8.3.3 Condensate Storage Tank Inventory

The calculated CST inventory required for decay heat removal during the 4 hour SBO coping time using NUMARC 87-00 methodology based on operating at an uprated core power of 4070 MW_t (3990 MW_t plus 2% for instrument loop uncertainty) is approximately 90,000 gallons. Technical Specification 3.7.6 requires a minimum level of 29.5 ft (inventory greater than 300,000 gallons) in the CST. Thus, there is ample condensate inventory even with the increased decay heat due to PUR.

Section 6.3.8.3.4 Reactor Coolant System Inventory

The original SBO RCS inventory analysis demonstrated the ability to sustain an SBO event with up to 120 gpm RCS leakage for 5 hours without core uncovery. Inclusion of the increased RCS inventory due to larger SGs would yield more shrink during cooldown, but it would also provide more inventory above the core. The overall impact would not produce core uncovery within the required 4 hours SBO duration. After 1 hour, the GTGs may be credited to restore power for one charging pump to supply makeup water to the RCS. Note that ample makeup inventory is available via the RWT (Technical Specification 3.5.5 requires 600,000 gallons during Mode 1) to account for a portion of the RCS leakage postulated to be lost during SBO coping time.

Core cooling following a SBO is maintained by reflux boiling in the SGs. The characteristics of the SGs would not impede this type of heat transfer. Therefore, sufficient RCS inventory and core cooling would be maintained following a SBO event after PUR implementation.

Section 6.4 Radiological Accident Evaluations

This section presents the radiological assessment of postulated accidents. The consequences of these incidents are compared to the acceptance criteria for acceptability. If an analysis is fuel cycle dependent, a bounding value for the source term/fuel failure was calculated based on the acceptance criteria of the applicable Section of the SRP (Reference 6-6). Accidents evaluated include all significant non-LOCA and LOCA events. These events are:

1. Increased heat removal by secondary system:
   - inadvertent opening of an ADV with a LOP (UFSAR Section 15.1.4) and
   - MSLB outside containment with a LOP (UFSAR Section 15.1.5).
2. Decreased heat removal by secondary system:
   - FWLB event with a LOP (UFSAR Section 15.2.8).
3. Decreased RCS flow:
   - single RCP sheared shaft with a LOP, (UFSAR Section 15.3.4).
4. Reactivity and power distribution anomalies:
   - CEA ejection (UFSAR Section 15.4.8).
5. Increase in RCS inventory:
   - no event in this category was reanalyzed for radiological consequences.
6. Decrease in RCS inventory:
   - DBLLOCUS (UFSAR Section 15.6.2.3.2),
   - SGTR with a single failure and LOP (UFSAR Section 15.6.3),
   - SGTR with LOP (UFSAR Section 15.6.3)
   - LOCA (UFSAR Sections 6.3 and 15.6.5),
   - SBLOCA (UFSAR Section 15.6.5.2.1), and
   - LBLOCA (UFSAR Section 15.6.5.5).
7. Radioactive release from a subsystem or component:
   - postulated radioactivity release due to liquid containing tank failure, (UFSAR Sections 15.7.3 and 2.4.13.3) and
   - fuel handling accident inside containment building and inside the fuel building, (UFSAR Sections 15.7.4.1.3.c and 15.7.4.2.3).
8. Limiting infrequent event.

Section 6.4.0 Methodology Used for Radiological Assessment Analyses

Several of the events discussed result in the release of steam or liquid from the RCS or main steam system. The methodology and input parameters used to assess the radiological consequences are discussed below. Event specific information is noted in applicable sections of this submittal. UFSAR Appendix 15B provides details of models used to assess radiological consequences of postulated accidents.

The atmospheric dispersion characteristics, i.e., Atmospheric Dispersion Factor (X/Q values), are discussed in detail in UFSAR Section 2.3.

LOCADOSE computer code was used for evaluations performed for CEA ejection, LBLOCA, SBLOCA, and events of UFSAR Section 15.7. For all other events, radiological consequences were calculated using a combination of the CENTS code and conservative hand calculations. The CENTS code provided the thermal hydraulic input that is needed for estimating time dependent dose analysis at EAB and LPZ. The data is then used to calculate radiological release to the atmosphere for determining thyroid and whole body doses.

The assumptions used for calculating radiological releases to the atmosphere follow. Any deviation from these assumptions is specified in the event specific discussions.

1. The initial primary system activity level is based on the maximum activity in the RCS due to continuous full power operation with 1% failed fuel (3.6 μCi/cc dose equivalent iodine) for CEA ejection, LBLOCA, SBLOCA, DBLLOCUS, and events of UFSAR Section 15.7. For all other events, Technical Specification, LCO 3.4.17, limit of 1.0 μCi/gm is assumed.
2. The initial secondary system activity level is equal to 0.1 μCi/gm dose equivalent I-131 per Technical Specification, LCO 3.7.16.
3. Core maximum isotopic inventory is provided in Table 6.4-1. Maximum core activity for each isotope is calculated using TID-14844 methodology (Reference 6-16) assuming infinite burn-up time, with the exception of Kr-85 (Reference 6-16). The value for Kr-85, was calculated using the ORIGEN-S code assuming 70,000 MWD/MTU burn-up. Core thermal power of 4070 MW_t (3990 MW_t with 2% power uncertainty) was used for these evaluations.
4. It is conservatively assumed that all fission products in the fuel rods gap (10% of fuel inventory of noble gases and iodines per Regulatory Guide. 1.77, Reference 6-26) would be released to the RCS instantaneously at a time when minimum DNBR is reached. The number of failed fuel rods equals the number of rods that experience DNB, as calculated with a statistical convolution technique. The statistical convolution technique involves the summation, over the reactor core, of the number of fuel rods with a specific DNBR value, multiplied by the probability of DNB at that DNBR value.
5. Primary-to-secondary SG tube leakage of 1 gpm/2 SGs is included in the calculation of activity releases to atmosphere via secondary system release path.
6. For events for which the SRP (Reference 6-6) requires consideration of “iodine spiking” the following are used:
    - GIS: the I-131 Dose Equivalent (DEQ) appearance rate is increased by a factor of 500.
    - PIS: for an abnormally high iodine concentration due to a previous iodine spike, a RCS activity of 60 μCi/cc DE I-131 is assumed.
7. SG iodine DFs used for non-LOCA scenarios are:
    - empty (dry), DF = 10 and

- level maintained, DF = 100.

8. Condenser generates an iodine DF = 100 used for non-LOCA analysis.
9. Breathing rates are obtained from Regulatory Guide 1.4 (UFSAR Table 15B-3, Reference 6-14).
10. X/Q is from Table 2.3-31 in the UFSAR.
11. Iodine inhalation dose conversion factors (thyroid) are based on the ICRP 30 and Technical Specifications Section 1.1 (Reference 6-27).
12. All other dose conversion factors are obtained from Regulatory Guide 1.109 (Reference 6-29).
13. The release of radionuclides from fuel pins which are predicted to experience clad failure is based upon core average source terms increased by the pre-accident radial peaking factors at which the individual fuel pins were operating.
14. Even for identical core average and hot channel conditions for a given transient event, the number of fuel pins that experience DNB, will vary from cycle to cycle. The calculated amount of fuel failure is highly sensitive to fuel loading pattern (i.e., pin power distribution). Therefore, it would be difficult to bound the calculated fuel failure of all future reloads based upon any one transient response. Instead, the amount of fuel damage (and associated RCS activity) that resulted in offsite doses approaching their acceptance criteria (i.e., 10 CFR Part 100) was back calculated for each event. For future reloads, a cycle-specific analysis will ensure that the amount of fuel failure (based upon actual fuel loading) does not exceed this upper limit.
15. The amount of controlled steaming necessary to remove decay heat, RCP heat, and cooldown the RCS (to SCS entry conditions) is approximately 2,550,000 lb_m. Employing the maximum RCS cooldown rate of 100 °F/hr results in approximately 1,000,000 lb_m of this total being released in the first two hours following the event.
16. The ANS/ANSI 5.1-1979 (Reference 6-36) decay heat curve with two-sigma uncertainty and representing heavy element decay and fission product neutron capture is used to calculate the decay heat.

Table 6.4-1
Core Isotopic Inventory at 3990 MW_t

Isotopes	Ci
I-131	1.02E+08
I-132	1.55E+08
I-133	2.29E+08
I-134	2.68E+08
I-135	2.08E+08
Kr-83m	1.69E+07
Kr-85	1.79E+06
Kr-85m	5.28E+07
Kr-87	8.77E+07
Kr-88	1.30E+07
Kr-89	1.69E+08
Xe-131m	1.06E+06
Xe-133m	5.63E+06
Xe-133	2.29E+08
Xe-135m	7.39E+07
Xe-135	2.18E+08
Xe-137	2.17E+08
Xe-138	2.02E+08

Section 6.4.1 Radiological Consequences of Increase in Heat Removal by the Secondary System

Section 6.4.1.1 Radiological Consequences of Inadvertent Opening of a Steam Generator Relief or Safety Valve

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described in Section 6.4.0. Although the results of the IOSGADV + LOP (UFSAR Section 15.1.4) transient simulation demonstrated that no fuel pin failures occurred (i.e., minimum DNBR remained above the SAFDL limit), the offsite dose calculation assumed 5.5% fuel failure to bound future fuel cycles. The NSSS response to this transient is detailed in Section 6.3.1.4.

Section 6.4.1.1.1 Change in Method of Evaluation

The previous dose calculation for this event assumed an iodine DF of 10 for releases from the unaffected SG. For the IOSGADV + LOP event, the conditions experienced within the unaffected SG are similar to those experienced during other UFSAR Chapter 15 transients. Specifically, SG mass inventory is maintained throughout the event. The doses presented for this event are based on a DF of 100 from the unaffected SG.

As an event initiator, an ADV opens as described in UFSAR Section 15.1.4. The release path for iodine and noble gas activity consists of uncontrolled steaming through the fully open ADV (in the affected SG) and controlled steaming through the ADVs in the unaffected SG. Due to a LOP, the condenser is unavailable and MSSVs and ADVs in the unaffected SG are employed to remove decay heat and cooldown the RCS. The affected SG empties and all RCS leakage into that SG is released directly to the atmosphere.

At 30 minutes, the operators are credited with closing the ADV (on the affected SG) and initiating a controlled cooldown (to SCS entry conditions) using the unaffected SG. During the controlled cooldown, the operators will maintain the appropriate subcooled margin in the RCS, and the appropriate SG level (40-60% NR in the affected SG).

The calculated EAB and LPZ doses are listed in Table 6.4-2. The doses remained below the acceptance criteria for this category of event, a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body). These calculated doses bound both the existing plant configuration and PUR.

Table 6.4-2
Radiological Consequences of IOSGADV + LOP

RCS Activity	Thyroid Dose (REM)		Whole Body Dose (REM)	
	2 hour EAB	8 hour LPZ	2 hour EAB	8 hour LPZ
Fuel Failure (5.5%)	28.0	13.2	0.9	1.0

Section 6.4.1.2 Radiological Consequences of Main Steam Line Break Outside Containment with a Concurrent Loss of Power

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described in Section 6.4.0. The RCS iodine activity was based upon three sources:

1. GIS,
2. PIS, and
3. assumed 1.0% fuel failure.

The NSSS response to this transient is detailed in Section 6.3.1.5.

The bounding MSLB scenario for offsite doses is MSLB outside containment, upstream of the MSIV, initiated at full power with a coincident LOP. Due to larger instrument uncertainties, the inside containment break event may yield more fuel failure than the outside containment break events. However, due to the limited release path for an inside containment break, radiological consequences (even with larger amounts of fuel failure) are bounded by outside containment breaks.

The iodine and noble gas release path for an outside containment MSLB consists of uncontrolled steaming through the affected SG and controlled steaming through the ADVs in the unaffected SG. Before MSIV closure, both SGs experience uncontrolled blowdown through the faulted steam line.

Due to a LOP, the condenser is unavailable and MSSVs and ADVs in the unaffected SG are employed to remove decay heat and cooldown the RCS. The affected SG empties and all RCS leakage into that SG is released directly to the atmosphere.

At 30 minutes, the operators initiate a controlled cooldown (to SCS entry conditions) using the unaffected SG. During the controlled cooldown portion of the event, the operators will maintain the subcooled margin in the RCS and the appropriate SG level (40-60% NR in the affected SG).

Table 6.4-3 lists the calculated EAB and LPZ doses. The doses remained below the acceptance criteria for this category of event - PIS or fuel failure (WRSO) : within 10 CFR Part 100 guidelines (i.e., 300 REM thyroid), GIS: small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid). These calculated doses bound both the existing plant configuration and PUR.

Table 6.4-3
Radiological Consequences of MSLB Outside Containment with a LOP

RCS Activity	Thyroid Dose (REM)		Whole Body Dose (REM)	
	2 hour EAB	8 hour LPZ	2 hour EAB	8 hour LPZ
GIS	3.0	5.5	0.1	0.1
PIS	2.5	1.5	0.1	0.1
Fuel Failure (1%)	20	20	0.5	0.5

Section 6.4.2 Radiological Consequences of Decrease in Heat Removal By The Secondary System

Section 6.4.2.1 Radiological Consequences of Feedwater System Pipe Breaks

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described above. The RCS iodine activity was limited to the Technical Specification value of 1.0 μCi/gm. No fuel failure is predicted to occur during the FWLB events. The NSSS response to this transient is detailed in Section 6.3.2.8.

The bounding FWLB scenario for offsite doses is a FWLB outside containment, downstream of the FWIV, and initiated at full power with a coincident LOP. The FW system design has two FW check valves that are located inside containment. These valves will prevent the blowdown of the affected SG for an outside containment break. For conservatism, the dose calculation ignores the presence of these two check valves and empties the affected SG directly into the atmosphere.

As an event initiator, a FWLB occurs outside containment. The release path for iodine and noble gas activity consists of uncontrolled blowdown of the affected SG and controlled steaming through the ADVs in the unaffected SG. Before FWIV/MSIV closure, both SGs experience uncontrolled blowdown though the faulted FW line.

Due to a LOP, the condenser is unavailable and MSSVs and ADVs in the unaffected SG are employed to remove decay heat and cooldown the RCS. The affected SG empties and all RCS leakage into that SG is released directly to the atmosphere.

At 30 minutes, the operators initiate a controlled cooldown (to SCS entry conditions) using the unaffected SG. During the controlled cooldown portion of the event, the operators will maintain the appropriate subcooled margin in the RCS, and the appropriate SG level (40-60% NR).

The calculated EAB and LPZ doses are listed in Table 6.4-4. The doses remained below the acceptance criteria for this category of event, small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body). These calculated doses bound both the existing plant configuration and PUR.

Table 6.4-4
Radiological Consequences of FWLB Outside Containment with a LOP and No Fuel Failure

RCS Activity	Thyroid Dose (REM)		Whole Body Dose (REM)	
	2 hour EAB	8 hour LPZ	2 hour EAB	8 hour LPZ
RCS at 1μCi/cc	2.5	1.0	0.1	0.1

Section 6.4.3 Radiological Consequences of Decrease in Reactor Coolant Flowrate

Section 6.4.3.1 Radiological Consequences of Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described in Section 6.4.0. The offsite dose calculation assumed 20% fuel failure to bound future fuel cycles. The NSSS response to this transient is detailed in Section 6.3.3.4.

Following the reactor trip, an operator opens a single ADV and it is assumed to stick fully open. The release path for iodine and noble gas activity consists of uncontrolled steaming through the fully open ADV (in the affected SG) and controlled steaming through the ADVs in the unaffected SG. Due to a LOP, the condenser is unavailable and MSSVs and ADVs in the unaffected SG are employed to remove decay heat and cooldown the RCS. The affected SG empties and all RCS leakage into that SG is released directly to the atmosphere.

At 30 minutes, the operators initiate a controlled cooldown (to SCS entry conditions) using the unaffected SG. During the controlled cooldown portion of the event, the operators will maintain the appropriate subcooled margin in the RCS, and the appropriate SG level (40-60% NR in the unaffected SG).

Section 6.4.3.1.1 Change in Method of Evaluation

The previous dose calculation for this event assumed no operator action to minimize offsite doses for the entire duration of the event. This assumption allows the affected SG to remain dry and continue releasing activity directly to the atmosphere (through the stuck open ADV). This dose calculation credits operators with re-establishing level in the affected SG, similar to the SGTRLOP single failure event. At 30 minutes, the operators begin re-filling the affected SG, covering the top of the tubes at 90 minutes. A DF of 100 is credited for all releases from the affected SG once level is re-established at 90 minutes.

The calculated EAB and LPZ doses are listed in Table 6.4-5. The doses remained below the acceptance criteria for this category of event, within 10 CFR Part 100 guidelines (i.e., 300 REM thyroid, 25 REM whole body). These calculated doses bound both the existing plant configuration and PUR.

Table 6.4-5
Radiological Consequences of Single RCP Sheared Shaft with a LOP

RCS Activity	Thyroid Dose (REM)		Whole Body Dose (REM)	
	2 hour EAB	8 hour LPZ	2 hour EAB	8 hour LPZ
Fuel Failure (20%)	260	92	3.1	3.5

Section 6.4.4 Radiological Consequences of Reactivity and Power Distribution Anomalies

Section 6.4.4.1 Radiological Consequences of Control Element Assembly Ejection

Methodology and regulatory criteria applicable for evaluating the radiological consequences of a CEA ejection accident are identical to those described in UFSAR Section 15.4.8. Assumptions are consistent with NRC Regulatory Guide 1.77 and SRP

Section 15.4.8. Dose contributions from containment leakage, containment power access purge, fluid leakage from ESF components, secondary releases from primary to secondary leakage and initial secondary system inventory releases are evaluated. The NSSS response to this transient is detailed in Section 6.3.4.8.

The activity in the fuel-clad gap is composed of 10% of the iodines and 10% of the noble gases of maximum fuel inventory. For rods where DNB is postulated, all of the fuel gap activity is modeled to be instantaneously available for release from containment leakage. The activity released to the containment is assumed to be mixed instantaneously and uniformly throughout the containment volume.

Analysis parameters have been reviewed for impacts and are consistent with those shown in UFSAR Table 15.4.8-6. Differences include:

1. core gas gap inventories are increased to reflect increased core power,
2. RCS mass is increased to reflect the larger SG configuration,
3. secondary steam release within 30 minute is increased,
4. a value that bounds PUR effect is used for the integrated RCS mass release in the power access purge evaluation, and
5. percent of fuel failure evaluated on a cycle-by-cycle basis.

The reload design process verifies that the consequences of the event are limiting for all core designs. Allowable doses for CEA ejection events with PUR remain within the values presented in UFSAR section 15.4.8. Total offsite doses from all sources remain within the applicable limits of 10 CFR part 100. Therefore, there are no changes to the consequences of the CEA ejection event for operation at PUR.

Section 6.4.5 Radiological Consequences of Increase in Reactor Coolant System Inventory

No event in this category was reanalyzed for radiological consequences.

Section 6.4.6 Radiological Consequences of Decrease in Reactor Coolant System Inventory

Section 6.4.6.1 Radiological Consequences of Double - Ended Break of a Letdown Line Outside Containment

Methodology and regulatory criteria for evaluating the radiological consequences of a DBLLOCUS are identical to those described in UFSAR Section 15.6.2.3.2. Assumptions are consistent with SRP Section 15.6.2. The event is indicated by several alarms listed in UFSAR Table 15.6.2-1, and the reactor operator isolates the break within ten minutes following initiation of the event. The NSSS response to this transient is detailed in Section 6.3.6.2.

The integrated mass release with PUR is not impacted relative to the value currently reported in the UFSAR. As reported in Section 7.6, the existing design source term is bounding. An iodine activity spike with a spiking factor of 500 is assumed to occur coincident with the initiation of the transient. The blowdown DF is assumed 3, and no credit is taken for the retention within the auxiliary building and filtration system.

Analysis parameters have been reviewed for impacts and are consistent with or bounded by those discussed in the UFSAR. The total 2-hour EAB thyroid dose remains bounded by existing UFSAR analysis and it continues to be within the SRP Section 15.6.2 criteria. The consequences of this event are within small fraction of (less than 10%) of applicable limits from 10 CFR Part 100.

Section 6.4.6.2 Radiological Consequences of Steam Generator Tube Rupture

Section 6.4.6.2.1 Radiological Consequences of Steam Generator Tube Rupture with a Concurrent Loss of Power and Single Failure

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described in Section 6.4.0. The dose calculation employed dynamic inputs from the CENTS transient simulation. The NSSS response to this transient is detailed in Section 6.3.6.3.

The evaluation of the radiological consequences of a SGTR assumes a complete severance of a single SG tube while the reactor is operating at full power (UFSAR Section 15.6.3.2.3). The event causes an increase in contamination of the secondary system due to RCS leakage through the affected tube. A manual reactor trip is initiated at 100 seconds that automatically trips the turbine. A LOP occurs 3 seconds after the turbine trip.

The SG pressure will increase rapidly, resulting in steam/activity release through the MSSVs. Venting from the affected SG continues until the secondary system pressure is below the MSSV blowdown pressure. Two minutes after the trip, a conservative assumption is made that the operator opens the ADVs on both SGs to prevent further challenges to the MSSVs and to stabilize RCS temperatures, the ADV on the affected SG opens fully. In order to reduce the radiological releases, the operator takes appropriate actions to recover the U-tubes of the affected SG. Actions assumed in this analysis included overriding the automatic isolation of AFW flow to the affected SG and diverting the flow to the affected SG.

The analysis of the radiological consequences of a SGTR considers the most severe release of secondary activity as well as primary system activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment is a function of primary-to-secondary coolant leakage rate, RCS activity, and the steam mass discharged to the environment.

In addition to the inputs listed above, the following assumptions and parameters were used to determine the activity releases and offsite doses:

1. The portion of the leaking primary fluid that flashes to steam upon entering the SG, was calculated based upon enthalpy of the specific leak. During periods of tube uncovery, the flashing fraction is set to 1.0.
2. The portion of the leaking primary fluid that flashes, is assumed to be released to the atmosphere with a DF of 1.0.
3. The unflashed portion of the leaking primary fluid mixes with the SG inventory and is released with a DF of 100.
4. The primary to secondary leak into the unaffected SG mixes with the inventory and is released with a DF of 100.
5. Transient primary specific activity was calculated using the dilution from HPSI injection.

The calculated EAB and LPZ doses are listed in Table 6.4-6. The doses remained below the acceptance criteria for this category of event, within 10 CFR Part 100 guidelines (i.e., 300 REM thyroid).

Table 6.4-6
Radiological Consequences of SGTR With LOP and Single Failure of an ADV

Plant Configuration	GIS (REM, thyroid)		PIS (REM, thyroid)	
	2 hour EAB	8 hour LPZ	2 hour EAB	8 hour LPZ
3876 MW_t	153	114	268	83
3990 MW_t	182	125	294	91

Section 6.4.6.2.2 Radiological Consequences of Steam Generator Tube Rupture with a Concurrent Loss of Power

EAB and LPZ offsite radiological consequences were calculated using the methods and inputs described in Section 6.4.0. The dose calculation employed dynamic inputs from the CENTS transient simulation. The NSSS response to this transient is detailed in Section 6.3.6.3.

The evaluation of the radiological consequences of a postulated SGTR assumes a complete severance of a single SG tube while the reactor is operating at full power. The accident results in an increase in contamination of the secondary system due to RCS leakage through the affected tube. A reactor trip occurs on hot leg saturation approximately 12 minutes after the event initiation. The reactor trip automatically trips the turbine, and a LOP occurs 3 seconds after the turbine trip.

The SG pressure will increase rapidly, resulting in steam/activity release through the MSSVs. Venting from the affected SG continues until the secondary system pressure is below the MSSV blowdown pressure. Two minutes after the trip, the analysis assumes opening the ADVs on both SGs to prevent further challenges to the MSSVs and to stabilize RCS temperatures. The steaming through both the affected SG and unaffected SG continues until the affected SG is isolated upon reaching target isolation temperature. After isolation of the affected SG, feeding and steaming the unaffected SG achieves cooldown to SCS entry conditions.

The analysis of the radiological consequences of a SGTR considers the most severe release of secondary activity as well as primary system activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment is a function of primary-to-secondary coolant leakage rate, RCS activity, and the steam mass discharged to the environment.

The assumptions and parameters used for determination of activity releases and offsite doses for the SGTRLOP case are same as those employed for SGTRLOP single failure. The only exception is that the portion of leakage that flashes does not consider tube coverage. The SGTRLOP transient may experience a brief uncovery of the top of the U-tube bundle following reactor/turbine trip. In general, this transient exhibits increasing SG liquid mass inventory (due to the RCS leakage into SG). The portion of primary fluid that flashes is calculated based upon the enthalpy of the leak for the duration of the transient.

The calculated EAB and LPZ doses for PUR are less then 20 REM for GIS/PIS. UFSAR Section 15.6.2 provides existing accident results. The doses remained below the acceptance criteria for this category of event - PIS: within 10 CFR Part 100 guidelines (i.e., 300 REM thyroid), GIS: small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid).

Section 6.4.6.3 Radiological Consequences of Loss-of-Coolant Accidents

The NSSS response to this transient is detailed in Section 6.1.

Section 6.4.6.3.1 Radiological Consequences of Small Break Loss-of-Coolant Accident

The methodology and the regulatory criteria applicable to the evaluation of SBLOCA radiological consequences are identical to those described in UFSAR Section 15.6.5.2. Assumptions consistent with NRC Regulatory Guides 1.4 and 1.77 are employed along with guidelines from SRP Section 15.6.5. Dose contributions from containment leakage, the power access purge, ESF containment sump leakage, secondary releases from primary to secondary leakage, and initial secondary system inventory release are evaluated.

For all SBLOCA transients, containment isolation occurs before core uncovery precluding the potential for large radioactive releases due to core uncovery related fuel cladding damage. The 0.02 ft^2 break was determined to be the smallest break that would exhibit core uncovery, and is the limiting small break relative to dose consequences for the PUR configuration.

Analysis parameters have been reviewed for impacts and are consistent with those shown in UFSAR Table 15.6.5-2. Differences include:

1. core gas gap inventories are increased to reflect increased core power,
2. the 0.01 ft^2 break does not uncover the core, and
3. RCS mass and volume are increased to reflect the new SG configuration.

Total offsite doses from all sources continue to be within the applicable limits of 10 CFR Part 100 and are within the consequences of the LBLOCA.

Section 6.4.6.3.2 Radiological Consequences of Large Break Loss-of-Coolant Accident

Methodology and regulatory criteria applicable for evaluating the radiological consequences of a LBLOCA are identical to those described in UFSAR Section 15.6.5.6. Assumptions consistent with Regulatory Guide 1.4 and SRP Section 15.6.5 are employed. Dose contributions from containment leakage, the power access purge, leakage via the depressurized secondary system, ESF containment sump leakage, and back leakage to the RWT are evaluated. Control room doses also include shine from the containment, accumulation on ESF signal filters, and direct cloud doses.

Analysis parameters have been reviewed for impacts and are consistent with those shown in UFSAR Table 15.6.5-2. Differences include:

1. core source term inventories are increased to reflect increased core power and
2. bounding maximum values for containment pressure and temperature are used for power access purge evaluations.

Table 6.4-7 lists the assumptions and parameters used for the evaluations. Table 6.4-8 lists the radiological consequence of LBLOCA radiological analysis. Total offsite doses and control room doses from all sources continue to be within the allowable limits from 10 CFR Part 100 and GDC 19. Control room habitability is discussed in Section 9.9.

Table 6.4-7
Assumptions and Parameters Used for LBLOCA Radiological Analysis

Parameter	Value
Source Term Data:	
1. Core Activity (2) (curies)	Table 6.4-1
2. RCS specific activity concentration before event:	
a. Noble gases	Section 7.6.2
b. Iodines (DEQ I-131), (μCi/gm)	60
3. RCS weight (at 70 °F), (lb_m)	571,776 (1)
4. Iodine composition:	
Elemental, Organic, Particulate	91%, 4%, 5%
Containment Data:	
1. Containment Net Free Volume, (ft^3)	2.62E+06
2. Initial Pressure, (psia)	16.7
3. Initial Temperature, (°F)	120
4. Containment pressure and temperature profile (2)	Section 6.2.2
Power Access Purge Model:	
1. Purge valve type	Butterfly
2. Purge valve size, (inch)	8
3. Number of release flow paths	2
4. Containment power access purge isolation time [duration of release], (sec)	12
5. Percent of RCS activity released to containment during the first 12 seconds.	100%
Containment Model:	
1. Source Terms (fraction of RCS activity released to containment atmosphere):	
a. Iodines	0.25
b. Noble gases	1.00
2. Containment leak rate, vol. (%/day):	
0-24 hr	0.1
>24 hr	0.05
3. Containment leak rate through the depressurized secondary system, (scfm)	0.9
4. Duration of containment leakage, (days)	30
5. Containment region net volumes, (ft^3):	
Main spray region	2.27E+06
Auxiliary spray region	0.20E+06
Unsprayed region	0.15E+06
6. Transfer rate between sprayed/unsprayed regions, (unsprayed volume change per hour)	3.3
7. Air transfer rates between the containment regions, (cfm):	
a. Main sprayed and unsprayed regions	7,582
b. Auxiliary sprayed and unsprayed regions	668

Table 6.4-7
Assumptions and Parameters Used for LBLOCA Radiological Analysis

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Parameter	Value
Spray Iodine Removal Coefficients (hr^{-1}):	
1. Main sprayed region:	
Elemental	19.6
Organic	0.0
Particulate	0.32
2. Auxiliary sprayed region:	
Elemental	6.05
Organic	0.0
Particulate	0.09
3. Spray elemental-iodine decontamination factor coefficients	6.51
Plate out Removal Coefficient of Elemental Iodine (hr^{-1}):	
1. Main sprayed region	2.14
2. Auxiliary sprayed region	14.4
3. Unsprayed region	14.4
4. Elemental-iodine decontamination factor credited for plate out	93.4
ESF Recirculation Leakage Model:	
1. Sump activity of iodine, as a percent of total core activity	50%
2. Sump volume, (ft^3) (2)	7.0E+04
3. Recirculation start time, (minutes)	20
4. Total ESF component leakage rate, (ml/hr)	3000
5. Percent of iodine in the leaked water assumed to become volatile	10%
Essential ESF Filtration Units Model:	
1. Filter efficiency, 2 inch charcoal (iodine):	
Elemental	95%
Organic	95%
Particulate	95%
2. Flow, (scfm)	6000+/-10%
3. Duration of ESF leakage, (days)	30
RWT Backleakage Model:	
1. Partition coefficient of iodine in sump water backleakage to RWT	1000
2. RWT net volume, (scf)	1.15E+05
3. Fuel building volume, (scf)	7.45E+05
4. Maximum leakage from SI system to RWT, (gpm)	43
Control Room Data:	Section 9.9

Table 6.4-7
Assumptions and Parameters Used for LBLOCA Radiological Analysis

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Parameter	Value
Transport Data:	
1. EAB X/Q, 0-2 hr, sec/m^3	2.3E-04
2. LPZ X/Q, sec/m^3:	
0-8 hr	6.4E-05
8-24 hr	4.8E-05
24-96 hr	2.6E-05
96-720 hr	1.1E-05
Credit for depletion of the effluent plume of iodine due to deposition on the ground	Not Assumed
Credit for radiological decay in transit	Not Assumed
Dose Calculation Data:	
1. Thyroid Inhalation dose conversion factors (DCFs) (ICRP-30), REM/Ci	
I-131	1.08E+06
I-132	6.44E+06
I-133	1.80E+05
I-134	1.07E+03
I-135	3.13E+04
2. Immersion [Beta Skin and Whole-Body] DCFs	RG 1.109

Notes: (1) Conservatively existing RCS mass is used to maximize the consequences of this event. Value is used to calculate sump concentration. A larger mass results in a reduction in concentration of radioisotopes.

(2) Indicate parameters that are changed due to this submittal.

Table 6.4-8
Radiological Consequence of LBLOCA Radiological Analysis

Contributor	0-2 hr EAB, REM		30-day LPZ, REM	
	Thyroid	Whole Body	Thyroid	Whole Body
Power Access Purge	1.085	5.2E-04	0.302	1.46E-04
Containment leakage	38.77	2.518	95.19	1.899
Containment leakage via Depressurized Secondary System	19.18	1.245	47.13	0.9392
ESF Component leakage	1.567	6.992E-03	10.15	0.01211
RWT Back-Leakage (IE-91-56, Reference 6-32)	0.02392	1.138E-04	7.724	4.807E-03
Contribution from containment direct shine	N/A	nil	N/A	nil
Contribution from outside cloud to control room	N/A	nil	N/A	nil
Contribution from control room essential filtration system	N/A	nil	N/A	nil
Total @ 4070 MW_t (102% of 3990 MW_t)	60.63	3.77	160.50	2.86
Total per UFSAR Table 15.6.5-2	58.75	4.03	155.66	2.91

Section 6.4.7 Radiological Consequences of Radioactive Material Release from a Subsystem or Component

Section 6.4.7.1 Radiological Consequences of Waste Gas System Failure

Methodology and regulatory criteria for evaluating the radiological consequences of a waste gas system failure are identical to those described in UFSAR Section 15.7.1. Assumptions are consistent with NRC Regulatory Guide 1.24 (Reference 6-33) and SRP Section 15.7.1. The event is characterized as a rapid release of the contents of a single waste gas decay tank to the environment (puff model). It is postulated that the tank contains its maximum inventory and that no action is taken to mitigate the consequences of the event.

Analysis parameters have been reviewed for impacts and are consistent with or bounded by those shown in UFSAR Table 15.7.1-1. The design source term for the RCS is discussed in Section 7.6 of this submittal. The existing analysis source term is bounding for PUR. Similarly, the existing RCS mass reported in Table 15.7.1-1 is conservative relative to the Gas Waste Tank inventory postulated for release from the tank. The radiological consequences are less than 1% of 10 CFR Part 100 limits.

Section 6.4.7.2 Radiological Consequences of Postulated Radioactivity Release Due to Liquid Containing Tank Failure

Methodology and regulatory criteria for evaluating the radiological consequences of a liquid containing tank failure are identical to those described in UFSAR Sections 15.7.3 and 2.4.13.3. Assumptions are consistent with NUREG/CR-3332 (Reference 6-37) and SRP Section 15.7.3. The hypothetical event is characterized as a rapid release of the contents of a RWT to the environment. It is postulated that the tank contains its maximum inventory of 60 Ci per the Technical Requirements Manual (TRM) T3.10.200 and that no action is taken to mitigate the consequences of the event (Reference 6-3).

Analysis parameters have been reviewed for impacts and are consistent with or bounded by those discussed in UFSAR Section 2.4.13.3.2. The design source term for the RCS is discussed in Section 7.6 of this submittal. The existing analysis source term in the UFSAR is bounding for PUR. The most limiting spill in the perched zone that reaches the exclusion boundary will be below 1% of any Maximum Permissible Concentration in water (MPC_w) listed in 10 CFR Part 20, Appendix B, Table II as required SRP.

Section 6.4.7.3 Radiological Consequences of Fuel Handling Accidents

Methodology and regulatory criteria for evaluating the radiological consequences of a fuel handling accident for PUR are identical to those described in UFSAR Section 15.7.4.1.3. Assumptions are consistent with NRC Regulatory Guide 1.25 (Reference 6-38) and SRP Section 15.7.4. Calculational methods and assumptions described in Regulatory Guide 1.25 apply since the values for maximum fuel rod pressurization, peak LPD for the highest power assembly discharged, and maximum centerline operating fuel temperature for the highest power assembly are less than the corresponding values in Regulatory Guide 1.25.

Values for fission product gas gap fractions are consistent with, or conservative relative to those specified in Regulatory Guide 1.25. Consistent with guidelines of Regulatory Guide 1.25, a decay time of 72 hours after shutdown was used in the analysis.

Analysis parameters have been reviewed for impacts and are consistent with those shown in UFSAR Table 15.7.4-1. The source term is the same as that assumed for the existing analysis in the UFSAR. In conclusion, the radiological consequences of fuel handing accident in the containment and in the fuel building are not impacted by the PUR. The radiological consequences of a fuel handling accident inside and outside the containment are less than one-third of 10 CFR Part 100 limits as required by SRP 15.7.4.

Section 6.4.8 Radiological Consequences of Limiting Infrequent Events

EAB and LPZ offsite radiological consequences for limiting infrequent event were calculated using the methods and inputs described above. The offsite dose calculation assumed 10% fuel failure to bound future fuel cycles. This dose calculation coincides with the transient detailed in Section 6.3.7.

The release path for iodine and noble gas activity consists of controlled steaming through the ADVs in both SGs. Due to a LOP, the condenser is unavailable and MSSVs and ADVs are employed to remove decay heat and cooldown the RCS.

At 30 minutes, the operators are credited with initiating a controlled cooldown (to SCS entry conditions). During the controlled cooldown portion of the event, the operators will maintain the appropriate subcooled margin in the RCS, and the appropriate SG level (40-60% NR in the unaffected SG).

The calculated EAB and LPZ doses are listed in Table 6.4-9, remain below the acceptance criteria for this category of event a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body). These calculated doses bound both the existing plant configuration and PUR.

Table 6.4-9
Radiological Consequence of Limiting Infrequent Event Radiological Analysis

RCS Activity	Thyroid Dose (REM)		Whole Body Dose (REM)	
	2 Hour EAB	8 Hour LPZ	2 Hour EAB	8 Hour LPZ
Fuel Failure (10%)	9.7	12.7	1.6	1.8

Section 6.5 Accident Source Term

LBLOCA source terms as result of the PUR core thermal power increase are addressed in this section. Non-LOCA accident source terms are described in the Section 7.6.

Source Terms for evaluating the radiological consequences of postulated DBAs (LBLOCA) have been developed in accordance with the recommendations of Regulatory Guide 1.4 (Reference 6-14), TID-14844 (Reference 6-16), and NUREG-0737 (Reference 6-25). With the exception of long-lived isotopes, (e.g., Kr-85, solids) the accident source term was generated using TID-14844 methodology. Because of relatively long half-life and high fuel burnup, the core inventory for long-lived isotopes was calculated using the code ORIGEN-S (SCAL 4.4 package). Activities for long-lived isotopes are conservatively based on assumed end of life at a cumulative value of 70,000 MWD/MTU fuel burnup and 5% enrichment.

Section 6.5.1 Large Break Loss-of-Coolant Accident Source Term

The accident source terms for LBLOCA are itemized in Table 6.5-1. The increase in source term using TID-14844 methodology is less than 3% for iodines and all noble gases with the exception of Kr-85 as compared to UFSAR Table 6.3.3.6-1. Kr-85 was calculated using the ORIGEN-S (APR-SCAL 4.4 library, Reference 6-39). The increase in Kr-85 is approximately 28%. This increase is due to use of bounding fuel cycle assumptions used in the ORIGEN-S model (e.g., maximum burn-up and fuel enrichment). Additional assumptions used in the analysis are:

1. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at the maximum core power level (102% of licensed power, 4070 MW_t, consistent with Regulatory Guide 1.49, Reference 6-15).
2. One hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from the containment.
3. Fifty percent of the core equilibrium radioactive iodine inventory is immediately released to the containment atmosphere. Half is plated out onto the internal surfaces of the containment and the other half is available for leakage from the containment.
4. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% in form of particulate iodine, and 4% is in the form of organic iodine.

Section 6.5.2 Other Accidents Source Term

The source term for other accident analyses are based on factors (e.g., % DNBR, failed fuel, etc.) that are specific to each event analyzed. The source terms have been developed and applied consistent with assumptions, guidelines, and criteria from Regulatory Guides and the SRP that govern the specific event under consideration.

Table 6.5-1 provides information for accident analyses for PUR. PUR results in a higher core inventory then those presently listing in UFSAR Table 6.3.3.6-1.

Table 6.5-1
Accident Source Terms for PUR

Isotope	Containment airborne T=0 min into LOCA (Ci)	Containment sump T=20 min into LOCA (Ci)
Kr-85	1.79E+06	-
Kr-85m	5.28E+07	-
Kr-87	8.77E+07	-
Kr-88	1.30E+08	-
Kr-89	1.69E+08	-
Kr-90	2.03E+08	-
Xe-131m	1.06E+06	-
Xe-133m	5.63E+06	-
Xe-133	2.29E+08	-
Xe-135m	7.39E+07	-
Xe-135	2.18E+08	-
Xe-137	2.17E+08	-
Xe-138	2.02E+08	-
Halogens	@ 25% of Core	@ 50% of Core
I-131	2.55E+07	5.11E+07
I-132	3.87E+07	7.75E+07
I-133	5.72E+07	1.14E+08
I-134	6.69E+07	1.34E+08
I-135	5.19E+07	1.04E+08

Section 6.6 References

Reference 6-1 Code of Federal Regulations, Title 10, Part 20 (old), Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure Effluent Concentrations; Concentrations for Release to Sewerage.”
Code of Federal Regulations, Title 10, Part 50, Section 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”
Code of Federal Regulations, Title 10, Part 50, Section 50.59, “Changes, Tests, and Experiments.”
Code of Federal Regulations, Title 10, Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants
Code of Federal Regulations, Title 10, Part 50, Appendix K, “ECCS Evaluation Models.”
Code of Federal Regulations, Title 10, Part 100, “Reactor Site Criteria.”

Reference 6-2 Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Technical Specifications, through Amendment No. 137, December 6, 2001 (Amendment No. 136 not yet implemented).

Reference 6-3 Technical Requirements Manual (TRM), for Palo Verde Nuclear Generating Station, Units 1, 2, 3, Revision 16, August 30, 2001.

Reference 6-4 CENPD-133P, “CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis,” August 1974.
CENPD-133P, Supplement 2, “CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications),” February 1975.
CENPD-133, Supplement 4-P, “CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis,” April 1977.
CENPD-133, Supplement 5-A, “CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis,” June 1985.

Reference 6-5 CENPD-135P, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” August 1974.
CENPD-135P, Supplement 2, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” February 1975.
CENPD-135, Supplement 4-P, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” August 1976.
CENPD-135-P, Supplement 5, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” April, 1977.

Reference 6-6 U. S. Nuclear Regulatory Commissions Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

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Reference 6-24 Lee, D. H., "An Experimental Investigation of Forced Convection Burn-out in High Pressure Water-Part IV, Large Diameter Tubes at About 1600 psia," A. E. E. W. Report R479, 1966.

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Reference 6-29 Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

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Initiatives Addressing Station Blackout at Light Water Reactors, August 1991."

Reference 6-32 NRC Information Notice 91-56, Potential Radioactive Leakage to Tank Vented to Atmosphere, September 19, 1991.

Reference 6-33 Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," Revision 0, March 23, 1972.

Reference 6-34 PCFLUD Computer Program Version 5.0 SQA Classification B, Bechtel Corporation.

Reference 6-35 ANS/ANSI-N58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions."

Reference 6-36 ANS/ANSI 5.1-1979, "Decay Heat Power in Light Water Reactors."

Reference 6-37 NUREG/CR 3332, "Radiological Assessment, A Textbook on Environmental Dose Analysis, Parts 1 and 2," dated September 1, 1983.

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Section 7 NUCLEAR FUEL

Section 7.1 Core Thermal-Hydraulic Design

This section describes the core thermal-hydraulic analyses performed in support of operation at an uprated core power of 3990 MW_t. The methods employed in these analyses are consistent with the methods used in the current reload analyses.

Section 7.1.1 Departure from Nucleate Boiling Analysis

Steady state Departure from Nucleate Boiling (DNB) analyses were performed using the following inputs:

- the TORC code described in Reference 7-1,
- the core inlet flow distribution model described in Reference 7-2.
- the CE-1 Critical Heat Flux (CHF) correlation described in Reference 7-3 and Reference 7-9, and
- the CETOP-D code described in Reference 7-4.

UFSAR Section 4.4 lists pertinent thermal-hydraulic parameters used for core thermal hydraulics design. The steady state DNB analysis using the methodology of Reference 7-5 determined that maintaining a TORC calculated Departure from Nucleate Boiling Ratio (DNBR) greater than or equal to 1.34 will provide assurance at the 95/95 probability/confidence level that the hot rod will not experience DNB. The 1.34 value includes penalties as the result of implementing Statistical Combination of Uncertainties (SCU) analysis discussed in Reference 7-6 (i.e., TORC code uncertainty and CE-1 CHF correlation cross validation uncertainty, as discussed in Reference 7-7). Other penalties included in the DNBR limit of 1.34 are a 0.01 DNBR penalty for HID-1 grids (discussed in Reference 7-7) and the rod bow penalty (discussed in Section 7.1.2).

The results of TORC DNBR analysis are used in the Modified Statistical Combination of Uncertainties (MSCU) analysis to derive overall uncertainty penalty factors for Core Protection Calculators (CPC) and Core Operating Limit Supervisory System (COLSS). This assures that at the 95/95-probability/confidence level, the hot rod will not experience DNB.

Section 7.1.2 Effects of Fuel Rod Bowing on Departure from Nucleate Boiling Ratio Margin

Effects of fuel rod bowing on DNBR margin is incorporated in the safety analysis, COLSS, and Core Protection Calculator System (CPCS) setpoint analyses in the manner discussed in Reference 7-8. The penalty used in the steady state DNBR analysis, 1.75% DNBR, is valid for all bundles with a peak pin that is within 90% of the peak pin in the core at any time in life up to and including a burnup of 30,000 MWD/T. This penalty is included in the calculated DNBR limit of 1.34.

For assemblies with burnup greater than 30,000 MWD/T sufficient available margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Hence, the rod bow penalty (based in Reference 7-8 for 30,000 MWD/T) is applicable for all assembly burnups for a typical PUR cycle.

Section 7.2 Core Design

Neither fuel assembly design nor overall fuel management strategy need to be significantly changed, and will continue per the requirements of UFSAR Section 4.2. Table 7.2-1 lists some general core design characteristics for Cycle 9 compared to target values for PUR and shows no significant differences.

Table 7.2-1
Core Design Characteristics

Parameter	Unit 1 Cycle 9	Unit 2 Cycle 9	Unit 3 Cycle 9	Typical PUR Cycle
Design Cycle Length (EFPD)	498	515	499	515
Number of Fresh Fuel Assemblies	100	96	92	92 - 108
Core Average Enrichment (wt% U235)	4.24	4.20	4.23	4.25
Maximum Pin Enrichment (wt% U235)	4.40	4.40	4.40	4.40
Core Average Burnup (GWD/MTU)	37.90	37.35	37.04	38
Pin Peak Burnup (MWD/MTU)	57,111	55,754	57,817	<60,000
Burnable Neutron Absorber	Erbium	Erbium	Erbium	Erbium

Section 7.3 Fuel Rod Design and Performance

The purpose of this evaluation was to review the fuel rod design criteria to determine the acceptability of fuel rod design under PUR conditions. The parameters used in the fuel rod design criteria evaluation for the PUR condition are summarized in Table 7.3-1.

Table 7.3-1
Summary of PUR Parameters Analyzed in Fuel Rod Design

Parameter	Units	3876 MW_t	3990 MW_t
Maximum Fuel Rod Axially Average Fluence (1)	n/cm^2	13.5E+21	13.5E+21
Core Inlet Temperature	°F	554	556.4
Minimum Flowrate	lb_m/hr	151.1E+06	151.1E+06
System Pressure	psia	2250	2250
Peak Rod Axial Average Burnup (2)	MWD/MTU	60,000	60,000
Residence Time	EFPH	48,500	51,000

Notes: (1) Bounding numbers for AOR using 4200 MW_t.
(2) Licensing limit for burnup. Some calculations conservatively used higher burnups as analysis input.

The following sections summarize the impact of the proposed PUR conditions on key fuel rod design criteria and the corresponding acceptance limits, and provide an assessment of the resulting impact on anticipated design margin based on the acceptance limits. The key criteria include rod-cladding collapse, clad fatigue, clad stress and strain, rod maximum internal pressure, and clad corrosion.

Section 7.3.1 Rod Cladding Collapse

Margin to the fuel rod clad collapse limit is impacted by changes in the core power rating because higher power levels result in higher fuel operating temperatures resulting in an increase in oxide thickness levels. The NRC approved collapse performance methodology and computer program CEPAN (Reference 7-10 and Reference 7-11) were used to evaluate rod collapse as a function of residence time. The results of this evaluation confirmed that rod collapse limits are satisfied for the assumed residence time under PUR conditions.

Section 7.3.2 Clad Fatigue

Margin to the fuel rod clad fatigue limit is impacted by changes in the core power rating because higher power levels result in higher fuel operating temperatures resulting in an increase in cyclic strain levels. The clad fatigue performance was evaluated for rod fatigue as a function of burnup. The results of this evaluation confirmed that rod fatigue limits can be satisfied for the EOL burnup listed in Table 7.3-1.

Section 7.3.3 Clad Stress and Strain

The local power duty during AOO events is a key factor in evaluating margin to clad stresses and strain limits. The results of the evaluation show that the PUR will not affect the fuel's capability to meet clad stress and strain limits. The fuel clad stress and strain during a fuel handling accident is bounded by the existing Analyses of Record (AOR).

Section 7.3.4 Rod Maximum Internal Pressure

The thermal performance of Erbia and UO_2 composite fuel rods that envelope the expected fuel rods of the fuel batches have been evaluated using the FATES3B version of the Combustion Engineering fuel evaluation model (Reference 7-12 and Reference 7-13). A bounding analysis was performed for the PUR cycle using a power history that enveloped the power and burnup levels representative of the peak fuel assembly at each burnup interval, from Beginning of Cycle (BOC) to End of Cycle (EOC) burnups.

The maximum predicted fuel rod internal pressure for the PUR cycle is below the critical pressure for no clad liftoff given in Reference 7-14.

Section 7.3.5 Cladding Waterside Corrosion

In accordance with Reference 7-15, an evaluation of waterside corrosion of fuel was completed under PUR conditions. Fuel management applied for the typical PUR cycle will assure that maximum oxide thickness levels and fuel duty are bounded for PUR conditions, based on criteria in Reference 7-15 and Reference 7-16. Therefore, the impact on thermal and mechanical performance will be acceptable for operation at PUR conditions.

Section 7.3.6 Conclusions

The fuel rod design criteria impacted by a change in core power rating have been reviewed with respect to the available margin to support the PUR. Although some design criteria are impacted, as stated above, the PUR conditions listed in Table 7.3-1 are supported. Typical PUR cycle-specific fuel performance will continue to be evaluated for each fuel cycle to confirm that all fuel rod design criteria are satisfied for the operating conditions specified for each cycle of operation. These evaluations support the Reload Safety Evaluation (RSE) that is performed for each typical PUR cycle of operation.

Section 7.4 Heat Generation Rates

The heating rates for various reactor internal components were previously evaluated for the 3876 MW_t conditions. The heating rates experienced by the various components have been evaluated for applicability of that analysis to the proposed PUR operation (3990 MW_t).

For each component, the allowed relative power densities for the assemblies affecting the component heating rate were recalculated to assure the validity of the AOR heating rates for the PUR operation. The calculated allowed relative power densities are specified as fuel management constraints for PUR reload core designs. The components included in this evaluation are the core shroud, core support barrel, surveillance capsule holders, vessel, lower support structure, fuel alignment plate, and upper guide structure components.

The maximum temperatures and internal heat generation rates are used as input to the structural integrity analyses of Reactor Vessel Internal (RVI) components in Section 5.2.3.

Section 7.5 Neutron Fluence

The calculated fluences for the existing AOR assume a core power level of 4200 MW_t and an out-in type fuel-loading pattern typical of first cycle operation. The PUR level of 3990 MW_t and the low-leakage fuel patterns (used since PVNGS Unit 2 Cycle 2) yield a neutron flux to the shroud and vessel that is lower than considered in the existing AOR (see Section 5.1.2). Therefore, the reactor vessel integrity AOR (i.e., pressure/temperature limits and Pressurized Thermal Shock (PTS) screening limits) are not affected by PUR operation at 3990 MW_t. In addition , fuel management guidelines for PUR cycles are set to ensure that the vessel fluence is bounded by the AOR.

Section 7.6 Source Terms

Section 7.6.1 Expected Source Term

The radioactive expected (normal with nominal failed fuel)and design source terms were evaluated for the increase in core-licensed power from 3876 to 3990 MW_t.

Expected source terms were evaluated using methodology from ANSI N237/ANS-18.1 (Reference 7-18) and NUREG-0017 (Reference 7-19). Three analysis parameters are affected by the PUR:

- Reactor Coolant System (RCS) mass,
- Chemical and Volume Control System (CVCS) purification constant (λ_{CVCS}), and
- core power.

The increased core power results in a proportional increase in the inventory of fission product isotopes in the RCS. On the other hand, the larger volume of SGs reduces the specific activities in the RCS. The effects from decrease of λ_{CVCS} (defined as the ratio of CVCS letdown flowrate to RCS volume) for different isotopes are shown in Table 7.6-1. Table 7.6-2 provides a summary of differences between parameters used in support of this submittal and the existing UFSAR. It is concluded that the net effect of these changes does not alter the expected (normal) source term as presented in the UFSAR Section 11.1 (Reference 7-17).

Table 7.6-1
PUR Impacts on Expected RCS Specific Activities

Isotopes	Estimated Impact (1)	Primary Influencing Factor(s)
Noble Gases (Except Kr-85)	Decrease by 6-10%	Power and RCS mass (small half lives)
Kr-85	Increase by ~ 3%	Power, (long half lives)
Halogens (Except I-131)	Decrease by 2-9%	Power, RCS mass (small half lives)
I-131	Increase by ~ 2%	Power, (long half lives)
Others	Increase by ~ 3%	Conservative Assumption

Note: (1) As compared to existing data from UFSAR Section 11.1.

Table 7.6-2
Summary of Input Parameters Used for Estimation of Normal Source Term

Parameter	Existing Plant Analysis	PUR	% Change
Thermal Power	3880 MW_t	3990 MW_t	2.8%
RCS Mass	571,776 lb_m	643,412 lb_m	12.5%
Letdown Flow	35,600 lb_m/hr	35,600 lb_m/hr	N/A
Purification constant	6.22E-02 1/hr	5.53E-02 1/hr	10% (max)

Section 7.6.2 Design Source Term (1% Failed Fuel Condition Equilibrium Activities)

Methodology and input parameters used to calculate design source term (1% failed fuel condition equilibrium activities) for fission products are described in UFSAR Section 11.1. Core power is the predominant input parameter and affected by the PUR. The existing core power analyzed and reported in the UFSAR is 4220 MW_t. This analyzed power results in bounding RCS specific activities relative to the proposed PUR of 4070 MW_t (3990 MW_t plus 2% calorimetric uncertainty, consistent with Regulatory Guide 1.49, Reference 7-20). There are no other impacts to fission product specific activities. Therefore, the fission product specific activities reported in UFSAR Section 11.1 remain conservative and bounding for PUR.

Section 7.6.3 Other Isotopic Source Terms

Design crud RCS specific activities are based on NUREG-0017 methodology. The values presented in UFSAR Section 11.1 are calculated for 4200 MW_t core power and are therefore conservative.

Other activation products (e.g., tritium, N-16, etc.) were also evaluated. Neutron flux and RCS loop transit time in the core are the primary parameter for calculation of production rates for these activation products. The N-16 activity rate at the reactor nozzle (as reported in UFSAR Section 11.1) is conservative and bounding for PUR. The relatively short half-life for N-16 combines with additional decay time during transit through the larger RCS loop to offset small increases in production due to increased core neutron flux. Existing production rates for tritium and carbon-14 (as reported in UFSAR Section 11.1) are also bounding and conservative for PUR. This is a result of conservative input values for core power and/or core flux in the existing AOR.

Section 7.6.4 Conclusions

In summary, the source term reported in UFSAR Section 11.1 remains bounding and conservative for PUR.

Section 7.7 References

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Reference 7-2 “Supplement 1-P to Enclosure 1-P to LD-82-054,” APS letter 102-02465/WFC/TRB/GAM, March 30, 1993.

Reference 7-3 CENPD-162-P-A, “Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution,” September 1976.

Reference 7-4 APS letter 161-01867-DBK/JRP, “Applicability of the RAR References,” April 26, 1989.

Reference 7-5 CEN-356(V)-P-A, Revision 01-P-A, “Modified Statistical Combination of Uncertainties,” May, 1988.

Reference 7-6 Enclosure 1-P to LD-054, “Statistical Combination of System Parameter Uncertainties in Thermal Margin Analyses for System 80,” submitted by letter from A. E. Scherer (C-E) to D. G. Eisenhut (NRC), May 14, 1982.

Reference 7-7 CESSAR SSER 2 Section 4.4.6, “Statistical Combination of Uncertainties (SCU),” September 1983.

Reference 7-8 CENPD-225-P-A, “Fuel and Poison Rod Bowing,” June 1983.

Reference 7-9 CENPD-207-P-A, “Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-uniform Axial Power Distribution,” December 1984.

Reference 7-10 "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," CE Report CENPD-187-A, March 1976.

Reference 7-11 "Software Verification and Validation Report - CEPANFL Version 996," V&V-FD-012, Revision 00, dated 1/16/97.

Reference 7-12 "C-E Fuel Evaluation Model Topical Report," Combustion Engineering, Inc., CENPD-139-P-A, July 1974.
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Supplement 1-P-A, "Improvements to Fuel Evaluation Model," CEN-161(B)-P, January 1992.

Reference 7-13 CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

Reference 7-14 CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

Reference 7-15 W CENP Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," ABB Combustion Engineering, Inc., August 1992.

Reference 7-16 W CENP Report CENPD-384-P, "Report on the Continued Applicability of 60 MWD/kgU for ABB Combustion Engineering PWR Fuel," ABB Combustion Engineering, Inc., September 1995.

Reference 7-17 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Reference 7-18 ANSI N237/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors."

Reference 7-19 NUREG-0017, Revision 1, Dated, 4/1/1985, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, PWR-Gale Code."

Reference 7-20 REGULATORY GUIDE 1.49, Revision 1, December 1973, "Power Levels of Nuclear Power Plants."

Section 8 BALANCE OF PLANT DESCRIPTION

Section 8.1 Balance of Plant Program Overview

A detailed Balance of Plant (BOP) review was performed to determine the plant system and component impacts related to increasing the licensed power from 3876 MW_t to 3990 MW_t. The BOP evaluations and analyses were coordinated with the Nuclear Steam Supply System (NSSS) system and component evaluations.

Section 8.2 Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) system and the Condensate Storage Tank (CST) are described in UFSAR Section 10.4.9 (Reference 8-1). The increase in licensed reactor power to 3990 MW_t does not affect the AFW system. Installation of the Replacement Steam Generators (RSGs) does have an impact on the AFW System hydraulics. This impact results from the higher elevation of the downcomer nozzle on the Steam Generator (SG) and the modified design of the SG downcomer feedring. The AFW system will remain capable of delivering a total required minimum flow of 650 gpm to the SG(s).

The primary source of AFW is the CST. There are no changes required to the condensate transfer system because of the PUR. The CST has a dedicated volume of 300,000 gallons available to support the AFW system for secondary side cooldown. The increased decay heat resulting from PUR affects the required condensate volume as follows:

- The required condensate volume calculated per NRC guidelines in Branch Technical Position (BTP) RSB 5-1 is increased due to PUR and RSGs. However, this increased volume remains below the existing required volume and is acceptable for operation at PUR conditions. Refer to Section 9.10 for additional discussion.
- The 10 CFR Part 50, Appendix R (Reference 8-2) safe shutdown analyses were revised to account for the increased decay heat. The revised analyses confirm that the dedicated 300,000-gallon CST volume is sufficient to support the Appendix R safe shutdown actions. Refer to Section 9.6 for additional discussion.
- The applicable safety analyses were revised and are acceptable for operation at PUR conditions as discussed in Section 6. The updated safety analyses considered the existing CST inventory of 300,000 gallons.

Section 8.3 Condensate and Feedwater

The Condensate (CD) and Feedwater (FW) systems are described in UFSAR Section 10.4.7.

Section 8.3.1 System Description

The CD and FW system is composed of a three shell/single division condenser (one shell for each low pressure turbine), three 50% CD pumps, three trains of low pressure FW heaters (four heaters per train), two variable speed turbine-driven main FW pumps, two heater drain pumps, and two trains of high pressure FW heaters (three heaters per train). 100% power operation can be achieved with two of the three CD pumps in service. 60% power operation can be achieved with one of the two FW pumps in service.

The CD and FW system was evaluated based on the 3990 MW_t (4013 MW_t total) PUR level and operation of the high pressure FW system in its original design configuration (full FW heating). Presently, operating procedures have been modified to bypass 25% of the high pressure FW flow around the high pressure FW heaters. Full FW heating will be resumed to regain steam turbine efficiency after the existing Steam Generators (SGs) are replaced. The system has been analyzed to show that acceptable performance will be achieved at the PUR increased flow rates.

In addition, the condenser air removal and condensate cleanup systems have been evaluated and found acceptable at PUR conditions.

Section 8.3.2 Condensate and Feedwater Pumps

The CD mass flowrate associated with PUR conditions will increase. Although the FW and CD flowrates exceed existing operating values, the PUR flowrates are bounded by the original Valves Wide Open (VWO) design condition. The FW and CD pumps were evaluated for PUR. The FW and CD pump parameters evaluated were:

- FW and CD pump total dynamic head and flow,
- FW and CD pump brake horsepower,
- FW and CD pump net positive suction head required and actual ($NPSH_R$ and $NPSH_A$),
- FW and CD pump minimum recirculation flow, and
- FW pump suction pressure alarm and pump trip setpoints.

Section 8.3.3 Heater Drain Pumps

The PUR extraction drain mass flowrate will increase. However, the increased flow is bounded by the VWO design mass flowrate. The heater drain pumps parameters evaluated were:

- total dynamic head and flow,
- $NPSH_R$ and $NPSH_A$,
- Discharge flow control valve margin, and
- pump brake horsepower.

Section 8.3.4 Low Pressure Feedwater Heaters

The PUR low pressure FW heater mass flowrates will increase. However, the increased flow is bounded by the VWO design flowrate. The low pressure FW heaters parameters evaluated were:

- FW nozzle velocity,
- drain outlet velocity,
- tubeside pressure drop,
- tubeside velocity,
- extraction steam pressure drop,
- extraction steam velocity,
- drain inlet velocity, and
- shell and tube side relief valve sizing.

Section 8.3.5 High Pressure Feedwater Heaters

The PUR high pressure FW heater mass flowrates will increase. The heater mass flowrates are bounded by the original VWO design flowrate. The high pressure FW heaters parameters evaluated were:

- FW nozzle velocity,
- drain outlet velocity,
- tubeside pressure drop,
- tubeside velocity,
- extraction steam nozzle velocity, and
- drain inlet velocity.

Section 8.4 Circulating Water

The circulating water system is described in UFSAR Section 10.4.5. The circulating water system is comprised of a three shell, triple pressure, single division condenser; three cooling towers; four circulating water pumps; and circulating water piping and valves. The circulating water system is a closed loop system that cools the main condenser. The heat from the power conversion process is rejected through three cooling towers. Water is added as needed by the cooling tower makeup system. The cooling towers continuously blowdown to the plant evaporation ponds to maintain chemistry specifications. Both the makeup and blowdown systems are sized to handle the increases in water associated with the PUR.

The circulating water system was reviewed to determine its performance at PUR conditions. The review included an evaluation of the steam turbine backpressure, cooling water makeup, and cooling tower emissions. A PVNGS model based on Heat Exchanger Institute (HEI) and Cooling Tower Institute (CTI) guidelines was used to predict condenser backpressure and circulating water temperature.

When operating at PUR conditions the main condenser heat load increases by 2%. This will result in an increase in condenser backpressure and circulating water temperature. Plant operating data has shown that when operating at current operating levels the condenser backpressure peaks at 4.75 inches Hg. during the high dew point season. Administrative controls are in place to reduce power generation and limit the condenser backpressure to less that 5.0 inches of Hg. The 2% increase in heat load due to PUR conditions will raise the condenser backpressure by 0.1 inches. All of the above parameters demonstrate acceptable performance after PUR.

The maximum calculated water temperature to the cooling towers will increase after PUR but will remain well below the cooling tower fill limiting temperature. Since the cooling tower fans are constant revolutions per minute (rpm) volume machines, operating at PUR conditions will not change their flowrate. The cooling tower makeup and blowdown rates will increase by 1%. This increase is within the makeup system capacity. The cooling tower salt emissions are evaluated in Section 9.8.

Section 8.5 Main Turbine

The main turbine system is described in UFSAR Section 10.2. The turbine evaluations were based on a best estimate SG operating pressure.

The turbine and associated components were reviewed to determine their performance at PUR levels. The review included a structural evaluation of the turbine components as well as system performance. The existing plant secondary side model was revised for PUR conditions to determine the performance requirements of the turbine. This information was then used to determine effect on the turbine and its associated components. The review concluded that no turbine nozzle, bucket, or piping changes are required to ensure thermodynamic adequacy for operations at PUR conditions. Turbine components that were reviewed include the following:

- high pressure 1st stage turbine buckets,
- low pressure bucket stall/flutter,
- shells, casings, and bolting at the increased operating temperatures and pressures,
- stationary steam path components at the increased operating temperatures and pressures,
- turbine bucket stresses for the increased stage outputs,
- rotors for increased torsional loads.
- couplings for increased torsional loads and output transmission
- rotors for dynamics and stability,
- increased thrust loads and potential thrust imbalances,
- increased journal bearing loads and temperatures,
- overspeed sensitivity and protection,
- turbine expansion and clearance at the increased temperatures and pressures,

- turbine piping and relief valves at increased operating temperatures, pressures and flowrates, and
- stop, control, and combined intermediate valves for the increased flows and pressures.

PUR does not change the recommended stall/flutter and high backpressure limits for the first stage low-pressure turbine blades. Susceptibility of low pressure discs to stress corrosion cracking and susceptibility for erosion of the last row blades were evaluated by comparing the low pressure steam temperature, and moisture levels at PUR conditions. The PUR low pressure steam temperature is the same as the existing operating conditions. Moisture levels will be lower after PUR. Therefore, PUR does not adversely affect the turbine's susceptibility to stress corrosion cracking or erosion. Since PUR does not raise the main steam operating pressure above the original design pressure, the turbine valves, piping, shells, casings, and bolting remains adequate for operations at PUR operating conditions.

The existing minimum overspeed setting is 110% and maximum overspeed setting is 111%. The turbine manufacturer has reviewed and concluded that, based on the PUR conditions, the existing overspeed settings remain acceptable.

A missile generation study was performed for the original turbine installation (UFSAR Section 3.5.1.3). A review of this study concluded the probability of missile generation is bounded for PUR conditions.

A recommendation is made by the turbine vendor to change the turbine control valve logic such that all four-turbine control valves open and regulate equally. Presently, three control valves open fully and the forth control valve regulates the flow to the turbine based on demand. This recommendation was made by the vendor to eliminate possibility of cyclical loading to the turbine. The change to the control valve logic will be implemented in support of the PUR implementation.

<u>Section 8.6</u> <u>Main Turbine Auxiliaries</u>

The main turbine auxiliaries are described in UFSAR Section 10.4. This evaluation assessed the adequacy of the moisture separator reheaters, gland steam seal system, gland steam seal relief valves, and gland steam seal exhauster for PUR conditions.

The moisture separator steam mass flowrate will increase after PUR. This is within the original VWO design parameters for the chevron flow separators and steam inlet/outlet nozzle velocity. Operating pressures will remain within the system design ratings. The gland steam seal system, gland steam seal relief valves, and gland steam seal exhauster components have been evaluated and found to be acceptable for operation at PUR conditions.

Section 8.7 Main Generator and Auxiliaries

The main generator and auxiliaries are described in UFSAR Sections 10.2 and 10.4. The performance of the following equipment and auxiliary systems was reviewed at the PUR conditions. Specifically:

- generator stator and rotor (generator capability curve, steady-state and transient stability, and winding vibration and phase imbalance),
- hydrogen gas cooling system,
- stator cooing water system,
- hydrogen seal oil system,
- Generex excitation system,
- excitation and rectifier cooling system,
- generator high voltage bushings, and
- existing transformers.

The review performed for these components evaluated the steady state and vibration stresses due to anticipated operational transients. Vibration stresses and natural frequencies were examined to assess dynamic stresses at PUR conditions. In addition, the review concluded that these components have adequate thermodynamic performance margin to meet their design basis after PUR. The evaluation shows that no modification to the existing components of the generator is required for the PUR conditions.

Section 8.8 Main Steam

This section describes the evaluation of the main steam system from the outlet of the SG nozzle to the turbine stop valves. Main steam piping stresses are addressed in Section 8.12 and environmental qualification of the safety-related electrical equipment is addressed in Section 9.4. The review of the main steam system includes the following:

- Main Steam Safety Valves (MSSV),
- Atmospheric Dump Valves (ADV),
- Main Steam Isolation Valves (MSIV),
- MSIV bypass valves,
- Turbine Bypass Control Valves (TBVs),
- Feedwater Isolation Valves (FWIVs), and
- main steam traps.

Section 8.8.1 Main Steam Safety Valves

The MSSVs are described in UFSAR Section 10.3.2.2.3. There are 20 MSSVs with 5 MSSVs on each main steam line. The total MSSV capacity is 22.56E+06 lb_m/hr and is greater then the total steam mass flow for PUR operation. In addition, since acceptable

MSSV performance has been demonstrated by the results found in Section 6 of this submittal, no changes to the MSSVs are required for PUR operation.

Section 8.8.2 Atmospheric Dump Valves

The ADVs are described in UFSAR Section 10.3.2.2.4. The ADVs maximum capacity is 1.47E+06 lb_m/hr at a saturation pressure of 1000 psia. These valves meet the performance requirements of the limiting UFSAR Chapter 15 events for PUR operation.

Section 8.8.3 Main Steam Isolation Valves

The MSIVs are described in UFSAR Section 10.3.2.2.2. The MSIVs have a maximum flow capacity of 4.545E+06 lb_m/hr. The PUR steam mass flowrate will remain below the maximum capacity of the MSIVs. The "worst case" condition (maximum differential pressure) for the MSIVs to operate is SG pressure at 0% power upstream and a Main Steam Line Break (MSLB) downstream of the valve. Neither the pressure upstream nor the postulated break downstream is changing from the original design conditions due to PUR. Therefore, the MSSVs' ability to fulfill their design function will not be affected. The valve design temperature and pressure envelop the system operating pressure for PUR. Therefore, the MSSVs will perform their design function under PUR conditions.

Section 8.8.4 Main Steam Isolation Valve Bypass Valves

The MSIV bypass valves are described in UFSAR Section 10.3.1. The MSIV bypass valves allow small amounts of steam to bypass the MSIVs during plant startups. When plant startup conditions require, using the MSIV bypass valves allows the piping downstream of the MSIVs to heatup before opening the MSIVs. The MSIV bypass valves were evaluated per EPRI requirements (Reference 8-8) and will be modified before implementation of this license amendment (see Section 9.1).

Section 8.8.5 Turbine Bypass Valves

The Turbine Bypass Valves (TBVs) are described in UFSAR Section 10.4.4. The eight TBVs mass flowrate is 10.88E+06 lb_m/hr (1.36E+06 lb_m/hr/valve) at 960 psia. The TBVs capacity remains greater than the desired flowrate (55% of the total steam flow) at PUR conditions. The operating pressures after PUR are bounded by the valve design requirement. Therefore, the valve stroking time is not adversely affected by PUR. The TBVs work in conjunction with the Steam Bypass Control System (SBCS) as described in UFSAR Section 10.4.4.

Section 8.8.6 Main Steam Traps

The main steam traps were evaluated based on a total steam mass flowrate increase after PUR. Although the PUR steam flowrate is increasing, the PUR steam quality is improving from a current design value of 0.25% to a design value of 0.1%. Therefore, based on the drier steam after PUR, the existing steam traps will perform as designed with the increased steam flows associated with PUR. The existing steam trap design temperature and pressure still bound the higher steam operating conditions associated with PUR.

Section 8.8.7 Feedwater Isolation Valves

The FWIVs are described in UFSAR Section 10.3. The FWIVs isolate the FW system from the SG if a MSLB, Feedwater Line Break (FWLB), or Loss-of-Coolant Accident (LOCA) occurs. The quick closure requirements of the valves can cause large magnitude dynamic pressure changes on the valves. The worst-case dynamic pressure condition on the valve occurs following a MSLB with both FW pumps in service. The FWIVs are designed to close against a pressure differential of 1875 psi. This pressure is greater than the FW pumps' maximum discharge pressure. The FWIV design bounds any potential change associated with PUR.

Section 8.8.8 Main Steam System Summary of Conclusions

After the modification of the MSIV bypass valves the plant main steam system and components will perform their design functions at the PUR operating conditions.

Section 8.9 Miscellaneous Cooling Water Systems

The miscellaneous cooling water systems discussed are described in UFSAR Section 9.2.

Section 8.9.1 Plant Cooling Water

The individual Plant Cooling Water (PW) component heat loads considered in the original design and as listed in UFSAR Table 9.2-31 remain bounding under PUR conditions.

Section 8.9.2 Turbine Cooling Water

The individual Turbine Cooling Water (TC) component heat loads considered in the original design and as listed in UFSAR Table 9.2-25 remain bounding under PUR conditions.

Section 8.9.3 Nuclear Cooling Water

The individual Nuclear Cooling Water (NC) component heat loads considered in the original design and as listed in UFSAR Table 9.2-8 remain bounding under PUR conditions.

Section 8.9.4 Essential Cooling Water

The Essential Cooling Water System (ECWS) is described in UFSAR Section 9.2.2. The ECWS:

1. In conjunction with the spray pond, provides heat removal to ensure a safe reactor shutdown coincident with a Loss of Offsite Power (LOP).
2. In conjunction with the Essential Spray Pond System (ESPS), prevents the outlet temperature of the ECWS heat exchanger from exceeding 135 °F during a postulated LOCA with a LOP.
3. In conjunction with the spray pond, provides cooling capability for the fuel pool when the fuel pool normal cooling system is not available.

The ECWS thermal performance analyses have been revised to account for increases in the decay heat and post-accident containment environmental heat resulting from the PUR. These analyses account for the environmental and equipment heat load from the essential chillers, and the spent fuel heat load from the Spent Fuel Pool Cooling and Cleanup (SFPCC) system. The individual component heat loads considered in the original design, as listed in UFSAR Table 9.2-1 have sufficient margin and bound operation for PUR.

The ECWS thermal performance evaluation was performed as an integral part of the ECWS and ESPS thermal performance analyses and is discussed in Section 8.16. The revised thermal performance analyses confirm that the ECWS will provide adequate cooling to both the SCS and the CSS operational modes of the Emergency Core Cooling System (ECCS), the essential chiller, and the SFPCC system following all postulated Design Basis Events (DBEs). The ECWS hydraulic performance is not affected by the PUR. In summary, the ECWS design and licensing basis bound the PUR.

Section 8.9.5 Spent Fuel Pool Cooling and Cleanup System

The SFPCC system is described in UFSAR Section 9.1.3. The SFPCC system provides:

1. spent fuel decay heat removal via the pool cooling heat exchanger and
2. spent fuel pool water quality and chemistry control via the cleanup portion of the system.

The maximum allowable spent fuel pool decay heat load is administratively controlled so that the heat load in the spent fuel pool is less than the available heat removal capability, considering single failure. The spent fuel pool heat load is analytically confirmed to be less than the available heat removal capacity before the return to power operation following a refueling outage.

While the PUR will result in an increase in the decay heat associated with an individual spent fuel assembly, increase in individual fuel assembly decay heat loads will not require modification to the existing administratively controlled maximum operating heat load limits on the spent fuel pool.

The maximum full core offload decay heat load placed into the spent fuel pool during an outage (scheduled or unscheduled) is increased, but remains below the available heat removal capability.

The SFPCC system heat removal capabilities are provided in UFSAR Table 9.1-2, and are greater than or equal to the allowable spent fuel pool heat loads. The non-safety related cleanup function of the SFPCC system is not affected by the PUR. Additionally, the SFPCC system hydraulic parameters are not affected by the PUR.

Section 8.10 Miscellaneous Mechanical Reviews

Section 8.10.1 Heating, Ventilation, and Air Conditioning Systems

The Heating, Ventilation, and Air Conditioning (HVAC) systems are described in UFSAR Section 9.4.

Section 8.10.1.1 Containment Heating, Ventilation, and Air Conditioning

The design basis for the containment heat load calculations considered a Reactor Coolant System (RCS) with T_{hot} of 621 ^{o}F. This T_{hot} value of 621 ^{o}F bounds the PUR T_{hot}. Therefore, the total heat load resulting from PUR will be less than the original design.

Section 8.10.1.2 Auxiliary Building Ventilation

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

The heat loads or other input parameters considered in the original Station Blackout (SBO) design remains bounding for PUR. The essential equipment rooms' original design environmental condition remains bounding for PUR.

Heat transfer due to fluid transport through ECCS piping was evaluated. The HA system heat loads have been revised to account for the new heat loads predicted to occur post-LOCA and MSLB. The heat loads remain within the individual room cooling coil capacities, and within the total HA and EC system capacities.

Section 8.10.1.3 Turbine Building Heating, Ventilation, and Air Conditioning System

The Turbine Building HVAC (HT) system heat loads are based upon the piping design temperatures, pump motor rated horsepower, mechanical equipment design temperatures, electrical, control equipment and lighting loads, and transmission loads from adjacent rooms/structures. The heat loads used for original plant design remain bounding for the PUR heat loads.

Section 8.10.1.4 Control Building Heating, Ventilation, and Air Conditioning System

The Control Building HVAC (HJ) system heat loads are based on the piping design temperatures, pump rated motor horsepower, mechanical equipment design temperatures, electrical and control room lighting loads, transmission loads from adjacent structures, and personnel loads. The heat loads used for the original plant design remain bounding for the PUR heat loads. Modifications to instruments installed in the control room do not affect the control room heat loads (Section 9.1). Refer to Section 9.9 for a discussion of control room habitability.

Section 8.11 Water Chemistry

Section 8.11.1 Steam Generator Blowdown Processing Systems

The SG blowdown system is described in UFSAR Section 10.3.5. The SG blowdown system is used in conjunction with the chemical addition system to control the chemical composition of the SG water within specified limits. The blowdown system also controls the buildup of solids in the SG water. The blowdown flowrates will not be impacted by PUR. The SG blowdown control valves have adequate margin to maintain design blowdown capability.

Section 8.11.2 Primary and Secondary Water Chemistry

The chemistry control program will not be affected by PUR. Chemistry will be controlled within the existing guidelines based on the EPRI Water Chemistry Guidelines (Reference 8-5).

Section 8.12 Secondary System Piping and Valves

Secondary system piping and valves were analyzed for the effects of PUR. The larger mass of the SG, the revised LOCA displacements on the SG nozzles and increased feed flow have been reanalyzed. LOCA displacements for the secondary side nozzles were derived from the LOCA analysis addressed in Section 5 of this report. Seismic response spectra at the secondary side nozzles were also obtained from the RCS coupled analysis addressed in Section 5. Re-analyses of secondary side piping and components confirm that requirements of the ASME Code have been met. The valve accelerations have been evaluated and are within the allowable standards for PUR.

Section 8.13 Low Temperature Overpressure Protection

Low Temperature Overpressure Protection (LTOP) is described in UFSAR Section 5.2.2. LTOP is a combination of measures that ensure that brittle fracture limits of the reactor vessel will not be exceeded during startup and shutdown operations in the case of a pressure transient. LTOP conditions exist when the RCS T_{cold} is less than the LTOP enable temperature and the RCS is capable of being pressurized. The LTOP enable temperature is a temperature below which the relief valves must be aligned to the RCS. It is determined using NRC guidelines in BTP MTEB 5-2 (Reference 8-4). LTOP encompasses:

- means for pressure relief, such as relief valves, with appropriate set point(s) and enable temperatures, and vents of a specified area,
- limitations on Reactor Coolant Pump (RCP) starts, and
- restrictions on RCS heatup and cooldown rates.

LTOP protection is provided by the two, redundant, SCS relief valves located, one each, on the hot leg SCS suction lines. The LTOP requirements are contained in the plant Technical Specifications and operating procedures.

Two major analyses constitute the basis for the determination of the LTOP requirements:

P-T limit analysis provides the maximum allowable pressure values as a function of T_{cold} for various heatup and cooldown rates. The values are calculated based on reactor vessel material properties that change with irradiation. This results in P-T limits being valid for a period of time, determined by projected neutron fluence at the end of the time period.

Pressure transient analysis consist of a limiting energy addition transient analysis and a limiting mass addition transient analysis. The analyses yield peak transient pressures that are compared with the P-T limits to identify LTOP limitations, including the final selection of heatup and cooldown rates.

The resulting peak pressures in these transients determine which P-T limits (i.e., for which heatup or cooldown rates) are protected by the LTOP system over specific RCS temperature ranges.

The impact of the PUR conditions on the previous LTOP functions is described below:

- PUR conditions affect the Mass and Energy (M&E) addition transients by increasing core decay heat.
- The greater RCS and SG secondary volume, greater SG metal mass, and greater heat transfer coefficient characteristics affect both the energy and mass addition transients.
- SG hydraulic characteristics change the RCS flowrate during LTOP mode operation. This affects P-T Limit correction factors.

Section 8.13.1.1 Input Parameters and Assumptions

For the PUR, an assessment of the existing LTOP analyses was performed. Core decay heat is a critical parameter associated with LTOP that is affected by the PUR. In addition, based on the larger SGs, changes to the RCS volume, SG volume, RCS flowrate, and SG heat transfer capacity were considered.

The effects of increased core power on projected vessel wall fluence are addressed in Section 5.1.2. The assumptions in the existing P-T limit analysis for projected fluence were reviewed and remain bounding.

Section 8.13.1.2 Acceptance Criteria for Analyses/Evaluations

LTOP controls and SCS relief valve setpoint will continue to provide adequate protection to ensure that the reactor vessel brittle fracture limits are not exceeded.

Section 8.13.1.3 Results and Conclusions

Sufficient margin and conservatism exists with the Analyses of Record (AOR) to address the direct effect of the plant change.

Section 8.14 Miscellaneous Electrical Reviews

The electrical systems are described in UFSAR Chapter 8. PUR will affect the electrical distribution system parameters and offsite grid. To determine the impact due to PUR, the following areas were evaluated:

1. Grid Stability
2. Main Power Transformers
3. Unit Auxiliary Transformer
4. Startup Transformers
5. Diesel Generators
6. Station Blackout Turbines
7. Isophase Bus
8. RCP Motors
9. Condensate Pump Motors
10. Heater Drain Pump Motors
11. Breaker Coordination/Relay Settings

The following is a summary of the evaluations that were performed to determine the impact of PUR on the offsite grid and plant distribution system:

Section 8.14.1 Grid Stability

A study was performed to verify grid stability with increased PUR generation capability. The study concluded that no stability problems would be encountered from either a maximum bucking or a maximum boosting condition. All single-contingency disturbances proved to be stable and within the study's criteria, even with an additional 7% megawatt generation above the projected generation capability associated with PUR.

Section 8.14.2 Main Power Transformers

The main transformers step-up the generator supplied voltage to the grid voltage (one transformer per phase). The PUR will increase the input/output currents in the transformer windings, thus increasing the heat generated in the transformer. The main transformers use oil to insulate and cool the transformer windings. The transformer generated heat is then dissipated by passing air through 6 banks of oil coolers each consisting of an oil pump and four fans.

The additional power to the three main transformers for PUR has been evaluated. PUR will raise the main transformer oil temperature. This oil temperature increase remains below the transformer's rated temperature capacity. Although not required by the PUR analysis, the main transformer will be modified to increase reliability.

Section 8.14.3 Unit Auxiliary Transformer

During normal operation, power for the onsite non-Class 1E AC system is supplied through the unit auxiliary transformer that is connected to the generator isolated phase bus. The unit auxiliary transformer has one primary winding and two secondary windings. Each secondary winding distributes non-Class 1E power to one of two busses that supply power to the unit's house loads via non-Class 1E switchgear.

An evaluation was performed to assess the impact of the Brake horsepower (Bhp) increase of the RCPs, heater drain pumps, and CD pumps on the unit auxiliary transformer. PUR affects the pump flowrate that, increases the pump Bhp. The evaluation results demonstrated the increased horsepower increases the unit auxiliary transformer load. However, the total electrical increase due to PUR is within the rated capacities of the unit auxiliary transformer. Therefore, there is no impact on the unit auxiliary transformer.

The effect of the horsepower load increase on the non-Class 1E 13.8 and 4.16 kV auxiliary electrical distribution system was also evaluated. The affected 13.8 kV switchgear, circuit breakers and cables and 4.16 kV transformers, switchgear, circuit breakers and cables were compared to their rated capacities. The values increased but were below each component's rated capacity.

Section 8.14.4 Startup Transformers

During unit outages and unit startup, offsite electrical power to the unit is supplied by the 13.8 kV secondary windings of the startup transformers. Power from the secondary windings enters the unit via two 13.8 kV switchgears

Once unit output generation is sufficient to safely supply house loads, the non-Class 1E house loads are transferred from the startup transformers to the unit's auxiliary transformer.

An evaluation was performed to determine the impact of increased load on the startup transformers from the RCPs, heater drain pumps, and CD pumps. The results show an increase in startup transformer load due to the increased pump Bhps. The total horsepower load for PUR is within the rated capacities of the startup transformers.

Section 8.14.5   Diesel Generators

The two Emergency Diesel Generators (EDGs) provide an independent source of Class 1E onsite power (4160 volts) for each of the two trains of Engineered Safety Features (ESF). The “A” and “B” train EDGs ensure onsite electrical power is provided to Class 1E switchgears for safe reactor shutdown in the event of a LOP and for post-accident operation of ESF loads.

Evaluation of the EDGs demonstrated that loads to either ESF train do not increase due to PUR. The existing EDG design specifications bound operation at PUR conditions.

Section 8.14.6   Station Blackout Turbines

The two SBO Gas Turbine Generators (GTGs) provide AC electrical power of sufficient capacity and reliability to operate the systems required for coping with a SBO for a minimum of 4 hours. The GTGs are an independent power source that can be connected to the unit through the primary winding of the ESF transformer that is normally aligned to the train A ESF 4.16 kV bus.

GTGs starting and loading is performed manually.

PUR will not add any additional load to the Class 1E ESF bus. Therefore, the GTGs design specifications bound operation at PUR conditions

Section 8.14.7   Isophase Bus

The isolated phase bus (or isophase bus) is the electrical connection from the main generator output terminals to the low voltage terminals of the main transformer and to the high voltage terminals of the unit auxiliary transformer.

The isophase bus consists of the electrical bus, enclosure, cooling system, heating unit, potential transformer, and connections to the main generator and transformers. The system is designed to sufficiently cool the bus without forced air up to 55% plant power. The isophase bus is normally cooled using forced air by the cooling cabinet in the stator cooling system whenever the generator is energized. The system has a redundant fan that is sufficient to maintain the cooling systems rating.

The isophase bus’ rating is 1600 MVA (forced cooling). This rating is greater than the main generator and main transformer ratings. Therefore, PUR has no impact on the isophase bus.

Section 8.14.8   Reactor Coolant Pump Motors

The RCPs receive non-Class 1E 13.8 kV power from the unit auxiliary transformer during normal unit operation. They receive non-Class 1E power from the startup transformers during unit startup and shutdown. The RCPs provide sufficient forced circulation flow through the RCS to assure adequate heat removal from the reactor core during power operation. Although the RCP Bhp is increasing, the new value is within the RCP motor rating. Therefore, the RCP motors remain acceptable for operation at PUR conditions.

See Section 8.14.3 and Section 8.14.4 for the RCP motor effects on the unit auxiliary and startup transformers.

Section 8.14.9 Condensate Pump Motors

The CD pumps receive non-Class 1E 4.16 kV power from the unit auxiliary transformer during normal unit operation. They receive power from the startup transformers during unit startup and shutdown. The CD pumps provide sufficient net positive suction head to the FW pumps under all normal and transient conditions. The CD pumps are also designed with redundancy to allow for one of the CD pumps to be removed from service without reducing unit rated load. Although the CD pump Bhp is increasing, the new value is within the CD pump motor rating. Therefore, the CD pump motors are acceptable for operation at PUR conditions.

See Section 8.14.3 and Section 8.14.4 for the CD pump motor effects on the unit auxiliary and startup transformers.

Section 8.14.10 Heater Drain Pump Motors

The heater drain pumps receive non-Class 1E 4.16 kV power from the unit auxiliary transformer during normal unit operation. They receive power from the startup transformers during unit startup and shutdown. The heater drain pumps operate in parallel, each taking suction from its high-pressure heater train drain tank and discharging to the suction header of the FW pumps. Although the heater drain pumps Bhp is increasing, the new value is within the heater drain pumps motor rating. Therefore, the heater drain pump motors are acceptable for operation at PUR conditions.

See Section 8.14.3 and Section 8.14.4 for the heater drain pump motor effects on the unit auxiliary and startup transformers.

Section 8.14.11 Breaker Coordination and Relay Settings

Each 4.16 kV ESF switchgear bus is equipped with four loss of voltage relays and four degraded voltage relays. The higher current due to increased non-class pump Bhp decreases the voltage at the 4.16 kV ESF bus and downstream equipment when the house loads are fed from the startup transformers. However, analysis of this effect demonstrates that the voltage decrease will not result in spurious operation of the loss of voltage or degraded voltage relays. In addition, the decreased voltage will not adversely affect the function of any class 1E equipment downstream of the breakers and relays.

The PUR will increase the electrical distribution system currents. An evaluation was performed to assess the impact of the higher currents on the affected breakers and protective relays. The results show that the higher currents are within the rated current and short circuit currents of the breakers and relays and acceptable for operation at PUR conditions.

Section 8.15 Miscellaneous Instrumentation and Control Reviews

The Instrumentation and Control (I&C) systems are described UFSAR Chapter 7. PUR impact evaluations were performed on the following I&C systems:

1. CD pump minimum flow control,
2. SG FW pump minimum flow control,
3. SG FW pump net suction pressure alarm and trip,
4. FW water level control system,
5. FW heater drains control,
6. CD hotwell level control,
7. Steam Bypass Control System (SBCS),
8. Reactor Regulating System (RRS), and
9. Reactor Power Cutback System (RPCS).

These evaluations show that the control systems bound operation for PUR.

Section 8.15.1 Condensate Pump Minimum Flow Control

The condensate pump minimum control valves are set to maintain pump flow above the low flow trip setpoint. The minimum control valves respond to the CD pump discharge flow signals. The valves open when the CD pump discharge flow decreases and modulate open as necessary to maintain the setpoint of the controller. Each CD pump has a flow transmitter and flow switch to provide low flow pump control.

The evaluation of the CD pump performance demonstrated that the current system design will bound the PUR conditions. No modifications will be required for the minimum flow control system or the minimum flow control valves.

Section 8.15.2 Steam Generator Feedwater Pump Minimum Flow Control

Each SG has its own Digital Feedwater Control System (DFWCS). Each system regulates FW flow to its corresponding SG by adjusting the position of the downcomer valve and/or the economizer valve or by regulating the speed of the FW pump. The FW pump mini flow valves assure that there is always sufficient flow through the FW pumps to protect them from overheating.

The evaluation of the FW system performed in support of this licensing amendment request demonstrates that the original system design bounds operation at PUR conditions. No modifications are required to the FW pump minimum flow control system and minimum flow control valves.

FW pump suction pressure is monitored to provide pump protection by initiating a FW turbine trip, if necessary.

Evaluation of the FW pump performance demonstrated that the PUR FW pump NPSH is bounded by the original design. There are no changes needed for the FW pump NPSH alarm/trip and the system is acceptable for operation at PUR conditions.

Section 8.15.3 Heater Drains Control

A study of the extraction steam and drains system was performed for PUR. The study concluded that total flows in and out of the heater drain tank and FW heaters will increase, however this increase would not exceed the capabilities of the monitoring instruments.

Evaluation of the heater drains control system demonstrated that operating level of the heaters is adequately maintained after PUR. There are no changes needed for the heater drain tank or FW heater level controls and the system is acceptable for operation at PUR conditions.

Section 8.15.4 Condenser Hotwell Level Control

An evaluation of the condenser hotwell level control system was performed for PUR. The study concluded that this system is not affected by PUR. No changes are needed for condenser hotwell level control for PUR.

Section 8.16 Essential Spray Pond System

The ESPS is described in UFSAR Sections 9.2.1 and 9.2.5. The ESPS provides essential heat removal from the following:

- the Essential Cooling Water System (ECWS) (essential and normal operating mode) and
- the EDG.

The ESPS thermal performance analyses have been revised to be consistent with Regulatory Guide 1.27 (Reference 8-7) for PUR. The thermal performance and inventory of the ESPS (Ultimate Heat Sink (UHS)), is modeled as an integral part of the ECWS and ESPS thermal performance analyses. The decay heat used in the analysis was determined in accordance with BTP ASB 9-2 (Reference 8-4). Consistent with the AOR, the thermal performance analyses were performed using the COPATTA code (Reference 8-6). The analysis included the reduced instrument uncertainties associated with the new temperature sensors that will be used to monitor the ESPS water temperature (see Section 9.1). This smaller uncertainty allows a lower initial ESPS temperature (equal to the Technical Specification SR 3.7.9.2 of 89 ^{o}F plus the instrument loop uncertainty). This lower initial temperature increases the thermal capacity of the ESPS and offsets the increased reactor decay heat loads. The revised thermal performance and inventory analyses confirm that the ESPS adequately removes heat from the ECWS and EDG system following normal and forced shutdowns and postulated DBEs. The UHS water inventory available to support operation of the ESPS during post-LOCA conditions has been verified to be sufficient to allow continued operation in excess of 26 days without any makeup water supply. This time period is consistent with the 26-day period required in the Technical Specification Bases 3.7.9.

The original design basis of the ESPS bounds the expected operation conditions associated with PUR.

Section 8.17 Conclusion

Current plant components can accommodate changes to the key plant operating conditions (steam flow, pressure, and temperature) affecting the BOP system performance characteristics for PUR. Changes will be made to the MT control logic, MSIV bypass valves, and ESPS temperature indicators (see Section 9.1).

Section 8.18 References

Reference 8-1 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Reference 8-2 Code of Federal Regulations, Title 10, Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

Reference 8-3 Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Technical Specifications, through Amendment No. 137, December 6, 2001 (Amendment No. 136 not yet implemented).

Reference 8-4 U. S. Nuclear Regulatory Commission Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

Reference 8-5 EPRI PWR Secondary Water Chemistry Guidelines - Revision 5 (TR-102134-R5).
EPRI PWR Primary Water Chemistry Guidelines Volumes 1 and 2 - Revision 4 (TR-105714-V1RA and TR-105714-V2RA).

Reference 8-6 Bechtel Power Corporation, "Containment Pressure and Temperature Transient Analysis (COPATTA)," as described in Bechtel topical report BN-TOP-3 Revision 4-"Performance and Sizing of Dry Pressure Containments."

Reference 8-7 Regulatory Guide 1.27, Ultimate Heat Sink for Nuclear Power Plants, Revision 2, January 1976.

Reference 8-8 Electric Power Research Institute (EPRI) Motor Operated Valve (MOV) Performance Prediction Model (PPM) Program, TR-103237, November 1994.

Section 9 MISCELLANEOUS TOPICS

Section 9.1 Modifications Required to Implement Power Uprate

The following modifications are required to support Power Uprate (PUR):

1- The Containment Spray System (CSS) must operate against a higher containment backpressure during postulated Loss-of-Coolant Accidents (LOCAs). Analysis has demonstrated successful system capability to deliver the desired flow to the containment environment at the desired pressure. However, margin for surveillance testing is reduced. A modification will be implemented to increase the margin for surveillance testing of the CSS pump. Specifically, the discharge orifices down stream of the CSS pumps that are used for measuring flow will be replace with annubar flow elements.

2- Revised thermal performance analysis of the Essential Spray Pond System (ESPS) credits a smaller instrument uncertainty for the ESPS temperature instruments than existing installed condition. Therefore, a modification will be implemented to replace the existing temperature instruments with new instruments that yield the credited instrument uncertainty.

3- Main Turbine (MT) analysis, which was performed by the turbine manufacturer, indicates a recommended change to the steam admission logic to the high pressure turbine. A change from two admission to single admission valve flow, to allow equal flow through all quadrants of high pressure turbine entrance will be made (see Section 8.5). This modification will eliminate steam induced cyclic loading and result in a reduction in overall stresses on the high pressure turbine blades.

4- The differential pressure that the Main Steam Isolation Valve (MSIV) bypass valves must close against following a Main Steam Isolation Single (MSIS) is increasing because of the increased MSIS setpoint. A modification will be implemented so that the valves are capable of closing against the increased differential pressure.

5- The Engineered Safety Features Actuation System (ESFAS) and Reactor Protection System (RPS) Steam Generator (SG) 1 - low and SG 2 - low functions allowable values will be changed from 890 psia to 955 psia. This change is required due to the increased main steam pressure associated with PUR condition.

6- PUR will increase Reactor Coolant System (RCS) average temperature (T_{ave}) and Feedwater (FW) flowrates. Therefore, various control system tuning will be made. In addition, a change will be made to the Steam Bypass Control System (SBCS) master controller to maintain an adequate dead band within the controller with SG operating pressure.

Section 9.2 Post-Loss-of-Coolant Accident Hydrogen Generation

The post-LOCA hydrogen generation analysis was performed for PUR. The existing containment aluminum and zinc inventories remain unchanged by PUR. The PUR post-LOCA containment temperature profile is discussed in Section 6.2.2. A core wide oxidation rate of 0.86% was used to predict the quantity of hydrogen released because of the zirconium metal water reaction. Consistent with the licensing basis, the hydrogen recombiners are assumed to be placed into service at 100 hours. The analysis concluded that the peak predicted hydrogen concentration remains less than 3.99% by volume.

As discussed in UFSAR Section 6.2.5 (Reference 9-10), and as accepted by the NRC as documented in the Safety Evaluation Report (SER) (Reference 9-11), under postulated LOCA conditions, the Reactor Drain Tank (RDT) room may become an essentially closed room with the only venting occurring through an annular opening in the ceiling. The potential therefore exists for accumulation of hydrogen in the RDT room.

The maximum hydrogen concentration has been analyzed in the RDT room using the same NRC reviewed methodology as the original design (Reference 9-12). The results of the analysis show that the gas plume exiting the room remains well below the combustible limit utilizing a conservative post-LOCA pressure, temperature, and bulk hydrogen profiles that bound those predicted to occur at PUR conditions. The results are consistent with the original NRC established limit. Additionally, the RDT room has been designed to eliminate all potential ignition sources within the room.

In conclusion, no hydrogen combustion would occur in the main containment volume. Furthermore, no potential for hydrogen pocketing exists within the RDT room due to PUR.

Section 9.3 Radiological Assessment

Section 9.3.1 Description

The impact of PUR on the radiological design basis was evaluated and considered three topics:

1. a review of normal/design shielding,
2. a review of source term impact on releases to offsite locations during normal operations from normal release pathways, and
3. a review of post-accident shielding.

Section 9.3.2 Scope of Review

PUR design source terms are discussed in Section 7.6. Radionuclide concentrations for all systems were evaluated. The evaluation included the reactor coolant loop, spent fuel, auxiliary systems that are contaminated/process radioactive effluents, and storage/processing tanks.

Source term impacts were evaluated as follows:

1. RCS source terms were reviewed relative to normal shielding and operation dose rates.
2. RCS source terms may be transported via liquid/airborne pathways outside of containment, and potentially to offsite release points. The evaluation considered PUR effects on normal releases.
3. Post-accident vital area doses were evaluated for all Design Basis Events (DBEs).
4. Control room design and habitability requirements are discussed in Section 9.9.

Section 9.3.3 Design Requirements

Shielding for normal operations is designed to be within the criteria from 10 CFR Part 20 for operator dose and access control. Per UFSAR Chapter 12 the criteria for shielding design is Regulatory Guide 8.8 (Reference 9-9). Radwaste equipment is designed to maintain offsite releases As Low As Reasonably Achievable (ALARA) and within limits specified in 10 CFR Part 20 (old) and 10 CFR Part 50, Appendix I (Reference 9-20). Additional guidance is provided in Regulatory Guide 1.143 (Reference 9-5).

Post-accident shielding and vital area doses are consistent with the guidelines of NUREG-0737 (Reference 9-7).

Section 9.3.4 Assumptions

The assumptions used to verify the radiological design basis are:

- Shielding designs are based on 1% failed fuel RCS source terms.
- Normal offsite releases are based on expected RCS source terms as described in Section 7.6. Other assumptions for release assessments are included in UFSAR Sections 11.3.3 and 12.4.2.2.
- Major assumptions for offsite and vital area dose evaluations are consistent with the criteria from applicable NRC Standard Review Plan (SRP) Sections and Regulatory Guides.
- Assumptions stated in UFSAR Sections 12.2.3 and 18.II remain applicable to post-accident shielding and vital area dose evaluations.

Section 9.3.5 Method of Evaluation

Shielding reviews included the difference between the existing RCS design source terms (UFSAR Chapter 11) and the PUR RCS design source terms (Section 7.6). Conservative factors from the original shielding calculations were utilized along with plant configuration changes that result from installation of new SGs.

Normal offsite doses and releases were evaluated by reviewing the difference between the existing RCS expected source terms and the PUR RCS expected source terms. The calculations demonstrate compliance with 10 CFR Part 20 (old) and 10 CFR Part 50 Appendix I requirements.

Section 9.3.6 Summary of Evaluations

Section 9.3.6.1 Normal Plant Shielding

Existing fission product design source terms remain bounding for PUR.

Design and expected source terms for fission products were evaluated to be bounded by existing source terms. Therefore, normal plant shielding and operation dose will remain within the original design.

The N-16 activity in the cold leg and at the letdown line will decrease due to longer loop transit times through the longer U-tubes in the new SGs. As a result, the N-16 in the CVCS delay coil will also decrease. Existing shielding for the auxiliary building therefore remains bounding for PUR.

Section 9.3.6.2 Normal Offsite Releases

UFSAR Table 11.3-6 provides annual plant gaseous releases for normal operations. Changes to these releases are proportional to changes to the RCS source term. Table 11.3-6 indicates that existing calculated releases are below Maximum Permissible Concentration (MPC) limits.

PVNGS does not release any liquid offsite. The only normal source of liquid effluents is discharge from the secondary and tertiary systems to the evaporation ponds. The Offsite Dose Calculation Manual (ODCM) controls these releases. Since the expected source term has not increased, a review of radiological calculations concluded that existing discharge limits to evaporation pond remain limiting for PUR.

For gaseous effluents, UFSAR existing calculated values for releases are below regulatory limits and bounds PUR conditions.

Section 9.3.6.3 Radioactive Waste Management Systems

The radioactive waste management system was reviewed for PUR. There are no impacts to the design of the liquid, gaseous, or solid radwaste systems. The quantity of gaseous and liquid waste produced for PUR is bounded by the original design.

Section 9.3.6.4 Post-Accident Shielding

Accident source terms consistent with the requirements of NUREG-0737 have been reviewed for PUR. Since the post-accident source term has increased (see Section 6.5), new shielding analyses were performed. The existing post-accident shielding analysis contains sufficient margin to compensate for an increase in dose rates due to

PUR. The revised analysis demonstrated that the post-accident zone maps presented in UFSAR Section 12.2 remain bounding.

Section 9.3.6.5 Post-Accident Vital Area Doses

Five plant locations have been identified as post-accident vital areas. They are:

- the control room,
- the Technical Support Center (TSC),
- the Emergency Offsite Facility (EOF),
- Auxiliary Building West @ elevation 100 ft, hydrogen recombiner installation location, and
- Sampling Station @ 140 ft, chemistry hot lab.

An evaluation to assess the impact of PUR on these vital areas was performed. The evaluation demonstrated that these vital areas continue to satisfy the criteria from 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 and/or NUREG-0737.

Section 9.3.6.5.1 Control Room

Control room habitability requirements are addressed in Section 9.9 of this submittal.

Section 9.3.6.5.2 Technical Support Center

The TSC dose assessments are based on the most limiting design basis LOCA fission product release (Section 6.5). Analysis parameters, including essential TSC Heating Ventilation and Air Conditioning (HVAC) and habitability parameters (UFSAR Tables 18.3.A-3 and 18.3.A-4), remain bounding for PUR. The assumed unfiltered in-leakage to the TSC has been confirmed by testing.

The total exposures resulting from a design basis accident (DBA) are provided in Table 9.3-1. A review of the calculated TSC doses demonstrated that exposures are below the dose limits specified by GDC 19.

Table 9.3-1
TSC Occupant 30 Day Post-LOCA Exposure (REM)

	Thyroid	Whole Body	Beta-Skin
Internal Cloud Exposure	15.5	1.17	26.0
Direct Dose Due to Iodine Build-up on Charcoal Filtration	N/A	0.1	N/A
Total Dose	15.5	1.27	26.0

Section 9.3.6.5.3 Emergency Operations Facility

The EOF dose assessments are based on the most limiting design basis LOCA fission product release (Section 6.5). Analysis parameters, including essential EOF HVAC and habitability parameters (UFSAR Tables 18.3.A-6 and 18.3.A-7), remain bounding for PUR. The assumed unfiltered in-leakage to the EOF has been confirmed by testing.

The total exposures resulting from the DBA are provided in Table 9.3-2. A review of the calculated EOF doses demonstrated that exposures are below the dose limits specified by GDC 19.

Table 9.3-2
EOF Occupant 30 Day Post-LOCA Exposure (REM)

	Thyroid	Whole Body	Beta-Skin
Internal Cloud Exposure	28.5	1.23	27.3
Direct Dose Due to Iodine Build-up on Charcoal Filtration	N/A	0.00 (1)	N/A
Total Dose	28.5	1.23	27.3

Note: (1) The EOF facility is located below ground and the charcoal filtration unit is at ground level. There is no direct line between the filtration unit and the EOF. The filtration unit is shielded by earth. Therefore, contribution from the filtration unit due to iodine buildup is neglected.

Section 9.3.6.5.4 Hydrogen Recombiner Area

Radiation evaluations of the hydrogen recombiner area were conducted in accordance with Section II.B.2 of NUREG-0737. Operator access to the hydrogen recombiner area during post-accident periods is required to allow installation and operation of the recombiners. Dose rate to the operator is not to exceed 5 REM/hr. The source terms correspond to those noted in Section 6.5. Radiation levels were based on radiation sources from the post-accident operation of the following Structures, Systems, or Components (SSCs):

- containment (shine from),
- Safety Injection System (SIS)/Shutdown Cooling (SCS)/CSS,
- Chemical and Volume Control System (CVCS) (up to purification filter inlet), and
- Post-Accident Sampling System (PASS).

Post-accident calculated dose rates are summarized in Table 9.3-3. The resulting total dose rate is less than 5 REM/hr, and is within the criterion from Section II.B.2.3-(b) of NUREG-0737.

Table 9.3-3
Maximum Hydrogen Recombiner Area Dose Rates

Source	Dose Rate (REM/hr)
Containment Shine	1.9
Piping and Equipment	0.3
Total Dose Rate	2.2

Section 9.3.6.5.5 Sampling System

APS has submitted a License Amendment request to implement Technical Specification Task Force (TSTF) Change TSTF 366, Elimination of Requirements for PASS. This Technical Specification change will revise the Administrative Section 5.0 to remove the requirements for the PASS system. This effort has recently been approved by the NRC in the form of a generic SER (Reference 9-6) and specifically for PVNGS, amendment 136 to the Operating License. Therefore, NUREG-0737 Section II.B.3 is no longer applicable.

Section 9.3.7 Summary of Conclusions

No changes to Structures, Systems, or Components (SSCs) are required to provide adequate radiation protection for operators or the public during normal and post-accident conditions. The plant shielding design remains bounding for PUR.

Section 9.4 Electrical Equipment Qualification

The revised LOCA and Main Steam Line Break (MSLB) analyses resulted in a change to the environmental parameters for the equipment required by 10 CFR Part 50.49 (Reference 9-20). These changes in the environmental parameters were reviewed for impact on the Qualification of electrical Equipment (EQ).

Section 9.4.1 Scope of Review

The LOCA and MSLB analyses were revised for PUR. The PUR environmental parameters (such as, radiation, temperature, and pressure) were assessed on the EQ required by 10 CFR Part 50.49.

Section 9.4.2 Summary of Evaluations

Table 9.4-1 provides a comparison of the equipment exposure doses inside and outside the containment building. The comparison demonstrated that the total integrated gamma dose has been reduced. The gamma dose reduction is due to a dose reduction in the containment sump (due to increased sump water volume as a result of increase in the RCS volume). The beta dose in the containment has increased as result of the increase in power level. The revised gamma doses inside and outside containment and beta doses inside containment for each component was compared with the EQ test doses. This comparison demonstrated that the electrical equipment remains qualified as required by 10 CFR Part 50.49 and in accordance with IEEE 323-1974 (Reference 9-21).

Table 9.4-1
Containment 180 Day Dose Summary in Megarads

Source	Main Spray (1)		Aux. Spray (1)		Unsprayed (1)	
	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t	3876 MW_t	3990 MW_t
Gamma Doses						
Airborne	7.02	7.23	3.37	3.47	3.09	3.18
Sump Shine	3.41	2.52	18.5	13.6	18.5	13.6
Plateout	0.95	0.98	0.95	0.98	0.95	0.97
Total Gamma	11.4	10.7	22.8	18.1	22.5	17.8
Beta Doses						
Airborne	116	124	116	124	117	124
Plateout	143	147	143	147	143	147
Total Beta	259	271	259	271	260	271
Total Doses						
Total Dose	270	282	282	289	283	289

Notes: (1) See UFSAR Section 6.5.2.3.

A comparison of the revised LOCA and MSLB (inside and outside containment) temperature profiles with the existing profiles indicates that the PUR peak accident temperature is decreased for MSLB and increased for LOCA (refer to Section 6.2). The containment pressure profile during LOCA also changed (refer to Figure 6.2-2). Conservative MSLB and LOCA long-term pressure and temperature profiles were developed based on the results provided in Section 6.2. These 180-day profiles were used to qualify the containment and Main Steam Support Structure (MSSS) equipment. The assessment of EQ data files and test reports reveals that the equipment required remains qualified with exception of non-standard Raychem splices and In-Core Instrument (ICI) connectors. The instrumentation required by the Regulatory Guide 1.97 (Reference 9-26) is qualified for the time its function is required post-accident.

Section 9.4.3 Summary of Conclusions

With the exception of ICI connectors and non-standard Raychem splices, the assessment of the existing EQ data files and test reports verified that the equipment in the EQ Program remains qualified. ICI connectors and non-standard Raychem splices will be qualified before implementation of this amendment.

Section 9.5 Valve Program

The design basis of the safety related Power-Operated Valves Motor (MOVs), Air (AOVs), and Solenoid (SOVs) were reviewed. The reviews considered the requirements of the NRC's Generic Letters 89-10 and 95-07 (Reference 9-18 and Reference 9-19). The design basis parameters included pressure, temperature, and differential pressure. These reviews concluded that:

1. There are no changes to the design basis pressure or temperature of any safety related power operated valves.
2. The bounding parameters used to establish the worst-case differential pressures for safety related power operated valves were reviewed. The review concluded that existing valve actuators will adequately perform their intended design function after PUR with the exception of the MSIV bypass valves. The MSIV bypass valves will be modified for the increased differential pressure before implementation of this license amendment.

Therefore, the existing design basis for all power operated valves (MOVs, AOVs, and SOVs), except for the MSIV bypass valves, bound operation at PUR.

Section 9.6 Fire Protection Program

The Fire Protection Program and the transient analysis for fire induced events are discussed in UFSAR Sections 9.5 and Appendix 9B respectively. The transient analysis for fire induced events has been re-analyzed for the PUR using the CENTS code. The revised analysis utilized the SRP Branch Technical Position (BTP) 9-2 (Reference 9-8) decay heat values for the auxiliary system design. The operator action time constraints identified in the revised transient analysis were then reviewed against the existing time constraints. The only time constraint that is directly affected by the PUR is the time it takes to deplete the Condensate Storage Tank (CST) and the Reactor Makeup Water Tank (RMWT) volumes during plant cooldown to the SCS entry condition. The revised transient analysis concluded that safe shutdown methodology and results identified in the UFSAR is maintained considering the modified operator response times for PUR.

Section 9.7 Probabilistic Risk Assessment

This license amendment request is not being submitted as a risk-informed request. Guidance on the use of risk information in license amendment reviews that are not risk-informed was published in NRC Regulatory Issue Summary 2001-02 (Reference 9-25). This guidance outlines a process that will be followed by the NRC if a non-risk informed submittal is believed to have a potential for a large increase in risk, to the degree that existing regulations do not provide adequate protection of the public health and safety.

This PUR is not large, but is incremental in nature, and is not expected to result in significant changes in risk. Compliance with existing NRC regulations will continue to assure adequate protection of public health and safety.

The existing Probabilistic Risk Assessment (PRA) for PVNGS will be updated to reflect the changes in plant design and operation due to PUR and new SGs. Updating the PRA will be done through the normal PRA update process, after the modifications have been completed to the plant. This process assures that the PRA continues to adequately reflect the as-built and as-operated condition of the plant.

Section 9.8 Environmental Impact Evaluations

An evaluation of the Final Environmental Statement (FES, Reference 9-13) was completed for PUR. The evaluation compared the expected PUR operating parameters with the existing parameters and conclusions in the FES.

Administrative procedures require all plant modifications to be evaluated to determine their effect on the environment. Operation at PUR condition has been evaluated and is bounded by the Analyses of Record (AOR) in the FES.

The cooling tower blowdown rate and discharge to the evaporation ponds will increase. The discharge to the evaporation ponds is bounded by the original design as described in UFSAR Section 2.4.8.2.3. Table 9.8-1 is a comparison of critical parameters found in the FES and demonstrates that PUR is bounded by the original environmental analysis.

Table 9.8-1
Parameter Comparison for Environmental Analysis

	FES-CP	3990 MW_t Licensed Uprate
Capacity	4100 MW_t	3990 MW_t reactor/4013 MW_t thermal
Heat rejection	9.3E+09 BTU/hr	9.15E+09 BTU/hr

The FES addresses “drift” including water and salt. Since the FES values exceed the proposed PUR values, the PUR remains bounded by the analyzed values in the FES.

The approved FES assumed heat rejection capacities are greater than the expected PUR values. The FES also assumed a cooling tower system design with drift rates higher than the existing system design. PUR is bounded by the original analysis.

Section 9.9 Control Room Habitability

Section 9.9.1 Control Room Radiological Design

Control room habitability is described in UFSAR Section 6.4. The control room is designed to 10 CFR Part 50, Appendix A, GDC 19 for all DBEs. Regulatory Guide 1.4 and 10 CFR Part 50.34 (a)(1)(ii)(D) are used as guidance for source term, release, and mitigation of consequences analyses.

The existing radiological design includes all known and unknown sources of unfiltered air leaking into the positively pressurized control room boundary. The leakage includes but is not limited to, ingress/egress (10 cfm per the SRP, Reference 9-8), essential/normal HVAC component in leakage, habitability boundary "wall" in leakage, and other system in leakage into the pressurized envelope (such as instrument air and nitrogen).

Section 9.9.1.1 Essential System Parameters used in Radiological Analysis

The following essential control room HVAC (HJ) parameters were used to determine the integrated in leakage rate:

1. Maximum outside air supplied by the essential HVAC system and filtered by a 2 inch (minimum depth) charcoal bed is 1000 scfm.
2. Minimum air recirculation rate for the essential HVAC system is 25,740 scfm, filtered by 2" (minimum depth) charcoal filter beds.
3. The filtration units meet the Regulatory Guide 1.52 requirements (Reference 9-15).
4. Minimum pressure differential for the control room radiological boundary is 1/8 inch (gauge) of water.
5. Maximum (net) control room volume is 1.61E+05 ft^3.
6. Normal control room HVAC system isolates upon activation of the essential HVAC.
7. 10-scfm in leakage into the habitability envelope is due to personnel ingress/egress (SRP Section 6.4).

Section 9.9.2 Single Failure Applied to Control Room Habitability Analysis

The control room habitability analysis assumed that both essential HJ trains would be actuated and control room operators would immediately turn one off. This is a conservative assumption since doubling the outside air supply would pressurize the control room envelope beyond the minimum design ΔP, thereby reducing in leakage. The single failure for this analysis is the same as the single failure for the DBE.

Section 9.9.3 Control Room Radiological Assessment

The control room dose with bounding unfiltered in leakage is evaluated for the following four limiting accidents:

1. LOCA, as described in UFSAR Section 15.6.5 and Appendix 15B.
2. Control Element Assembly (CEA) ejection, as discussion in UFSAR Section 15.4.8 and Appendix 15B.
3. RCP sheared shaft with pre-existing iodine spike in the reactor and a stuck open Atmospheric Dump Valve (ADV), as discussed in UFSAR Section 15.3.4 and Appendix 15 B.
4. Steam Generator Tube Rupture (SGTR) with a stuck open ADV with a pre-existing iodine spike to the secondary side, as described in UFSAR Section 15.6.3 and Appendix 15 B. It has been determined that Pre-accident Iodine Spike (PIS) dominates the Generated Iodine Spike (GIS).

Section 9.9.3.1 Radiological Parameters used for Control Room Evaluation

During the accident, control room personnel may receive doses from the following sources:

1. Direct whole-body gamma dose from the radioactivity present in the containment building.
2. Direct whole-body gamma dose from the radioactive cloud outside the control building.
3. Direct whole-body gamma dose from the control building essential filtration system inside the control building.
4. Whole-body gamma, thyroid inhalation, and beta skin doses from the airborne radioactivity present in the control room and in the environment surrounding the control room. Airborne radioactivity will be drawn into the control room due to the intake of outside air required to maintain a positive pressure in the control room.

For calculating the exposure to control room personnel, occupancy factors and breathing rates are as follows:

0 to 24 hours	occupancy factor = 1	breathing rate = 3.14E-04 m^3/sec
1 to 4 days	occupancy factor = 0.6	breathing rate = 1.75E-04 m^3/sec
4 to 30 days	occupancy factor = 0.4	breathing rate = 2.32E-04 m^3/sec

Radioactivity concentration (Ci/m^3) in the radioactive cloud surrounding the control room is the product of the building leak rate (Ci/sec) and the control room atmospheric dispersion factor, χ/Q (sec/m^3). Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) χ/Q's are presented in UFSAR Section 2.3. A tabulation of control room χ/Q's is presented in UFSAR Table 15B-5.

Credit is taken for concrete shielding provided by the control room walls and ceiling. Table 9.9-1 summarizes the most limiting atmospheric dispersion factors (from all buildings points of release to nearest control room intake).

Table 9.9-1
Most Limiting Χ/Q's

Time Period	Control Room Χ/Q (sec/m^3)$^{(1)}$
0 to 8 hours	1.56E-03
8 to 24 hours	1.08E-03
1 to 4 days	4.15E-04
4 to 30 days	1.03E-04

Note: (1) Includes occupancy factors.

Section 9.9.3.2 Results and Conclusions

As shown in Table 9.9-2, the limiting organ dose (thyroid) controlling accidents are the RCP sheared shaft and the SGTR with stuck open ADV. These events are less than the guidelines provided in SRP Section 6.4. These events establish the upper bound for the unfiltered in leakage for control room habitability. These analyses assume 10-scfm in leakage for ingress/egress for a total of 63 scfm-unfiltered in leakage from all other sources.

In conclusion, the radiological consequences to control room operators are within the requirements of 10 CFR Part 50 Appendix A GDC 19 for all DBEs.

Table 9.9-2
Summary of Control Room Radiological Assessment

Event	Condition of Fuel During the Accident	Release Duration (hr)	Control Room Isolation Time/Signal	Whole Body Dose (REM)	Limiting Organ Dose (REM) (Thyroid)
LOCA	100% Fuel melt (1)	720	12 sec Safety Injection Actuation Signal (SIAS)/Containment Isolation Actuation Signal (CIAS) + 50 sec for control room to close = 72 sec	1.95	14.7
CEA Ejection	Fuel experiences Departure from Nucleate Boiling (DNB) (2)	720	69 sec SIAS/CIAS + 50 sec for control room to close = 119 sec	0.912	21
RCP Shear Shaft with Stuck Open ADV	Fuel experiences DNB	11	Not applicable release is via secondary	0.966	26.9
SGTR with Stuck Open ADV	No Failed Fuel/PIS	24	Not applicable release is via secondary	0.363	26.8

Notes: (1) Regulatory Guide 1.4 model.
(2) Refer to UFSAR Section 15.4.8.

Section 9.9.4 Testing and Conformation of Design Bases Parameters

The control room essential filtration units are tested per the requirements of Technical Specification 3.7.11 (Reference 9-22) using standard testing methods that conform to Regulatory Guide 1.52. High Efficiency Particulate Air (HEPA) filter banks are tested in conformance with ANSI N510 (Reference 9-17). HEPA filter banks comply with Regulatory Guide 1.52 Position C.5.c. The control room pressure envelope is tested for positive pressurization per Technical Specification 3.7.11. Additionally, a special, integrated pressure boundary leak test was preformed to validate the total unfiltered in leakage assumption used in the control room habitability analysis. The results of this validation test demonstrated that the design assumption of 63 SCFM unfiltered in leakage bounds the actual as-built plant condition.

Section 9.10 Natural Circulation Cooldown Analysis

Compliance with the requirements of BTP RSB 5-1 (Reference 9-8) has been documented using a combination of both analyses and actual plant testing. Specifically, a computer simulation of a plant natural circulation cooldown to SCS entry conditions was conducted for CESSAR-F (Reference 9-1). Later an actual natural circulation cooldown test was conducted. Results from this test along with a detailed evaluation were submitted and approved by the NRC (Reference 9-2 and Reference 9-3).

In 1995, the RSB 5-1 simulation was re-performed from hot standby to SCS entry conditions for 3876 MW_t conditions. Along with this analysis, a separate analysis was performed for the SCS. This analysis demonstrated the capability of one train of SCS to cool the plant from SCS entry conditions to 200 °F.

The PUR natural circulation cooldown analysis followed the same RSB 5-1 BTP guidance. The analysis contained two cases.

1. One case or simulation was performed in the same manner as the original analysis using the CVCS for both inventory makeup (via the charging pumps) and for depressurization (via pressurizer auxiliary spray).
2. A second case was added to demonstrate an alternate method of RCS depressurization using the reactor coolant gas vent system. This case used one High Pressure Safety Injection (HPSI) pump for inventory control in place of a charging pump. This case was performed to demonstrate that the alternate method provides a diverse and redundant means of accomplishing RCS depressurization and inventory makeup.

The PUR natural circulation cooldown analysis demonstrated compliance with the requirements of BTP RSB 5-1. All BTP RSB 5-1 criteria with respect to time, use of safety grade equipment, operator action outside the control room, single failure, redundancy of equipment, power available and GDC 1 through 5 are met. The results of the PUR natural circulation cooldown analysis are bounded by the original design criteria.

Section 9.11 Impact of Increased Power on Operations

The effects of PUR and larger SGs on plant operations have been assessed. These effects are discussed in the following sections for impacts on the Control Room, Operations Procedures, PVNGS Simulators, and Training. After implementation of this license amendment, no changes to the operator's multi-unit license are required.

Section 9.11.1 Impact on the Control Room

The PUR will have a limited impact on the operator interfaces for control room displays, controls, and alarms. The plant modification process will implement the required changes through programmatic reviews.

There are control room indications that have "tick marks," that are controlled by Operation's Department administrative control procedures. These tick marks indicate a Technical Specification limit or setpoint. One parameter limit is affected by PUR, specifically the limit for RCS cold leg temperature (T_{cold}). Operators will be trained on the new RCS T_{cold} before implementation of this license amendment per the requirements of administrative control procedures.

The following control room alarms are affected by the required Technical Specification changes:

- RCS T_{cold} and
- Low Steam Generator Pressure (LSGP) trip and pre-trip for RPS and MSIS.

The alarm setpoints will be changed before operation at PUR. Operators will be trained on the new alarm setpoints per the requirements of administrative control procedures.

As stated previously, the control room ESPS temperature indicators will be changed because of the PUR. Operators will be trained on the new SP temperature indicators before operation at PUR per the requirements of administrative control procedures.

PVNGS has no "zoned indications" that are utilized by operations that are affected by the PUR. Original meters in the GE Electro-Hydraulic Control (EHC) panel for throttle pressure, intermediate pressure, load set, and load have a red "zoned" area at the upper end of their relative scales. However, these zoned areas are not addressed in any operations procedures and are not used by the operations staff for main turbine control. Therefore, PUR has no effect on any zoned indications.

Qualified Safety Parameter Display System (QSPDS) will be modified for the larger SGs (i.e., larger RCS volume, larger S/G volume, etc.). Operators will be trained on the QSPDS changes before operation at PUR per the requirements of administrative control procedures.

Section 9.11.2 Impact on Operations Department Procedures

Operation at PUR will result in changes to the existing plant response during transients. There will be changes in instrumentation and associated instrument uncertainties. An assessment of the expected plant response indicates that minor Emergency Operating Procedure (EOP)/Abnormal Operating Procedures (AOP) changes are expected. These EOP/AOP changes will not affect credited operator actions or mitigation strategies. Therefore, no unit-specific mitigation strategies are required for PUR.

PUR results in changes in operations procedures, such as surveillance tests, normal operating, general operating, and/or alarm response procedures. Any required procedure change will be identified and incorporated before operation at PUR.

Section 9.11.3 Impact on the PVNGS Simulators

The PVNGS simulators are modeled after Unit 1. Related hardware changes are incorporated into both simulators, after the modifications are incorporated into Unit 1. The changes made to the simulators will be tested in accordance with ANSI/ANS 3.5-1985, Section 5.4.1 (Reference 9-24), as required for a limited change.

With the implementation of this requested license amendment, there will be some changes to plant response to transient and accident scenarios. Therefore, in order to support the licensed operator training, a separate software model will be developed. This new model will replicate the appropriate differences caused by the larger SGs and PUR. The new model will be used by operations to demonstrate the plant differences, as well as act as a tool for development of applicable startup test procedures. This model will not replace the Unit 1 model for normal examination/evaluation and the simulator fidelity will not be impacted.

Section 9.11.4 Impact on Training

As discussed above, the Operations Department staff will be trained on the required modifications, Technical Specification changes, procedural changes, as well as the changes in plant response to transients and accident scenarios for PUR. This will be done to assure that the Operations Department staff receives the required training to ensure safe and continued operations of Palo Verde Unit 2.

In addition, the systematic approach to training will be utilized to identify other required training needs for Operations and other departments, to assure that appropriate staff are qualified and trained to the level required to support safe and continued operations of Palo Verde Unit 2.

This training will also incorporate applicable lessons learned from other utilities that have implemented a PUR.

Section 9.12 Testing

The PUR and larger SGs result in design changes/modifications. In order to perform the required retests to verify the design basis parameters and to verify continued safe operation, an Integrated Startup Test Plan will be developed.

The results of the integrated Startup Test Plan will be incorporated in a Startup Test Report and submitted in accordance with Technical Requirements Manual (TRM) Section T5.0.600.2 (Reference 9-23). The Startup Test Plan will verify acceptable results from both individual modification retests, as well as integrated systems tests, as required.

Some of the tools that will be used to develop the Startup Test Plan are:

- Review of all modifications requiring retests.
- Review of the UFSAR Chapter 14.
- Benchmarking of other facilities.
- Review of the Combustion Engineering System 80 CESSAR Chapter 14 (Reference 9-52).

Required test procedures will be developed before implementation of this license amendment.

Section 9.13 Human Factors

NUREG-0700 specifies guidance on human factors (Reference 9-4). This guidance applies to components located in the control room and the remote shutdown facility. Human factors requirements ensure that the design of the control room workspace, instrumentation, controls, and other equipment accounts for both system demands and operator capabilities.

Control room components will be replaced and/or installed because of PUR. The function of control room components is two-fold:

1. provide the system status information, control capabilities, feedback and performance aids necessary for control room operators to accomplish their tasks effectively and
2. eliminate, or acceptably minimize, characteristics of control room instrumentation, control, other equipment, and physical arrangements that may detract from operator performance.

Modifications that affect control room components will be installed per NUREG-0700. Specifically, the ESPS temperature indicators will include the replacement of temperature indicators in the control room. The new indicators meet all of the human factors requirements listed above.

Human factors design requirements for PUR are acceptable.

Section 9.14 High Energy Line Breaks

The PUR does not result in any changes to the high energy lines discussed in UFSAR Tables 3.6-1 and 3.6-2. Additionally, there is no impact to the methods of protection of safety-related systems from High Energy Line Breaks (HELBs), as discussed in UFSAR Table 3.6-3.

The bounding HELB events inside containment and MSLB in the MSSS are discussed in Section 6.2 of this submittal.

With the exception of the MSLB in the MSSS (see Section 6.2.4), the HELB events and consequences outside containment, as reported in UFSAR Sections 3.6.1, are bounded by the existing condition.

The erosion/corrosion program discussed in Section 9.15 ensures that the potential for a secondary system HELB is not increased because of increases in secondary system flowrates.

Section 9.15 Erosion/Corrosion Program

The secondary system operational flowrates will increase as discussed in Section 4.2, Section 8.3, and Section 8.8. The existing Erosion/Corrosion Program inspection acceptance criteria will be maintained. The erosion/corrosion program will be updated to incorporate the revised secondary system flowrates to ensure that the piping is inspected at the required frequencies. The Erosion/Corrosion Program will be updated as part of implementation of this license amendment request.

Section 9.16 Flooding

Section 9.16.1 Containment Sump pH and Containment Flooding

The increased primary side SG volume will increase the post-LOCA containment sump pH and flood level. The increase in volume is small when compared to the total volume of water in the containment post-LOCA. This increase does not alter the conclusion of the existing analyses that the containment flood level remains below 91’ 0”, and the post-LOCA sump pH remains between a minimum of 7.0 and a maximum of 8.5, as reported in the UFSAR. The actual margin available for the post-Recirculation Actuation Signal (RAS), HPSI, Low Pressure Safety Injection (LPSI), and CSS pump minimum available Net Positive Suction Head (NPSH) will be increased because of the increased primary side SG volume.

Containment sump screen plugging was evaluated for PUR. It was concluded that none of the input parameters or assumptions used to evaluate sump screen plugging are affected by PUR. Therefore, implementation of PUR is bounded by the existing condition. Affects of the increased/new insulation design on the SGs will be evaluated through the 10 CFR Part 50.59 process.

Section 9.16.2 Outside Containment Flooding

Effect of the PUR on the building flooding is bounded by the existing analysis as discussed in UFSAR Section 3.6.1 or the response to Question 3A.20. For areas outside the containment, the existing flooding calculations use the worst-case flow from high or moderate energy piping systems. The parameters used in these analyses have been reviewed and verified to bound the expected PUR conditions.

The new SG has a larger secondary side volume and results in an increase in the inventory available following a Feedwater Line Break (FWLB). This increase in available inventory remains bounded by the inventory considered in the existing FWLB flooding analysis.

The existing MSSS flooding evaluation is not affected by the changes in the FW flowrates. The existing analysis conservatively considers a condenser inventory and FW flowrates that bound the expected PUR values.

Section 9.17 Computer Code Applications

Table 9.17-1 lists the safety related computer codes used to support this license amendment. Included is a list of supporting documents/references that have previously been supplied to/by the NRC for review of the codes.

Table 9.17-1
Computer Codes

(Page 1 of 4)

Code	Report Section	Analytical Description	Reference
ANSYS	Section 5.2, Section 5.3, Section 5.4, and Section 5.5	ANSYS analysis capabilities include static and dynamic; elastic, plastic, creep and swelling; small and large deflections; steady state and transient heat transfer and fluid flow (for core coolability).	Reference 9-27 Described in UFSAR Section 3.9.
CE-DAGS	Section 5.4	DAGS (Dynamic Analysis of Gapped Structure) performs a piecewise linear direct integration solution of the coupled equations of motion of a three dimensional structure.	Reference 9-28 Described in UFSAR Section 3.9.
CEFLASH-4A	Section 6.2	CEFLASH-4A is used to calculate transient conditions resulting from a flow line rupture in a water/steam flow system (LOCA).	Reference 9-29 Described in UFSAR Sections 3.9, 6.2, and 6.3.
CEFLASH-4B	Section 5.2	CEFLASH-4B predicts the reactor pressure vessel pressure and flow distribution during the subcooled and saturated portion of the blowdown period of a LOCA (LOCA motion). CEFLASH-4B is a modified version of the CEFLASH-4A code.	Reference 9-30 Described in UFSAR Section 3.9.
CENTS	Section 6.3, Section 6.4, and Section 9.6	Replaces CESEC III, used to simulate the NSSS	Reference 9-31 Replaces CESEC. Described in UFSAR Section 15.
CEPAN	Section 7.3	The CEPAN computer model is used for stress analysis for determining clad collapse resistance.	Reference 9-32 Described in UFSAR Section 4.2.
CESEC III	N/A	CESEC III version of CESEC computer program is currently used to simulate the NSSS in the PVNGS UFSAR. CESEC has been replaced by CENTS with the exception of a portion of the CEA ejection analysis.	Reference 9-33 Described in UFSAR Section 15.

Table 9.17-1
Computer Codes

(Page 2 of 4)

Code	Report Section	Analytical Description	Reference
CESHOCK	Section 5.2	CESHOCK solves for the response of structures that can be represented by lumped-mass and spring systems and are subjected to a variety of arbitrary type loadings.	Reference 9-34 Described in UFSAR Section 3.9.
CETOP-D	Section 6.3 and Section 7.1	CETOP-D is used to compute the thermal margin analysis in the core.	Reference 9-35 Described in UFSAR Sections 4.4 and 15.
COAST	Section 6.3	COAST is used to calculate the reactor coolant flow coastdown transient.	Reference 9-36 Described in UFSAR Section 15.
CONTRANS2	Section 6.2	CONTRANS2 is used for calculating containment backpressure during LOCA for ECCS evaluation.	Reference 9-37 Described in UFSAR Section 15.6.
COPATTA	Section 6.2 and Section 8.16	The COPATTA model predicts both the pressure and temperature within the containment regions and the temperatures in the containment structures.	Reference 9-38 Described in UFSAR Section 6.2.
CPC FORTRAN	Section 6.3 and Section 7.1	Core protection calculator FORTRAN model.	Reference 9-39 Described in UFSAR Sections 7 and 15.
FATIGTS	Section 5.5	FATIGTS has been used for the fatigue analysis of the SG tubesheet.	Analysis code provided by the Replacement Steam Generator (RSG) manufacturer to demonstrate ASME Code compliance. Methodology will be described in UFSAR Section 3.9.
FATES3B	Section 7.3	Steady-state fuel temperatures are determined by FATES3B.	Reference 9-40 Described in UFSAR Sections 4.2 and 4.3.

Table 9.17-1
Computer Codes

(Page 3 of 4)

Code	Report Section	Analytical Description	Reference
FLOOD3	Section 6.2	The FLOOD3 (updated version of FLOOD-MOD2) hydraulics code calculates flowrates and pressures.	Reference 9-41 Described in UFSAR Section 6.2.
HERMITE	Section 6.3	HERMITE is used to predict the reactor core response during a LOF.	Reference 9-42 Described in UFSAR Sections 4 and 15.
HRISE	Section 6.3	HRISE is a thermal-hydraulic fuel code.	Reference 9-43 Described in NUREG-0852, Appendix H
LOCADOSE	Section 6.4	LOCADOSE is used for dose assessment in accordance with the guidelines of Regulatory Guide 1.4, Regulatory Guide 1.77 (Reference 9-16), and SRP Section 15.6.5.	Reference 9-10 Described in UFSAR Section 15.6.
MEC-21	Section 5.4.2	MEC-21 is a program used for flexibility analysis of the main loop piping and components.	Reference 9-44 Described in UFSAR Section 3.9.
ORIGEN-S	Section 6.4 and Section 6.5	Code to quantify fission product inventories.	Reference 9-45 Described in UFSAR Section 15.7.
PCFLUD	Section 6.2 and Section 9.14	Pressure-temperature analyses, used to establish both the structural and the environmental design parameters for components.	Reference 9-46 Described in UFSAR Section 3.6.
Principal Stress Program	Section 5.5.2	This code sums stresses for three load conditions and computes principal stress intensity, stress intensity range, and fatigue usage factor. This program is used in the fatigue analysis of SG components.	Reference 9-10 and Reference 9-52 Described in UFSAR Section 3.9.
PWR-Gale	Section 7.6	Used to estimate releases of radioactive materials in gaseous effluents.	Reference 9-47 Described in NUREG-0017.

Table 9.17-1
Computer Codes

(Page 4 of 4)

Code	Report Section	Analytical Description	Reference
RANGE	Section 5.5	RANGE program takes the stress components and computes the maximum stress intensity range for axisymmetric shells and nozzles.	Analysis code provided by the RSG manufacturer to demonstrate ASME Code compliance. Methodology will be described in UFSAR Section 3.9.
RANGETS	Section 5.5	RANGETS is used for evaluation of the stress intensity range for the new SG tubesheet.	Analysis code provided by the RSG manufacturer to demonstrate ASME Code compliance. Methodology will be described in UFSAR Section 3.9.
SGNIII	Section 6.2	Used to determine the effect of MSLB on containment pressure analyses.	Reference 9-48 Described in UFSAR Section 6.2.
STRIKIN-II	Section 6.3	Used for LBLOCA calculations.	Reference 9-48 Described in UFSAR Sections 6.3 and 15.
STRUDL DYNAL	Section 5.4	STRUDL DYNAL is designed as a structural information system.	Reference 9-50 Described in UFSAR Section 3.
TORC	Section 6.3 and Section 7.1	TORC program is used to simulate the fluid conditions within the reactor core.	Reference 9-51 Described in UFSAR Sections 4.4 and 15.

Section 9.18 References

Reference 9-1 Letter LD-83-074 dated August 12, 1983, Natural Circulation Cooldown Reanalysis for CESSAR-F.

Reference 9-2 APS Letter ANPP-40069-JGH/BJA/98.05, dated February 9, 1987, PVNGS Natural Circulation Cooldown Test Report (CENP Document No. PS-2000-0034, Revision 000).

Reference 9-3 USNRC Letter dated April 18, 1988, Evaluation of the Natural Circulation Cooldown Capability for Palo Verde (TAC Nos. 56647 and 56648) (CENP Document No. PS-2000-0033, Revision 000).

Reference 9-4 NUREG-0700 dated September, 1981, “Guidelines for Control Room Design Reviews.”

Reference 9-5 Regulatory Guides 1.143, Revision 0, July 1978, “Design Guidance for Radioactive Waste Management Systems Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

Reference 9-6 Combustion Engineering Owners Group (CEOG) letter to the NRC, “Submittal of Topical Report CE NPSD-1157 Regarding Elimination of PASS Requirements,” dated May 5, 1999.

Reference 9-7 NUREG-0737, November 1980, Clarification of TMI Action Plan Requirements,” and Supplement 1 to NUREG-0737, December 17, 1982, “Requirements for Emergency Response Capability,” (Generic Letter 82-33).

Reference 9-8 U. S. Nuclear Regulatory Commission Standard Review Plan (SRP), NUREG-75/087, Revision 1, November 1975.

Reference 9-9 Regulatory Guide 8.8, Revision 3, June 1978, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.”

Reference 9-10 Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Revision 11, June 2001.

Reference 9-11 Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 108 to Facility Operating License No. NPF-41, Amendment No. 100 to Facility Operating License No. NPF-51, and Amendment No. 80 to Facility Operating License No. NPF-74, Arizona Public Service Company, et al, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Docket Nos. STN 50-528, STN 50-529, and STN 50-530, dated May 23, 1996.

Reference 9-12 NUREG-0857, Safety Evaluation Report related to the operation of PVNGS Units 1, 2, and 3, Supplement 4.

Reference 9-13 NUREG 0841, dated, February 1, 1982, “Final Environmental Statement Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2 and 3.”
NUREG-0036, dated, February 1, 1978, “Final Environmental Statement Related to the Construction of Palo Verde Nuclear

Generating Station, Units 1, 2, and 3."
NUREG-75/078, dated September, 1975, "Final Environmental Statement (FES)."

Reference 9-14 Regulatory Guide 1.4, "Assumptions Used for Evaluating the Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974.

Reference 9-15 Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 1, July 1976.

Reference 9-16 Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Revision 0, May 1974.

Reference 9-17 ANSI N510-1980, "Testing of Nuclear Air Treatment Systems."

Reference 9-18 NRC Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance, original issue through supplement 7.

Reference 9-19 NRC Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, dated August 17, 1995.

Reference 9-20 Code of Federal Regulations, Title 10, Part 20 (old), Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure Effluent Concentrations; Concentrations for Release to Sewerage."
Code of Federal Regulations, Title 10, Part 50, Section 50.43, "Additional Standards and Provisions Affecting Class 103 Licenses for Commercial Power."
Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
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Code of Federal Regulations, Title 10, Part 100, "Reactor Site Criteria."

Reference 9-21 IEEE 323-1974, “General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations.”

Reference 9-22 Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Technical Specifications, through Amendment No. 137, December 6, 2001 (Amendment No. 136 not yet implemented).

Reference 9-23 Technical Requirements Manual (TRM), for Palo Verde Nuclear Generating Station, Units 1, 2, 3, Revision 16, August 30, 2001.

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Reference 9-25 NRC Regulatory Issue Summary 2001-02, January 18, 2000.

Reference 9-26 Regulatory Guide 1.97, Revision 2, December, 1975, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident.”

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Reference 9-31 “Technical Manual for the CENTS Code,” CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].“
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Section 10 ACRONYMS

ACRONYM	DEFINITION
λ	purification constant
$\Delta\rho$	change in density
χ/Q	atmospheric dispersion factor
°F	degrees Fahrenheit
ΔP	change in pressure
AC	Alternating Current
ADV	Atmospheric Dump Valve
AFAS	Auxiliary Feedwater Actuation System
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating Procedure
AOR	Analyses of Record
AOV	Air Operated Valve
APS	Arizona Public Service
ASGT	Asymmetric Steam Generator Transient
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BDAS	Boron Dilution Alarm System
Bhp	Brake horse power
BLPB	Branch Line Pipe Break
BOC	Beginning of Core life
BOP	Balance of Plant
BPC	Bechtel Power Corporation
BTP	Branch Technical Position
Btu	British thermal unit
BWR	Boiling Water Reactor
cal	calories
CD	Condensate System
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CEDMCS	Control Element Drive Mechanism Control System
CENTS	Combustion Engineering Nuclear Transient Simulator
CEOG	Combustion Engineering Owners Group
CESSAR	Combustion Engineering Standard Safety Analysis Report
cfm	cubic feet per minute
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
CIAS	Containment Isolation Actuation Signal
cm	centimeter

ACRONYM	DEFINITION
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CPIAS	Containment Purge Isolation Actuation Signal
CREFAS	Control Room Essential Filtration Actuation Signal
CRVIAS	Control Room Ventilation Isolation Actuation Signal
CSAS	Containment Spray Actuation Signal
CSS	Containment Spray System
CST	Condensate Storage Tank
CTI	Cooling Tower Institute
CVCS	Chemical And Volume Control System
CWP	Control Element Assembly Withdrawal Prohibit
DAFAS	Diverse Auxiliary Feedwater Actuation System
DBA	Design Basis Accident
DBE	Design Basis Event
DBLLOCUS	Double-Ended Break of the Letdown Line Outside Containment Upstream of the letdown line control valve
dc	direct current
DEDLSB	Double-Ended Discharge Leg Slot Break
DEHLSB	Double-Ended Hot Leg Slot Break
DEQ	Dose Equivalent
DESLSB	Double-Ended Suction Leg Slot Break
DF	Decontamination Factor
DFWCS	Digital Feedwater Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTT	Diverse Turbine Trip
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECWS	Essential Cooling Water System
EDG	Emergency Diesel Generator
EFPD	Effective Full Power Day
EFPH	Effective Full Power Hour
EHC	Electro-Hydraulic Control
EIS	Electronic Isolation Signal
EOC	End of Core life
EOF	Emergency Operating Facility
EOL	End of Life
EOP	Emergency Operating Procedure
EQ	Equipment Qualification
ERFDADS	Emergency Response Facility Data Acquisition Display System
ESD	Excess Steam Demand
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESPS	Spray Pond
FBEVAS	Fuel Building Essential Ventilation Actuation Signal
FES	Final Environmental Statement
FFBT	Fast Transfer Bus
FLCEAD	Full-Length Control Element Assembly Drop
ft	feet or foot

ACRONYM	DEFINITION
FTC	Fuel Temperature Coefficient
FW	Feedwater
FWCV	Feedwater Control Valve
FWIV	Feedwater Isolation Valve
FWLB	Feedwater Line Break
g	acceleration due to gravity
gal	Gallon
GDC	General Design Criterion
GIS	Accident Generated Iodine Spiking Factor
gm	Gram
gpm	gallons per minute
GTG	Gas Turbine Generator
GWD	Gegawatt Days
HA	Auxiliary Building Heating Ventilation and Air Conditioning
HEI	Heat Exchanger Institute
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HFP	Hot Full Power
HJTC	Heated Junction Thermocouple
HJTCIFA	Heated Junction Thermocouple Instrumentation Flange Assembly
HLPT	High Log Power Trip
HLR	Head Lift Rig
HPPT	High Pressurizer Pressure Trip
HPSI	High Pressure Safety Injection
hr	hour
HT	Turbine Building Heating Ventilation and Air Conditioning
HVAC	Heating Ventilation and Air Conditioning
Hz	Hertz
HZP	Hot Zero Power
I&C	Instrumentation & Controls
ICI	In-Core Instrumentation
ICRP	International Commission on Radiological Protection
ID	Inadvertent Deboration
ISLOCA	Interfacing System Loss-of-Coolant Accidents
In	Inch(es)
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
Kips	1000 pounds of dead weight load
Ksi	Kips per square inch
kV	Kilovolt
kW	Kilowatt
LBB	Leak Before Break
LBLOCA	Large Break Loss-of-Coolant Accident
lb_m	pounds mass
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOF	Loss of Flow
LOFW	Loss of Feedwater

ACRONYM	DEFINITION
LOP	Loss of Offsite Power
LPD	Local Power Density
LPMS	Loose Parts Monitoring System
LPSI	Low Pressure Safety Injection
LPZ	Low Population Zone
LSGLT	Low Steam Generator Level Trip
LSGP	Low Steam Generator Pressure
LTOP	Low Temperature Over Pressure
M&E	Mass and Energy
MCL	Main Coolant Loop
MCLB	Main Coolant Line Break
MFIV	Main Feedwater Isolation Valve
MFLB	Main Feedwater Line Break
MFW	Main Feedwater
min	minute
ml	milliliter
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MRP	EPRI Materials Reliability Project
MSCU	Modified Statistical Combination of Uncertainties
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSR	Main Steaming Rate
MSSS	Main Steam Support Structure
MSSV	Main Steam Safety Valve
MT	Main Turbine
MTC	Moderator Temperature Coefficient
MTU	Metric Ton of Uranium
MVA	Megavolt-Ampere
MVAR	Megavolt-Ampere Reactive
MWD	Megawatt Days
MW_e	Megawatts Electrical
MW_t	Megawatts Thermal
N/A	Not Applicable
NC	Nuclear Cooling Water
NOP	Normal Operating Pressure
NOT	Normal Operating Temperature
NR	Narrow Range
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OSG	Original Steam Generator
P_a	peak calculated containment internal pressure for the design basis LOCA
PASS	Post Accident Sampling System
PDIL	Power Dependent Insertion Limit
PIS	Pre-Existing Iodine Spiking Factor
PLCS	Pressurizer Level Control System
PLHGR	Peak Linear Heat Generation Rate

ACRONYM	DEFINITION
POL	Power Operating Limit
PPCS	Pressurizer Pressure Control System
ppm	parts per million
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
psia	pounds per square inch absolute
psid	pounds per square inch differential
psig	pounds per square inch gauge
PSV	Pressurizer Safety Valve
PTS	Pressurized Thermal Shock
PUR	Power Uprate
PVNGS	Palo Verde Nuclear Generating Station
PW	Plant Cooling Water
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QSPDS	Qualified Safety Parameter Display System
RAR	Reload Analysis Report
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDT	Reactor Drain Tank
REM	Roentgen Equivalent Man
RHR	Residual Heat Removal
RPCS	Reactor Power Cutback System
RPI	Reload Process Improvement
rpm	revolutions per minute
RPS	Reactor Protection System
RRS	Reactor Regulation System
RSE	Reload Safety Evaluation
RSG	Replacement Steam Generator
RSPT	Reed Switch Position Transmitter
RTD	Resistance Temperature Detector
R-t-P	Return to Power
RV	Reactor Vessel
RVI	Reactor Vessel Internals
RWT	Refueling Water Tank
SAFDL	Specified Acceptable Fuel Design Limit
SAM	Seismic Anchor Movement
SBCS	Steam Bypass Control System
SBLOCA	Small Break Loss-of-Coolant Accident
SBO	Station Blackout
SC	Secondary Chemical Control
scf	standard cubic feet
scfm	standard cubic feet per minute
SCM	Subcooling Margin
SCU	Statistical Combination of Uncertainties
SCS	Shutdown Cooling System
sec	second (time)
SDM	Shutdown Margin

ACRONYM	DEFINITION
SFPCC	Spent Fuel Pool Cooling and Cleanup
SFWLB	Small Feedwater Line Break
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SGTRLOP	Steam Generator Tube Rupture, with a Loss of AC Power
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIRCP	Startup of an Inactive Reactor Coolant Pump
SIS	Safety Injection System
SIT	Safety Injection Tank
SLB	Steam Line Break
SOV	Solenoid Operating Valve
SPS	Supplemental Protection System
SR	Surveillance Requirement
SRP	Standard Review Plan
SRSS	Square-Root-of-the-Sum-of-the-Squares
SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
TAV	Turbine Admission Valve
T_{ave}	average reactor coolant temperature
TBV	Turbine Bypass Valve
TC	Turbine Cooling Water
T_{cold}	cold leg reactor coolant temperature
temp	temperature
TGCS	Turbine-Generator Control System
T_{hot}	hot leg reactor coolant temperature
TRM	Technical Requirements Manual
TSC	Technical Support Center
TSTF	Technical Specification Task Force
UFSAR	Updated Final Safety Analysis Report
UGS	Upper Guide Structure
UHS	Ultimate Heat Sink
VCT	Volume Control Tank
VOPT	Variable Over-Power Trip
vs.	versus
VWO	Valves Wide Open
W CENP	Westinghouse Combustion Engineering Nuclear Production
WR	Wide Range
WRSO	Worst Rod Stuck Out