CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

DEC 1 4 2000

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SERIAL: HNP-00-175
10CFR50.4
10CFR50.59(c)
10CFR50.90

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
LICENSE AMENDMENT APPLICATION
POWER UPRATE

Dear Sir or Madam:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L) hereby submits an application to amend the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to allow plant operation at an increased (uprated) reactor core power level of 2900 megawatts thermal (MWt). The proposed change to Facility Operating License NPF-63 and the Technical Specifications, Appendix A to the Operating License, will revise the specified maximum power level and the definition of rated thermal power. A Technical Specification Bases change associated with this amendment application is also provided herein. The proposed changes will permit HNP operation at the uprated core power level of 2900 MWt when the unit is restarted after the tenth refueling outage.

CP&L recently submitted proposed Technical Specification changes required for steam generator replacement at HNP (ref. letter SERIAL: HNP-00-142, dated October 4, 2000). The analyses and evaluations performed to support the Technical Specification changes associated with steam generator replacement also support the proposed power uprate, because the majority of the analyses and evaluations were performed at the uprated core power level of 2900 MWt.

CP&L has drawn upon the experience of Westinghouse Electric Company in support of power uprate projects at other nuclear power plants. In particular, the license amendment applications for the Joseph M. Farley Nuclear Plant (FNP) and the V. C. Summer Nuclear Station (VCSNS) were used as reference models for the HNP power uprate. The technical bases for the proposed changes address the substantive issues presented by NRC requests for additional information, licensee responses, and NRC safety evaluations associated with previous power uprate license amendment applications and uprate programs.

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Additional analyses and evaluations required to address power uprate conditions have been performed and are summarized in Enclosure 6 to this letter. These include: (1) analysis of a Large Break LOCA (LBLOCA) accident, (2) analysis of Inadvertent Operation of the Emergency Core Cooling System (IOECCS) accident, and (3) an evaluation of the Spent Fuel Pool Cooling and Cleanup System (FPCCS) to support power uprate conditions. The LBLOCA and IOECCS analyses presented in Enclosure 6 supersede the corresponding analyses submitted for the steam generator replacement, and are submitted herein as the technical basis for the proposed power uprate changes. The evaluation of FPCCS included in Enclosure 6 is submitted as a supplement to the FPCCS analysis submitted for the steam generator replacement and takes into account the operating conditions expected after implementation of power uprate.

Enclosure 1 provides background information, a description of the proposed changes, and the basis for the proposed changes to the Facility Operating License and Technical Specifications.

Enclosure 2 provides, in accordance with 10 CFR 50.91(a), the basis for the determination that the proposed changes to the Facility Operating License and Technical Specifications do not involve a significant hazards consideration. CP&L has determined the proposed changes will not significantly increase the amount of any effluent that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure.

Enclosure 3 provides the page change instructions for the proposed changes to the Facility Operating License and the Technical Specifications.

Enclosure 4 provides the marked-up and retyped pages for the proposed changes to the Facility Operating License and the Technical Specifications.

Enclosure 5 provides an environmental evaluation which determines the proposed license amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement is required for approval of this license amendment application.

Enclosure 6 provides the additional analyses and evaluations required for power uprate.

A copy of these proposed changes is being sent to the North Carolina Department of Environment and Natural Resources (NCDENR) in accordance with 10 CFR 50.91 (b)(1).

CP&L requests NRC review and approval of the proposed license amendment by September 1, 2001. This license amendment issuance date is necessary to support unit restart from refueling outage 10, which is scheduled to begin in September 2001.

Please refer any questions regarding the enclosed information to Mr. Eric McCartney at (919) 362-2661.

Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

Darlene S. Yarborough
Notary (Seal)

My commission Expires: 2-21-2005

KWS/kws

Enclosures:

1. Basis for Proposed Changes and Description of Amendment Request
2. No Significant Hazards Consideration Determination (10 CFR 50.92)
3. Page Change Instructions
4. Operating License and Technical Specification Page Changes
5. Environmental Evaluation (10 CFR 51.22)
6. Power Uprate Analyses and Evaluations

c: (all with Enclosures)

Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator – Region II

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BASIS FOR PROPOSED CHANGES

Carolina Power & Light Company (CP&L) proposes to revise the Harris Nuclear Plant (HNP) Facility Operating License Section 2.C.(1), Technical Specification page 1-5, and Technical Specification Bases page B 3/4 7-1. These changes are required to implement power uprate and provide the following:

- A revision to item 2.C.(1), Maximum Power Level, of HNP Operating License NPF -63 to reflect the uprated reactor core power level of 2900 MWt.
- A revised definition of Rated Thermal Power (RTP) to reflect the uprated reactor core power level of 2900 MWt. This value represents the total reactor core heat transfer rate from the reactor core to the reactor coolant and does not include heat generated by the reactor coolant pumps.
- Revision to the Technical Specification Bases for the specified valve lift settings and relieving capacities for the Main Steam line Code safety valves.

Proposed Technical Specification changes associated with the installation of replacement steam generators were previously submitted for staff review and approval (ref.: SERIAL: HNP-00-142, dated October 4, 2000). Many of the Technical Specification changes typically expected for a power uprate were included in the above-referenced amendment application associated with steam generator replacement. The majority of the new analyses and detailed evaluations performed to support the installation of replacement steam generators also support uprating the licensed reactor core power from 2775 MWt to 2900 MWt (NSSS power from 2787.4 MWt to 2912.4 MWt).

CP&L has completed a steam generator replacement and Power Uprate (SGR/Uprate) analysis and licensing project to support the re-licensing of HNP with the following conditions:

- Original Model D4 (pre-heater type) steam generators replaced with Westinghouse Model Delta 75 (feed ring type) steam generators
- Uprating the licensed reactor core power from 2775 MWt to 2900 MWt (NSSS power from 2787.4 MWt to 2912.4 MWt)

- Analyses and evaluations, with the exception of the Steam Generator Tube Rupture (SGTR) event re-analysis and, as noted in Enclosure 6, the Large Break Loss of Coolant Accident (LBLOCA) and analysis of Inadvertent Operation of the Emergency Core Cooling System (IOECCS) During Power Operation, performed for a reactor vessel average temperature ranging from 572 °F to 588.8 °F. The upper range value of 588.8 °F is the original HNP design basis T_{avg} temperature.

This program was structured consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant." The technical bases, analytical techniques, engineering analyses, and evaluations performed in support of the power uprate changes are documented in the NSSS and BOP licensing reports submitted as enclosures to the amendment application for steam generator replacement (ref.: SERIAL: HNP-00-142, dated October 4, 2000).

The proposed changes to the Operating License and Technical Specifications are supported by overall program results. The results of analyses or evaluations are consistent with and continue to comply with the current HNP licensing basis acceptance requirements.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS (10 CFR 50.92)

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. Carolina Power & Light Company (CP&L) has evaluated the proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92, a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety

The proposed change to the Technical Specifications is categorized as a definition change and is evaluated below. The Technical Specification Bases change is associated with the Main Steam System and is also included below.

I. DEFINITION CHANGE

Technical Specification Section 1.28 – Rated Thermal Power Definition

The change to the reactor core rated thermal power definition from 2775 MWt to 2900 MWt is based on the HNP uprated power value. This change is also applicable to the maximum power level defined in Facility Operating License Section 2.C (1). The revised power level is taken into account in the supporting SGR/Uprate conditions evaluations and analyses; therefore, this change is acceptable.

II. MAIN STEAM SYSTEM CHANGE

Bases 3/4.7.1.1 Turbine Cycle – Safety Valves

The main steam safety valves evaluation results demonstrate the relief capacity is sufficient to limit the maximum steam pressure to less than 110% (ASME Section III) of main steam system (steam generator shell side) design pressure. The updated Technical Specification Bases reflects the evaluation results.

Based on the information presented above and the analyses and evaluations performed for the proposed SGR/Uprate conditions, the following conclusions can be reached with respect to 10 CFR 50.92:

1. The proposed changes do not significantly increase the probability of a previously evaluated accident. The comprehensive analytical efforts performed to support the proposed SGR/Uprate conditions included a review and evaluation of components and systems (including interface systems and control systems) that could be affected by this change. The revised reactor core power value was used as an input to applicable safety analyses and engineering evaluations. Plant systems will continue to function as designed, and performance requirements for these systems have been evaluated and found acceptable. The proposed changes do not initiate or contribute to the initiation of any previously evaluated accident; therefore, the probability of an accident previously evaluated has not been significantly increased. Likewise, the proposed changes do not result in a significant increase in the consequences of an accident previously evaluated. Therefore, the probability or consequences of an accident previously evaluated has not significantly increased.

2. The proposed changes do not create the possibility of a new or different kind of accident than any accident previously evaluated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed changes do not involve a significant reduction in a margin of safety. Analyses and evaluations supporting the proposed SGR/Uprate conditions reflect the increased rated thermal power value. Applicable acceptance criteria continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above information and on the analyses performed to support the SGR/Uprate conditions, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

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PAGE CHANGE INSTRUCTIONS

Remove:	Insert:
Operating License page 4	Operating License page 4
Technical Specification page 1-5	Technical Specification page 1-5
Technical Specification page B 3/4 7-1	Technical Specification page B 3/4 7-1

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FACILITY OPERATING LICENSE NPF-63 CHANGED PAGES

(MARKED-UP PAGE FOLLOWED BY RETYPED PAGE)

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2900 2775 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

Delete

(2) Technical Specifications and Environmental Protection Plan

Delete

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 100 are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)*

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval of a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

Delete

(1) Maximum Power Level

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TECHNICAL SPECIFICATIONS CHANGED PAGES

(MARKED-UP PAGES FOLLOWED BY RETYPED PAGES)

PROCESS CONTROL PROGRAM

Delete

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.
Delete 2900 Add

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

in excess of 105% Add Delete

Add 12.9

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.36 x 10^7 lbs/h which is 111% of the total secondary steam flow of 12.2 x 10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

Delete

maximum calculated Add

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$Hi\varnothing = (100/Q)\frac{(w_s\ h_{fg}\ N)}{K}$$

Where:

$Hi\varnothing$ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

Delete

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

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1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

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The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.36×10^7 lbs/h which is in excess of 105% of the maximum calculated steam flow of 12.9×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$Hi\varnothing = (100/Q)\frac{(w_s\ h_{fg}\ N)}{K}$$

Where:

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Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

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w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

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ENVIRONMENTAL EVALUATION (10 CFR 51.22)

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company (CP&L) has reviewed this amendment request and determined the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement needs to be prepared in connection with the issuance of the amendment. The basis for this determination is as follows:

CP&L has completed a comprehensive engineering analysis and review program in support of steam generator replacement together with power uprate at the Harris Nuclear Plant (HNP). The results of the analyses and evaluations support plant operation at an uprated core power level of 2900 MWt.

CP&L's environmental evaluation performed in support of the proposed steam generator replacement at HNP also considered conditions associated with power uprate and was included within Enclosure 7 to HNP-00-142, dated October 4, 2000, the license amendment application for steam generator replacement. The environmental evaluation for steam generator replacement and power uprate is based upon the assessment of environmental impact reported in the Final Environmental Statement related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2, prepared by the U.S. Nuclear Regulatory Commission (NRC) in October 1983 (FES-OL). The environmental evaluation for steam generator replacement and power uprate addresses the environmental impacts addressed in the FES-OL, including impacts on cooling tower drift, thermal discharges to the ultimate heat sink (i.e., the Auxiliary Reservoir), non-radiological effluents, noise impacts, and terrestrial impacts. In the FES-OL, the NRC staff determined ". . . The Shearon Harris Station can be operated with minimal environmental impact." The environmental impacts associated with steam generator replacement and power uprate are bounded by the impacts reported in the FES-OL for the proposed 2-unit site. It is concluded, therefore, that there is no significant change in the types or significant increase in the amounts of any effluents that may be released off site.

In summary, HNP operation at the increased reactor core power level of 2900 MWt as described herein involves no significant hazards consideration (see Enclosure 2), does not significantly change the types or significantly increase the amounts of any effluents that may be released offsite, and does not significantly increase individual or cumulative occupational radiation exposure. The power uprate activities do not constitute any other type of new and appreciable environmental impact. It is concluded, therefore, that the power uprate activities are benign with respect to the human environment.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement is required.

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POWER UPRATE ANALYSES AND EVALUATIONS

- Large Break LOCA Analysis – Licensing Report Section 6.1 (FSAR Chapter 15.6.3)
- Inadvertent Operation of the Emergency Core Cooling System (IOECCS) Analysis – Licensing Report Section 6.2.29 (FSAR Chapter 15.5.1)
- Spent Fuel Pool Cooling

6.1 LOCA Transients

6.1.1 Large Break LOCA

6.1.1.1 Introduction

The purpose of performing a large break loss of coolant accident (LBLOCA) analysis is to verify the Technical Specification peaking limits and axial dependent power peaking limit (K(z)) curve as well as the adequacy of the Emergency Core Cooling System (ECCS) by demonstrating that the criteria set forth in 10 CFR 50.46(b) are met. The SGR/Uprating conditions supported by this analysis include:

- A rated thermal core power of 2900 MW;
- Model Delta 75 steam generators with 3% tube plugging;
- A total peaking factor (F_Q^T) of 2.41 and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.66; and,
- Vessel average coolant temperatures from 580.8°F to 588.8°F, inclusive.

6.1.1.2 Description of Analyses and Evaluations

A LBLOCA is initiated by a postulated large rupture of the Reactor Coolant System (RCS) piping. A spectrum of double ended cold leg guillotine (DECLG) and single ended cold leg split (SECLS) breaks are considered. The limiting break location is on the pump discharge side of a cold leg pipe.

A Loss of Offsite Power (LOOP) is conservatively assumed to occur at event initiation, resulting in reactor coolant pump (RCP) coastdown. RCP coastdown at event initiation exacerbates flow stagnation and decreases the time to departure from nucleate boiling (DNB), leading to higher cladding temperatures during blowdown.

The break initiates a rapid depressurization of the RCS. A reactor trip signal is issued when the Low Pressurizer Pressure trip setpoint is reached; however, reactor trip and scram are conservatively neglected in the analysis.

A Safety Injection Actuation Signal (SIAS) is issued when the High Containment Pressure setpoint is reached. Due to LOOP, there is a time delay for diesel generator startup in addition to the time delays for high head safety injection (HHSI) and low head safety injection (LHSI) pump startup. The single failure criterion is met by assuming that either one diesel generator fails (loss of diesel generator) or one LHSI pump fails (loss of LHSI pump).

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is injected into the cold legs. This fluid is assumed to bypass the core and flow to the break until the time sustained downflow in the downcomer is predicted to occur.

Following the end of bypass, ECCS fluid from the accumulators and, later, the HHSI and LHSI fills the downcomer and lower plenum until the time liquid level reaches the bottom of the core, which is defined as the beginning of core recovery (BOCREC) time. During the refill period, the fuel rods are cooled by radiation heat transfer.

The reflood period begins at the BOCREC time. ECCS fluid filling the downcomer provides the driving head for reflooding the core. As the quench front moves up the core, steam is generated. Steam binding occurs as the steam flows through the intact and broken loop steam generators and RCPs. It is conservatively assumed that the rotor of each RCP has seized (per Appendix K of 10 CFR 50), which tends to reduce the reflood rate. The fuel rods are cooled by radiation and, eventually, by convection as the quench front moves up the core.

A full break spectrum and axial shape study was performed to determine the limiting break size, break configuration, single failure and axial shape. The analysis considered beginning of cycle (BOC) stored energy with both BOC and middle of cycle (MOC) axial shapes and MOC stored energy with an end of cycle (EOC) axial shape. The effects of gadolinia bearing fuel rods and end of life (EOL) peak average rod exposure were also considered.

The U. S. Nuclear Regulatory Commission (NRC) approved Siemens Power Corporation (SPC) evaluation model (SEM/PWR-98) (Reference 1) was used to perform the analysis. This model incorporates the requirements of 10 CFR 50 Appendix K and consists of the following computer codes:

- RODEX2 for computation of the initial fuel stored energy, fission gas release, and fuel cladding gap conductance;
- RELAP4-EM for the system blowdown and Accumulator/SIS flow calculations;
- CONTEMPT/LT-22, as modified in accordance with NRC Branch Technical Position CSB 6-1, for computation of the containment backpressure;
- REFLEX for the system reflood calculation; and,
- TOODEE2 for the fuel rod heatup calculation during the refill and reflood portions of the LBLOCA event.

The quench time, quench velocity, and carryover rate fraction correlations in REFLEX and the heat transfer correlations in TOODEE2 were based on SPC Fuel Cooling Test Facility (FCTF) data.

The governing conservation equations for mass, energy, and momentum transfer were used, along with appropriate correlations consistent with Appendix K of 10 CFR 50. The reactor core was modeled in RELAP4 with heat generation rates determined from reactor kinetics equations with reactivity feedback, and with actinide and decay heating as required by Appendix K. The input parameters and biasing for the analysis of this event were consistent with the approved methodology.

6.1.1.3 Acceptance Criteria

The acceptance criteria are stated in 10 CFR 50.46(b), specifically:

- The calculated peak fuel cladding temperature does not exceed 2200°F;
- The amount of fuel cladding which reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the core;
- The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation does not exceed 17% during or after quenching; and,
- The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

6.1.1.4 Results

Calculations were performed for a spectrum of DECLG and SECLS break sizes. Three axial power shapes were analyzed for each break size to bound the limiting peak cladding temperature (PCT) and to verify the K(z) curve. Table 6.1.1-1 summarizes the results of the break spectrum and axial shape study.

The results of these calculations indicate that the 0.8 DECLG break with the MOC axial shape, BOC stored energy and loss of LHSI pump single failure was the limiting case. The hot rod results, event times and transient plots for the limiting case are shown in Tables 6.1.1-2 and 6.1.1-3 and Figures 6.1.1-1 through 6.1.1-19.

Additional calculations were performed to address peak rod average exposures to EOL and the effects of gadolinia upon PCT. The results of these calculations are bounded by the limiting case.

6.1.1.5 Conclusions

Consistent with the current licensing basis, the results demonstrate that 10 CFR 50.46(b) acceptance criteria are met for SGR/Uprating conditions. The analysis for this event supports operation with the Delta 75 replacement steam generators at a core power of 2900 MWt with nominal primary T_{avg} at full power from 580.8°F to 588.8°F for steam generator tube plugging from 0% to 3%. The analysis also bounds operation with the Delta 75 replacement steam generators at the currently licensed core power of 2775 MWt with nominal primary T_{avg} at full power from 580.8°F to 588.8°F for steam generator tube plugging from 0% to 3%.

6.1.1.6 References

1. EMF-2087(P)(A), SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications, Siemens Power Corporation, June 1999.

Table 6.1.1-1 Summary of PCT Results

Single Failure Assumption	Break Type	Break Size	Peak Clad Temperature (°F)		
			BOC	MOC	EOC
Loss of LHSI Pump	DECLG	1.0	1783	2084	2010
		0.8	1786	**2090**	2013
		0.6	1725	2061	1987
		0.4	1657	1962	1915
	SECLS	1.0	1782	2078	2000
		0.8	1680	1996	1928
Loss of Diesel Generator	DECLG	1.0	-	2066	1993
		0.8	-	2073	1996

Table 6.1.1-2 Summary of Results for 0.8 DECLG Break With MOC Axial Shape and Loss of LHSI Pump Single Failure

PCT	
Temperature	2090°F
Time	149.32 sec
Elevation	11.125 ft
Hot Rod Burst	
Time	39.87 sec
Elevation	8.875 ft
Channel Blockage Fraction	0.39
Metal Water Reaction	
Local Maximum	7.58%
Elevation of Local Maximum	11.125 ft
Core Maximum	< 1.0%

Table 6.1.1-3 Calculated Event Times for 0.8 DECLG Break With MOC Axial Shape and Loss of LHSI Pump Single Failure

Event	Time [sec]
Begin Analysis	0.0
Break Opened	0.05
SIAS Issued	1.15
Start of Broken Loop Accumulator Injection	2.90
Start of Intact Loop Accumulator Injection	12.08
End Of Bypass (Beginning of Refill)	19.31
Broken Loop Accumulator empties	24.64
Begin of Core Recovery (Beginning of Reflood)	30.10
Start of HHSI and LHSI	30.16
Intact Loop Accumulator empties	33.64
Fuel Cladding Rupture Occurred	39.87
PCT Occurred	149.32

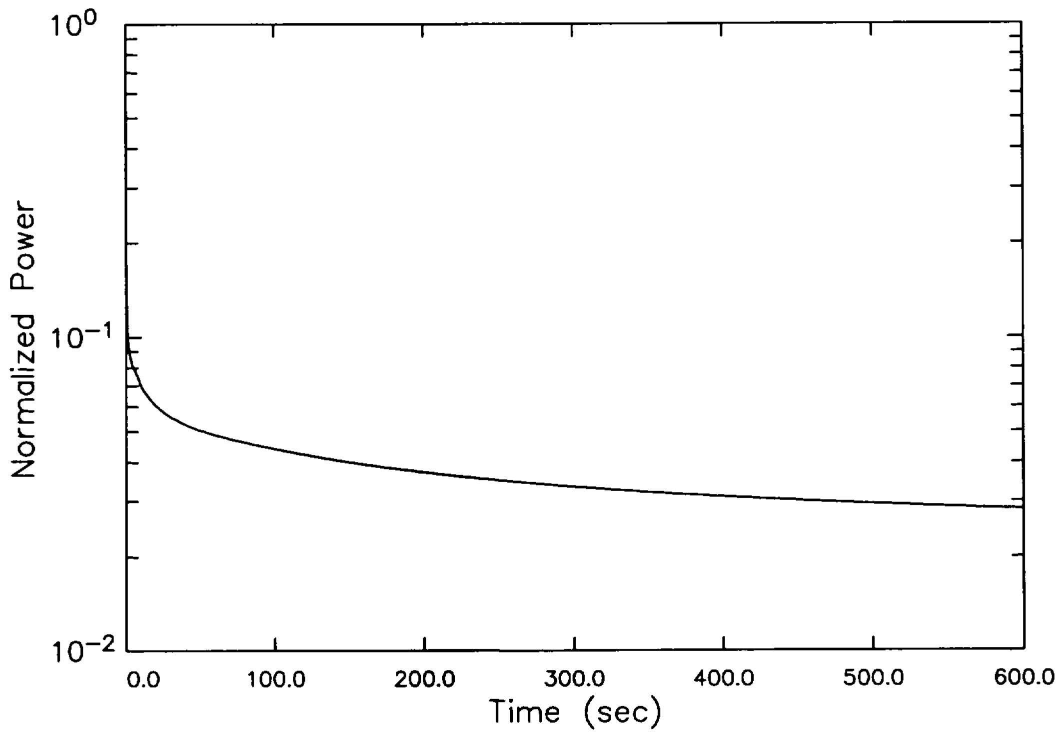

Figure 6.1.1-1 Normalized Power

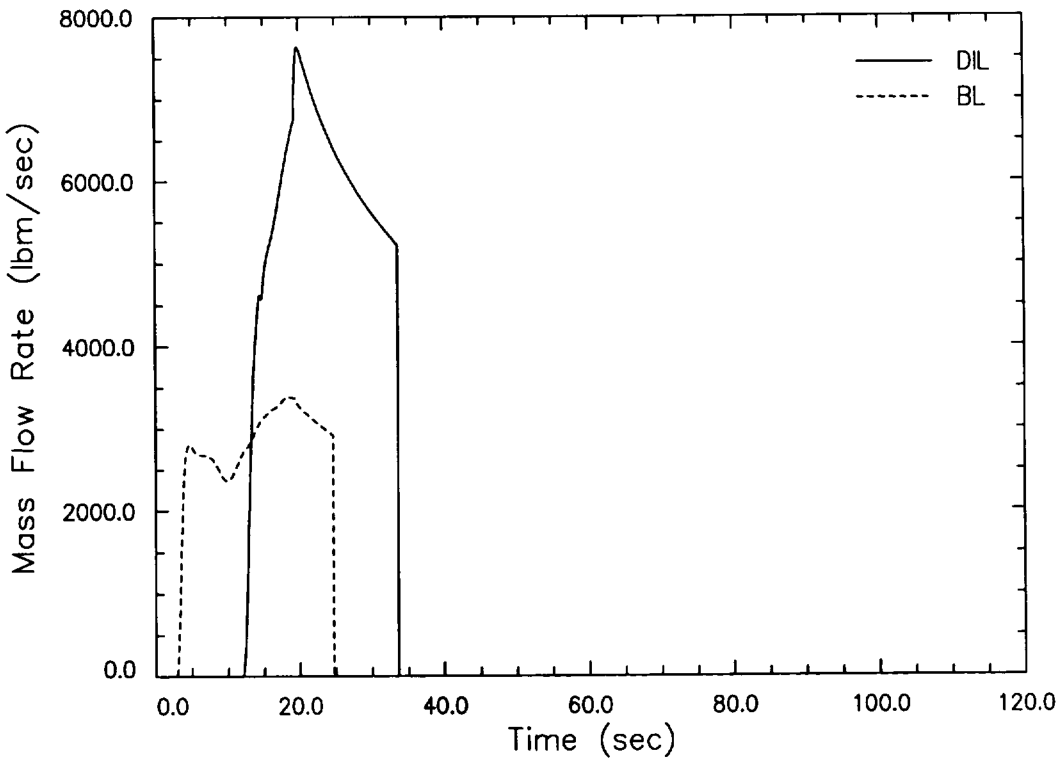

Figure 6.1.1-2 Accumulator Discharge Rates

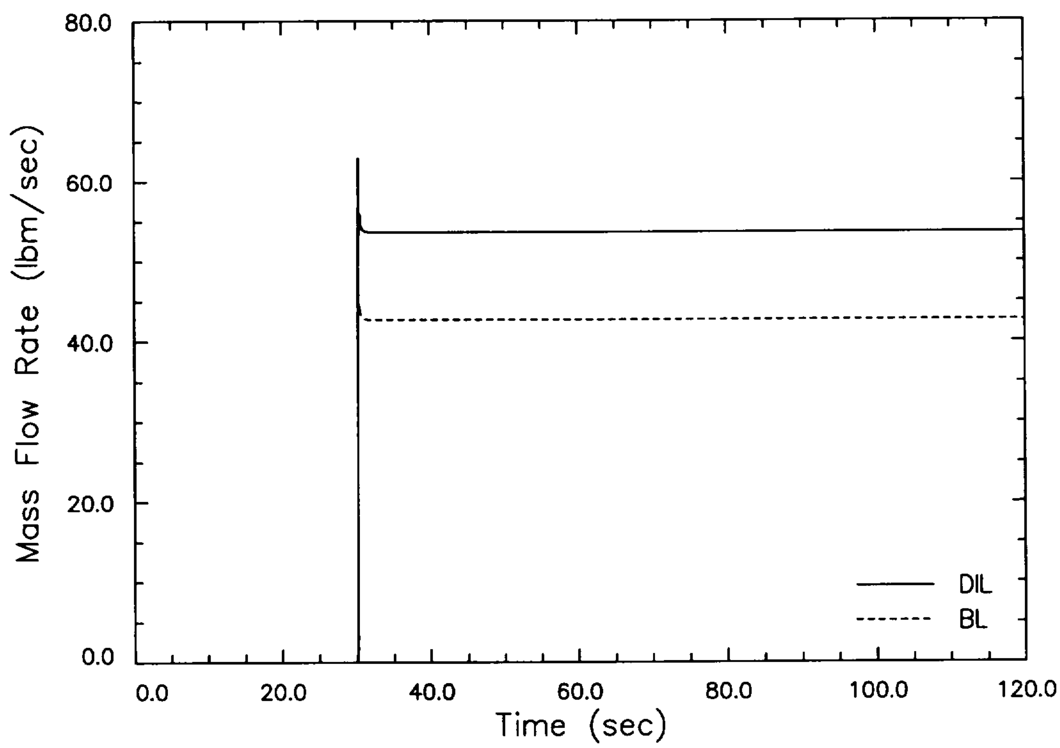

Figure 6.1.1-3 HHSI Flow Rates

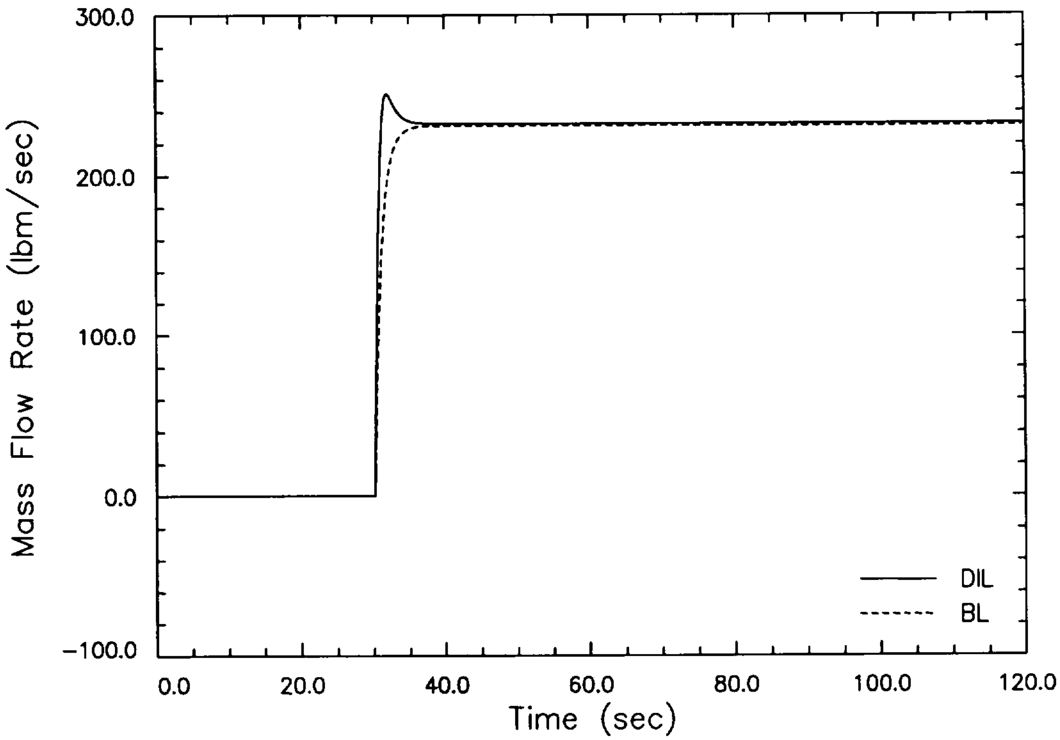

Figure 6.1.1-4 LHSI Flow Rates

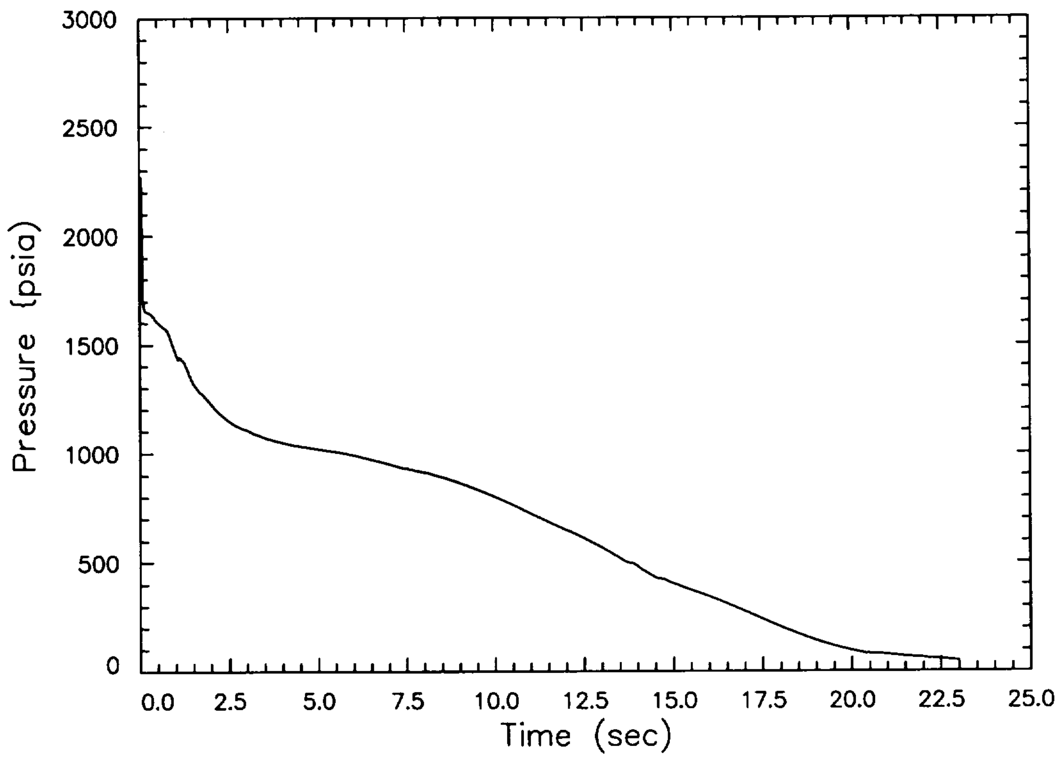

Figure 6.1.1-5 Upper Plenum Pressure During Blowdown

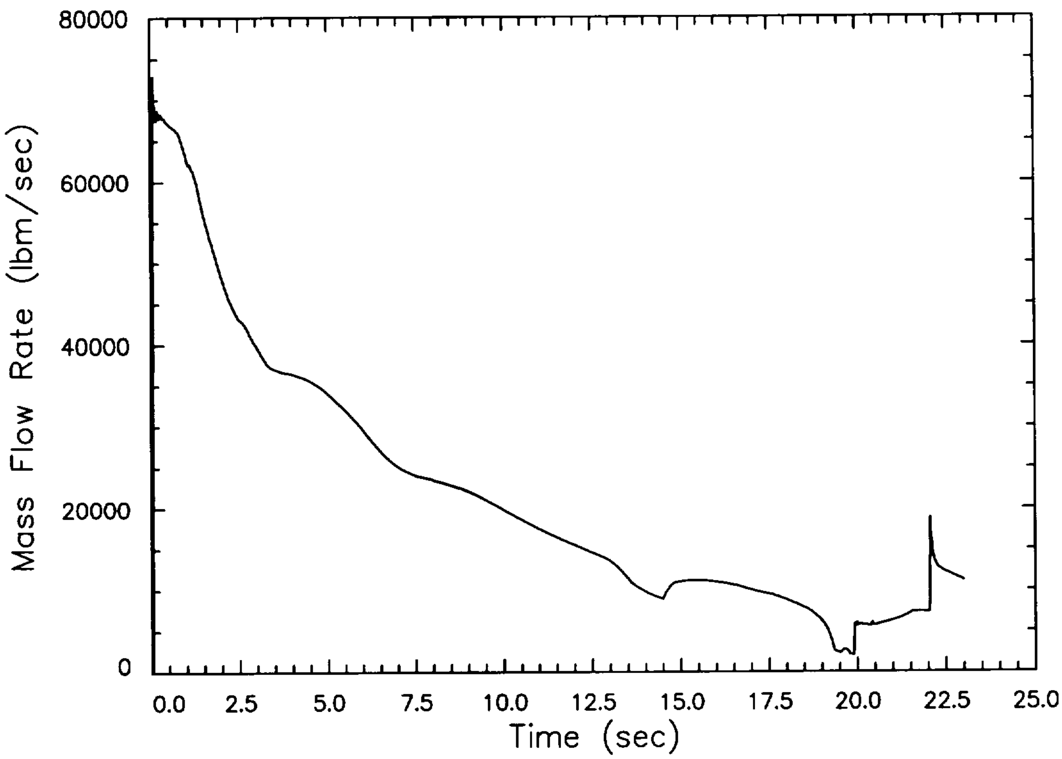

Figure 6.1.1-6 Total Break Flow Rate During Blowdown

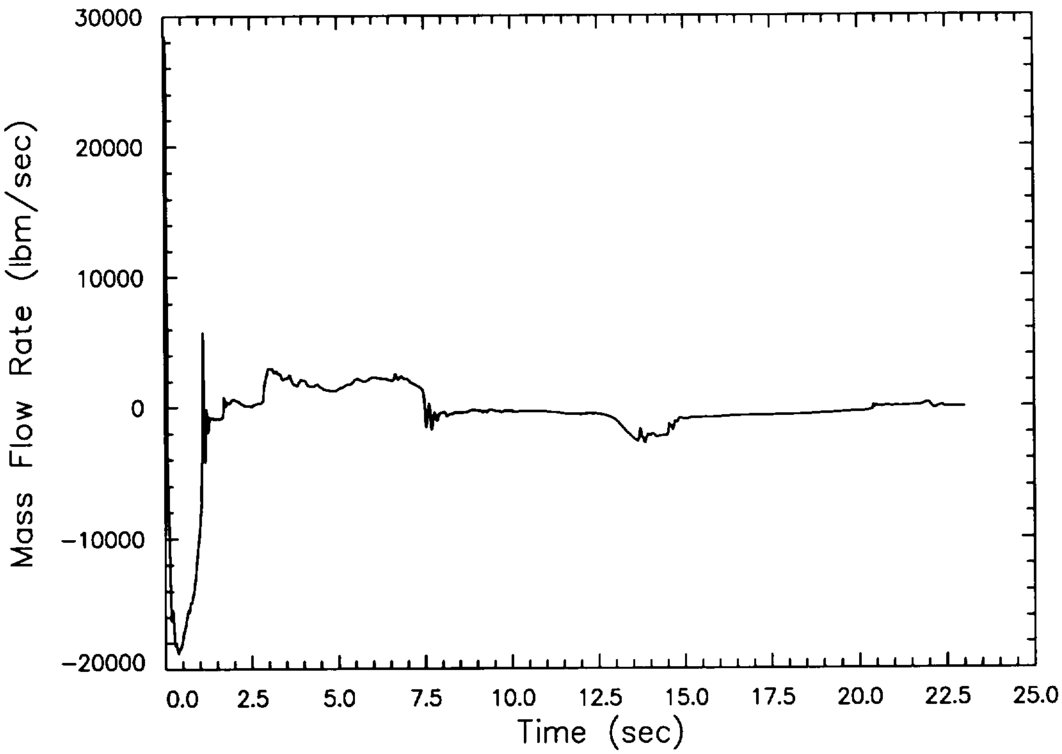

Figure 6.1.1-7 Average Core Inlet Flow Rate During Blowdown

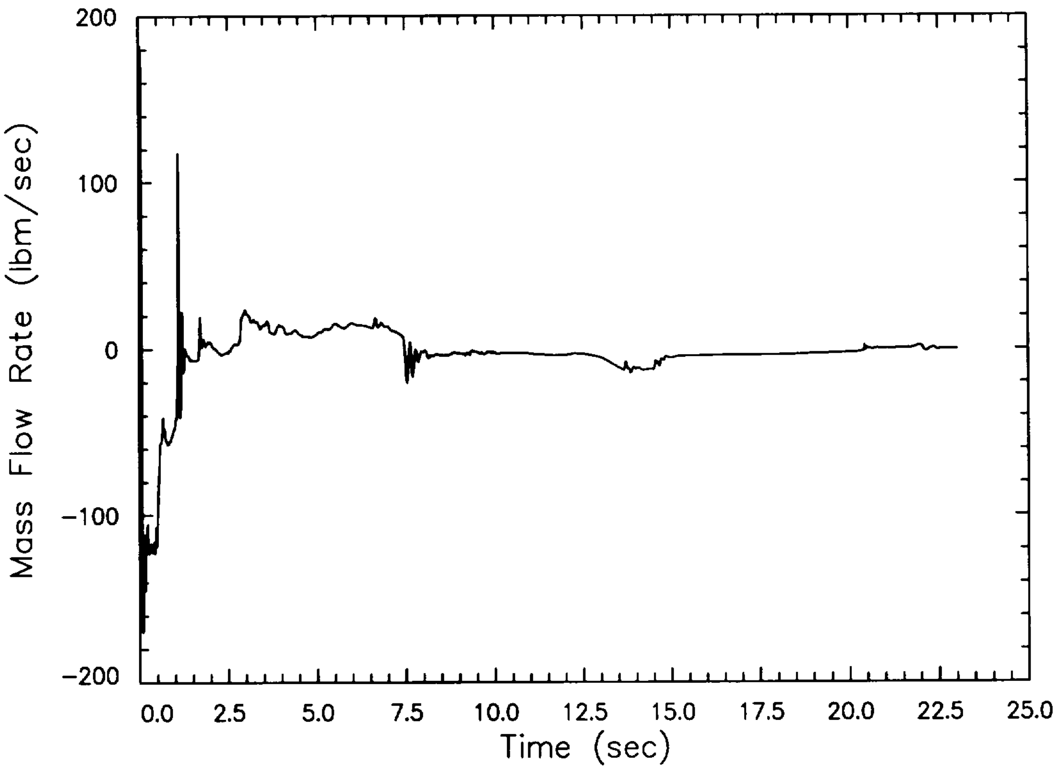

Figure 6.1.1-8 Hot Assembly Inlet Flow Rate During Blowdown

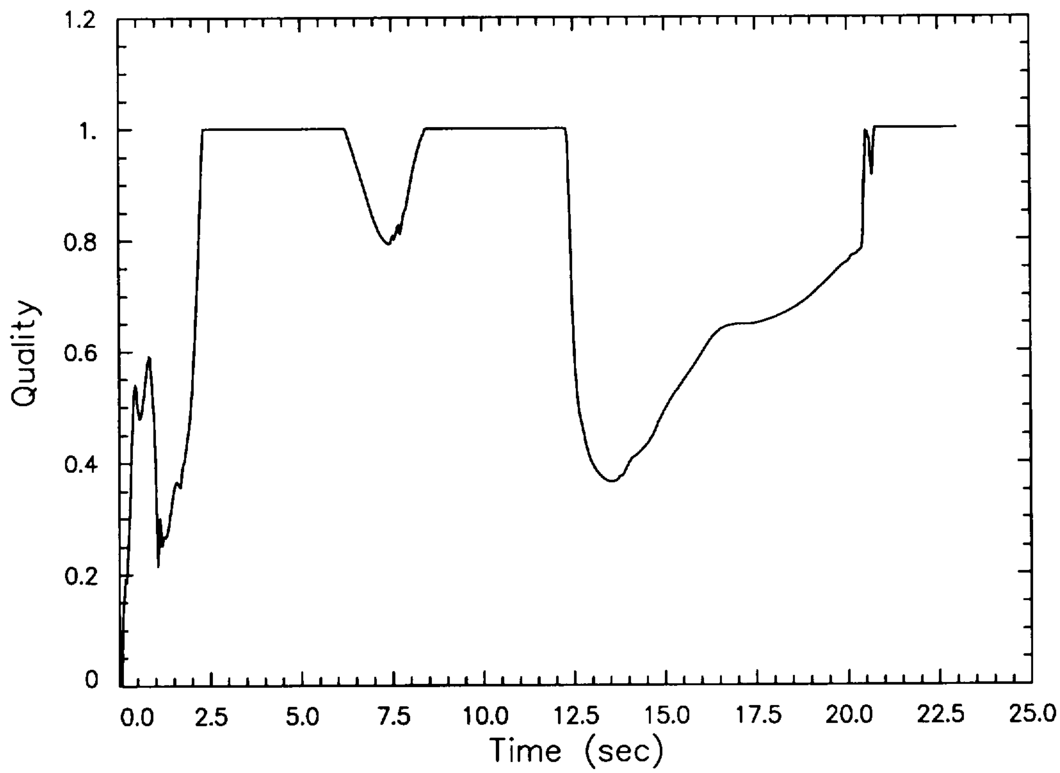

Figure 6.1.1-9 PCT Node Fluid Quality During Blowdown

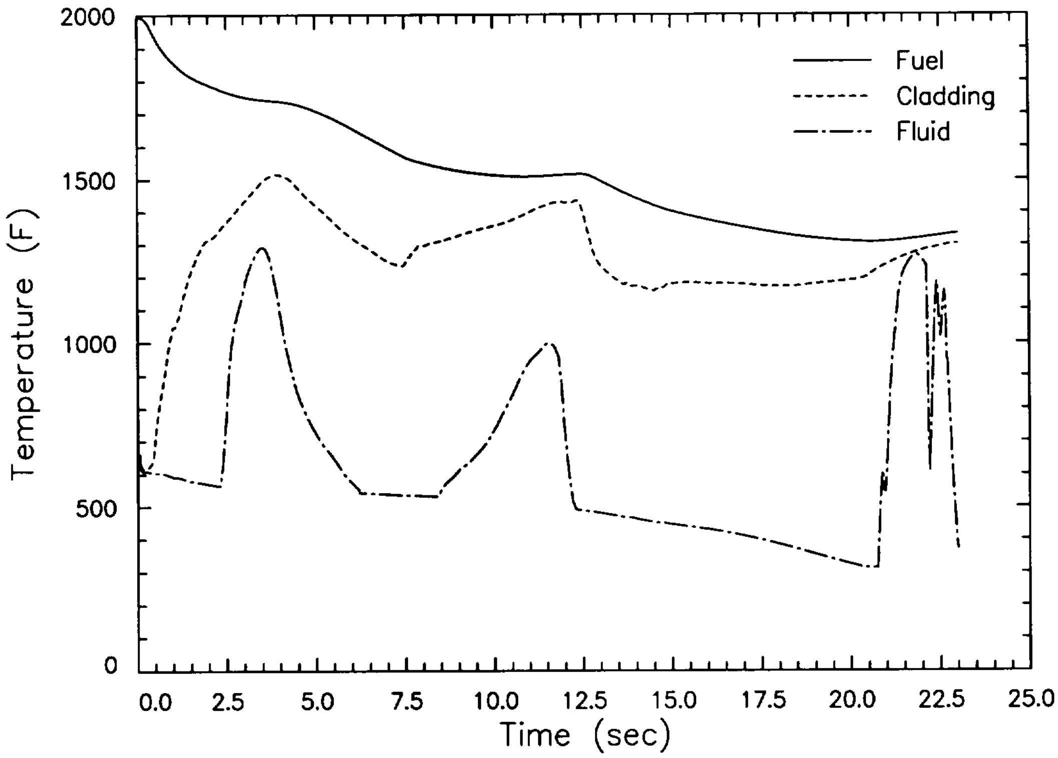

Figure 6.1.1-10 PCT Node Fuel (Average), Cladding and Fluid Temperatures During Blowdown

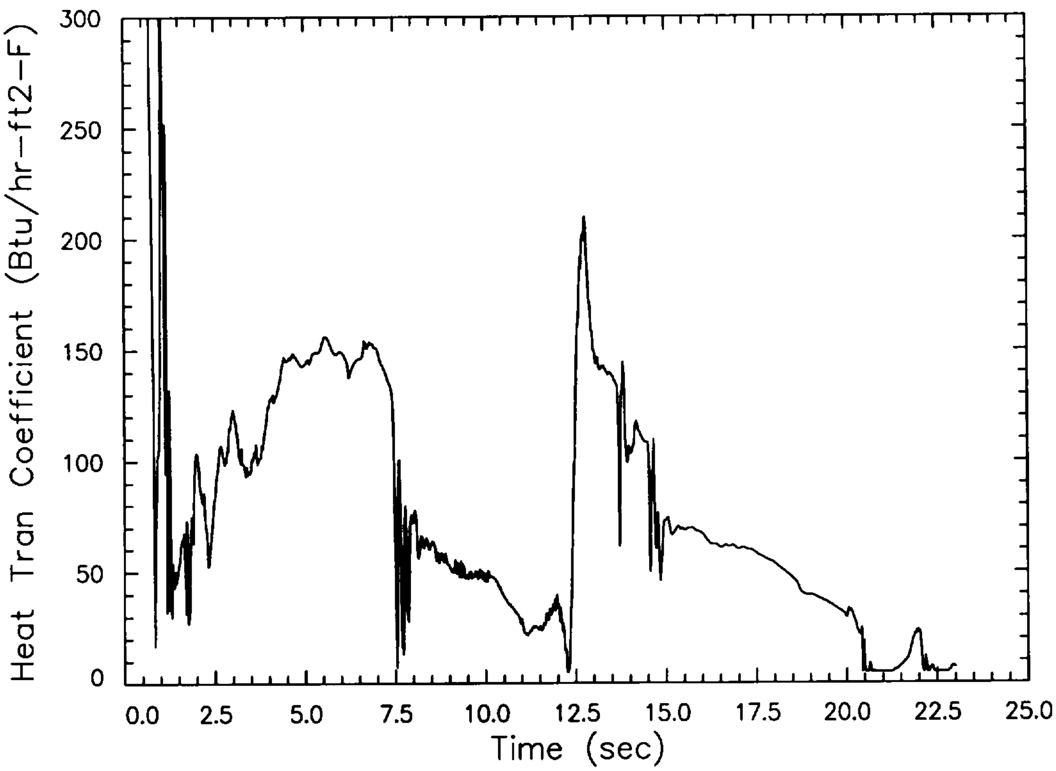

Figure 6.1.1-11 PCT Node Heat Transfer Coefficient During Blowdown

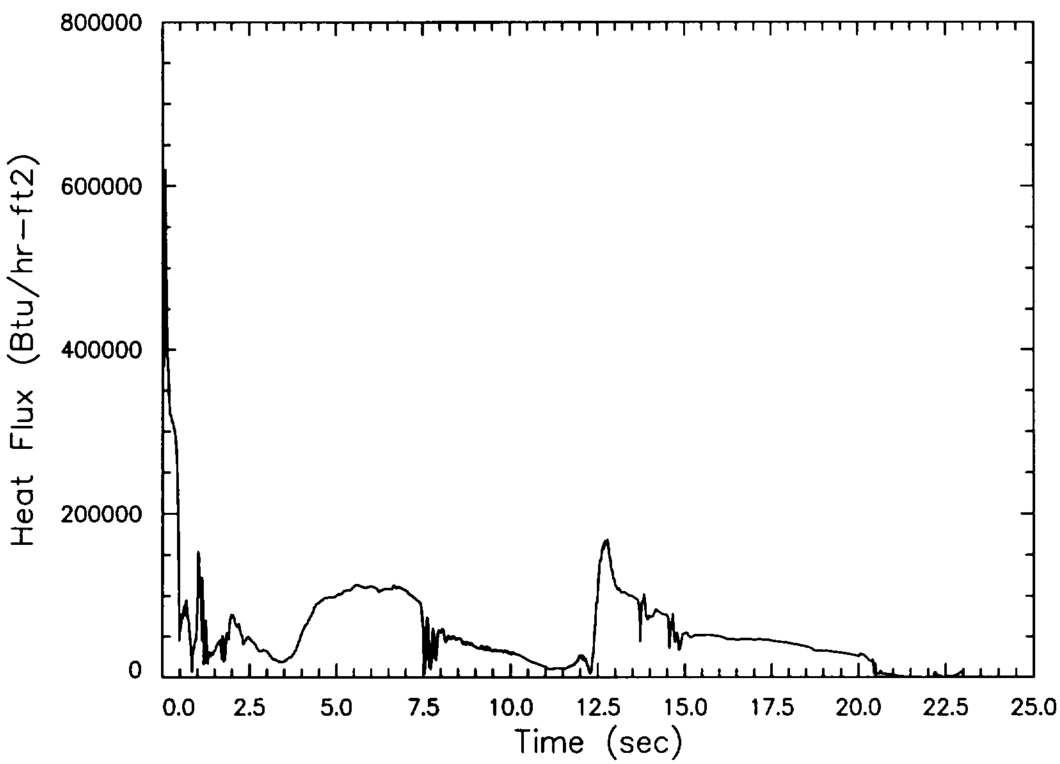

Figure 6.1.1-12 PCT Node Heat Flux During Blowdown

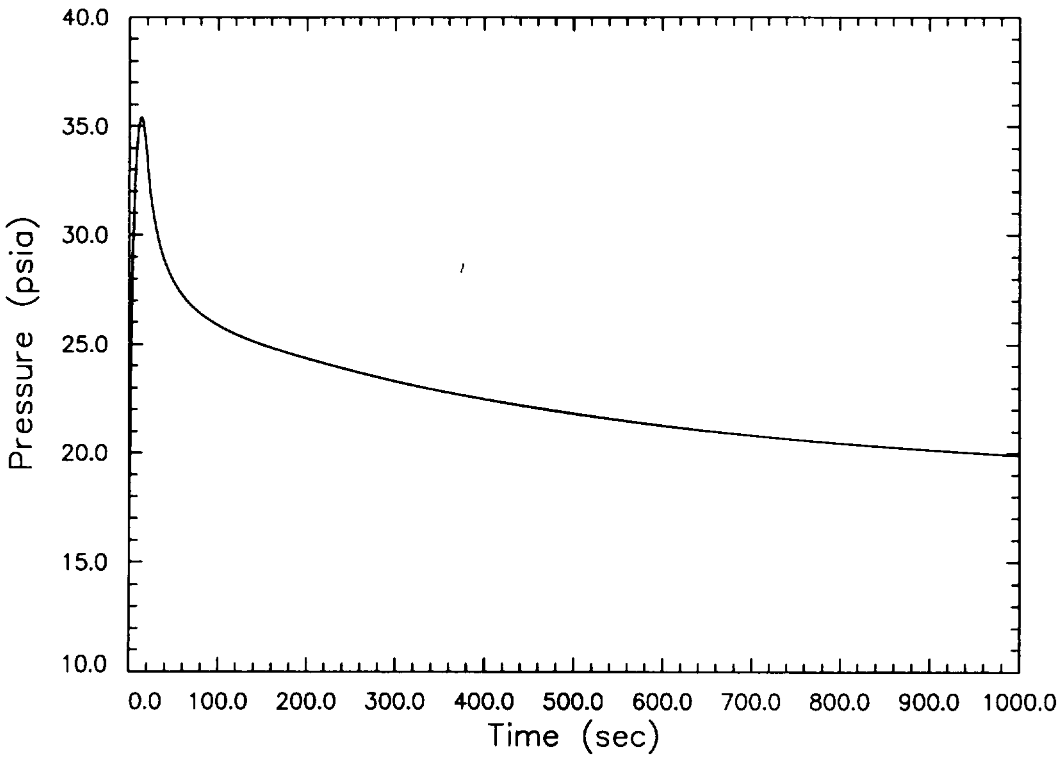

Figure 6.1.1-13 Containment Pressure

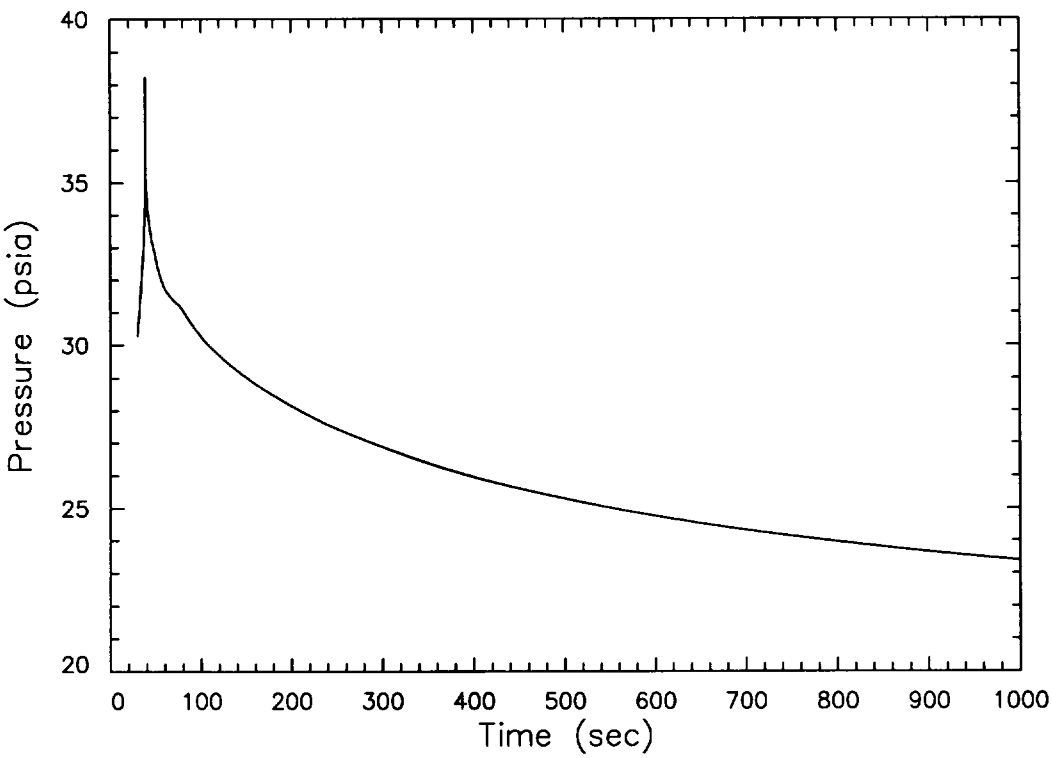

Figure 6.1.1-14 Upper Plenum Pressure After Blowdown

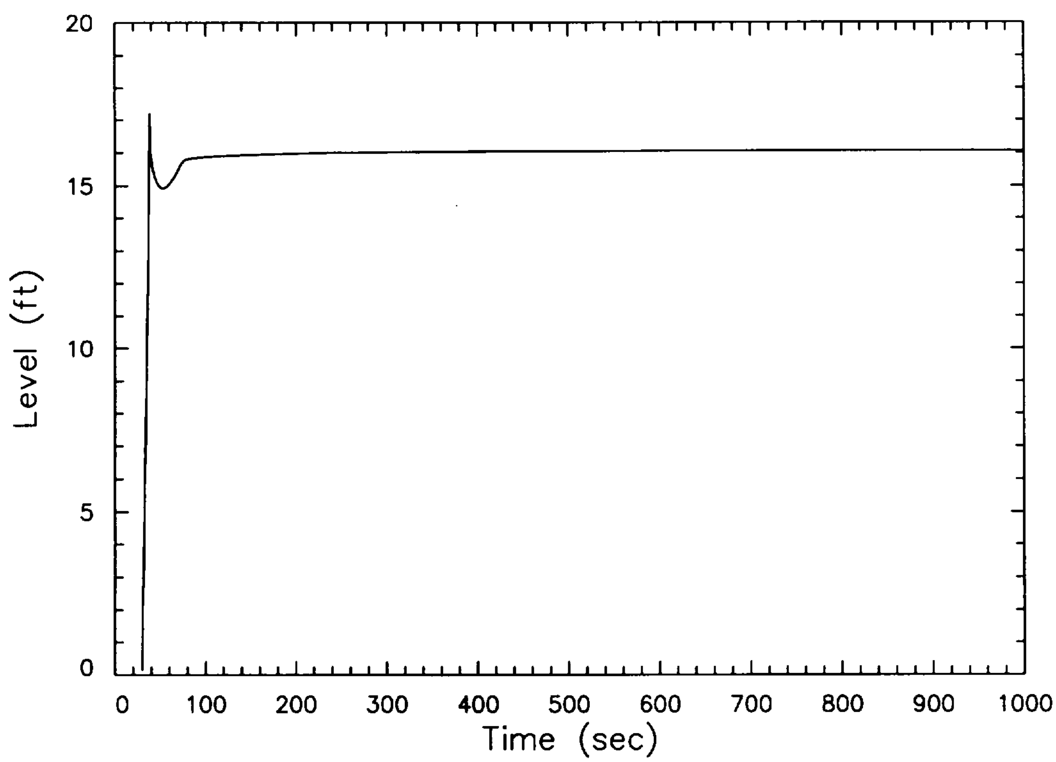

Figure 6.1.1-15 Downcomer Collapsed Liquid Level

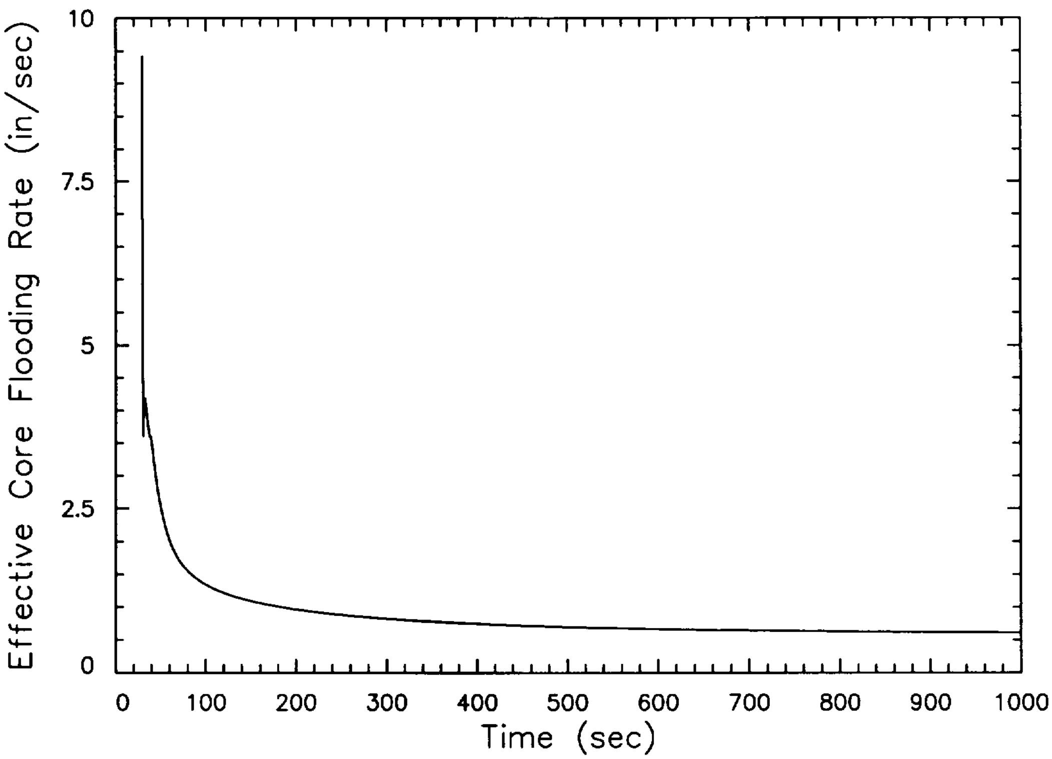

Figure 6.1.1-16 Core Effective Flooding Rate

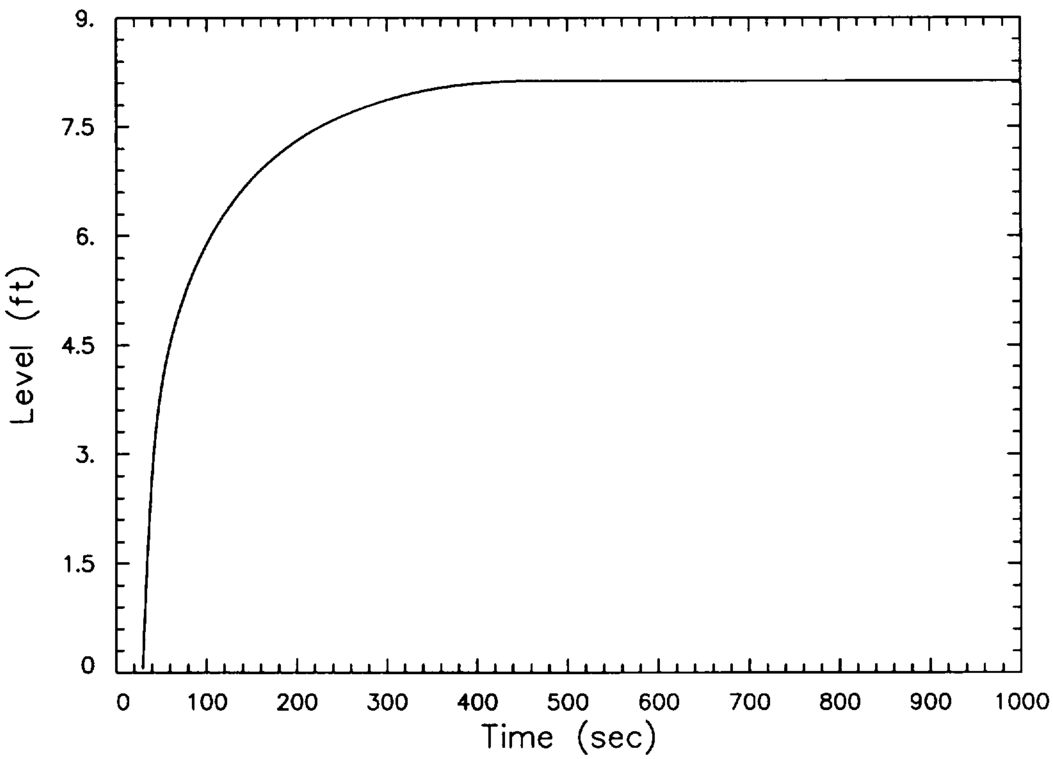

Figure 6.1.1-17 Core Collapsed Liquid Level

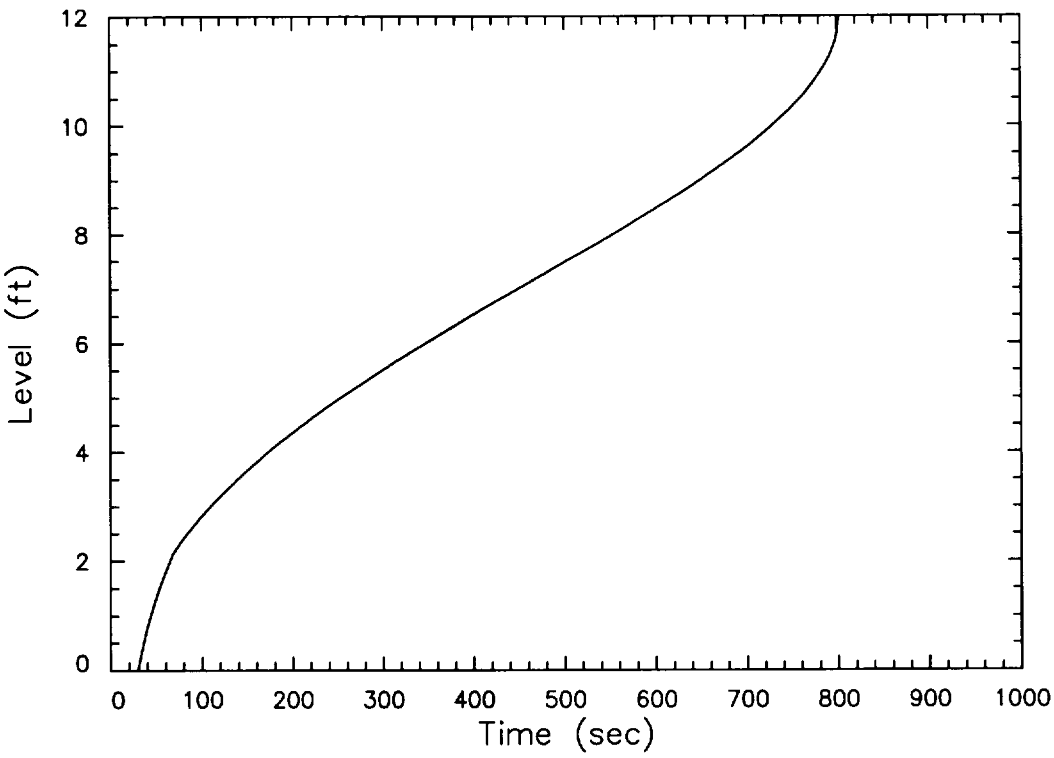

Figure 6.1.1-18 Core Quench Level

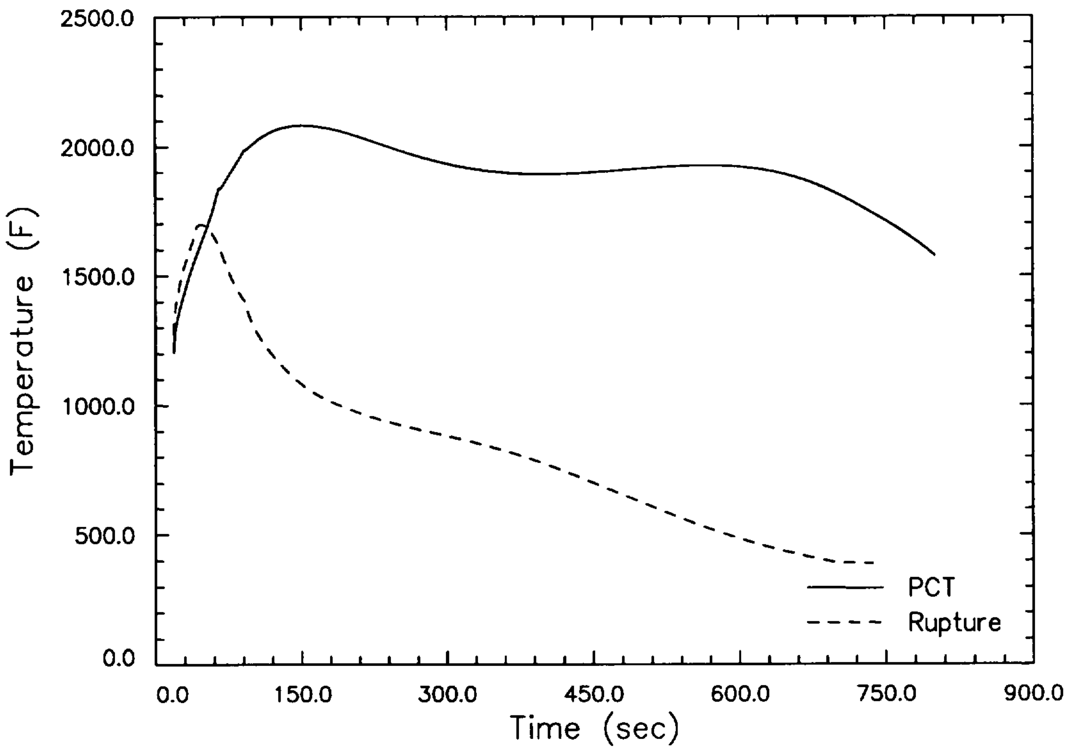

Figure 6.1.1-19 PCT Node and Ruptured Node Cladding Temperatures

6.2.26 Inadvertent Operation of the Emergency Core Cooling System During Power Operation (FSAR Event 15.5.1)

6.2.26.1 Identification of Causes and Accident Description

Inadvertent Operation of the Emergency Core Cooling System (IOECCS) could result from operator error or a false electrical actuation signal. Following the actuation signal, the suction of the charging pumps is re-aligned to the Refueling Water Storage Tank (RWST) from the Volume Control Tank (VCT). The valves isolating the Boron Injection Tank (BIT) from the charging pumps and the valves isolating the BIT from the injection header automatically open. The charging pumps then force concentrated boric acid from the RWST into the Reactor Coolant System (RCS). If a reactor trip does not occur coincident with safety injection actuation, the turbine throttle valves will open to offset the addition of negative reactivity from the Safety Injection System (SIS). The transient is eventually terminated by the reactor protection system due to low pressurizer pressure or manual trip. The time to trip is affected by the initial operating conditions including core burnup history that affects boron concentration, rate of change of boron concentration, and doppler and moderator coefficients.

The operator will determine if Safety Injection should be terminated. For spurious occurrence, the operator would stop the safety injection after ensuring satisfactory plant conditions per operating procedures and maintain the plant in hot standby conditions.

6.2.26.2 Description of Analysis

For the HNP Power Uprate project, three ANF-RELAP transient analyses were performed. Two of the analyses were performed to examine minimum departure from nucleate boiling ratio (MDNBR) with beginning of cycle (BOC) and end of cycle (EOC) kinetics. The third case examines pressurizer overfill and the thermal-hydraulic conditions at the pressurizer Power Operated Relief Valve (PORV) and Safety Relief Valve (SRV) inlets.

<u>BOC MDNBR Case:</u> The main emphasis for this case is to minimize RCS pressure during the transient, thereby posing a challenge to MDNBR. This event was initiated from hot full power and an average temperature of 588.8 °F to minimize the initial DNBR margin.

<u>EOC MDNBR Case:</u> This case also examines MDNBR consequences but the biasing is established with the intent of avoiding a premature reactor trip on low pressurizer pressure. With minimum boron concentration and automatic rod control enabled there is a possibility of a power increase due either to automatic rod control (ARC) initiated rod movement (Tavg error) and/or positive moderator reactivity feedback. This event was initiated from hot full power and an average temperature of 588.8 °F to minimize the initial DNBR margin.

Pressurizer Overfill Case: The intent of this case is to examine the fluid thermal-hydraulic conditions at the inlet to the pressurizer PORVs and SRVs. The event was biased to conservatively ensure that the fluid temperatures seen by the pressurizer valves will be as low as realistically achievable.

The transient response of the reactor system is calculated using the ANF-RELAP computer program (Reference 5). The Reference 5 methodology contains the following safety evaluation report (SER) restriction that relates to this event:

> *The [ANF-RELAP] methodology cannot be used in situations where... the boron tracking model is needed without further justification. If... the steam line break boron tracking model is used with the [ANF-RELAP] methodology, then the applicability of the steam line break methods to the non-LOCA event under considerations should be justified.*

The boron reactivity feedback model from the NRC approved MSLB methodology was conservatively implemented in this analysis. This model is justified as conservative for use in this application because the negative reactivity associated with the boron injection is delayed until the borated water is near the top of the core. Any positive feedback effects due to colder water entering the core is seen before the boron effects. This delay increases potential for higher core power levels that could result in lower MDNBRs. Addition of boron in the pressurizer overfill cases does not impact the results.

6.2.26.3 Acceptance Criteria

The HNP FSAR Section 15.5.1.4 states that the IOECCS event has three acceptance criteria as follows:

1. The pressure in the reactor coolant and main steam systems should be maintained below 110% of design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

6.2.26.4 Results

An analysis was performed to support plant operation following steam generator replacement and power uprate (SGR/PUR). The analysis bounds plant operation for up to 2900 MWt nominal core power (2958 MWt with 2% uncertainty), and up to 3% steam generator tube plugging, and with RWST boron concentrations between the Technical Specification limits of 2400 ppm and 2600 ppm. The analysis was performed at RCS T_{avg} of 588.8°F and bounds operation at a reduced RCS T_{avg} of 580.8°F.

The results of the analysis are discussed with respect to each of the acceptance criteria as follows:

Acceptance Criteria 1

The pressure in the reactor coolant and main steam systems should be maintained below 110% of design value. As seen in Figures 6.2.26-4, 6.2.26-11, and 6.2.26-17, the pressurizer/RCS systems are maintained below, at, or near the PORV and/or Safety valve setpoints, which adequately protect the design pressure of the RCS system.

Acceptance Criteria 2

Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs. The analyses of the BOC and EOC cases show that this criterion is met as described below.

BOC MDNBR Case: The sequence of events and results are given in Table 6.2.26-3. The responses of key system variables are given in Figures 6.2.26-1 to 6.2.26-7. The ANF-RELAP calculated DNBR trend clearly shows that this event is not a challenge to minimum DNBR since the DNBR generally increases throughout the event. MDNBR is predicted to occur shortly after event initiation, but is insignificantly different from the initial value. Thus, this event poses no challenge to the DNB specified acceptable fuel design limit (SAFDL).

EOC MDNBR Case: The sequence of events and results are given in Table 6.2.26-4. The responses of key system variables are given in Figures 6.2.26-8 to 6.2.26-14. The ANF-RELAP calculated DNBR trend clearly shows that this event is not a challenge to minimum DNBR since the DNBR generally increases throughout the event. MDNBR is predicted to occur after event initiation, but is insignificantly different from the initial value. Thus, this event poses no challenge to the DNB SAFDL.

Acceptance Criteria 3

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. If the Reactor Protection System (RPS) initiates a reactor trip on a SIS signal, the plant would be brought to hot standby or cold shutdown condition after ensuring satisfactory plant conditions per operating procedures and Technical Specifications. For this condition, the potential exists for the pressurizer to overfill from continued ECCS injection. The liquid flow capacity of the pressurizer PORVs and/or SRVs greatly exceeds the capacity of the high-head safety injection system; thus, RCS overpressurization is not a concern.

The pressurizer PORVs and SRVs were previously evaluated to determine if the valves remain operable during the discharge of subcooled water in accordance with NUREG 0737 II.D.1 as outlined in FSAR – TMI Appendix. The PORVs were shown to remain operable at inlet pressures of 2532 to 2545 psia and temperatures of 446 to 670 °F. The SRVs were shown to remain operable at an inlet pressure of 2475 psia and an inlet temperature of 635 °F .

An analysis of the IOECCS event was performed to evaluate the thermal hydraulic conditions at the inlet to the pressurizer SRVs and PORVs subsequent to SGR/PUR. The event analysis was biased to conservatively ensure that the fluid temperatures seen by the pressurizer PORVs and SRVs are as low as realistically achievable. The assumed state of plant systems and input biases for this case is given in the pressurizer overfill columns of Tables 6.2.26-1 and 6.2.26-2.

Two of the three pressurizer PORVs are safety-related. The associated pneumatic power and controls are designed to function by remote manual operation. The controls associated with manual operation of the valve are safety-related and the accumulator and piping leading from the accumulator to the valve operators are safety-related. The remaining portions of the pneumatic supply and the automatic actuation/control system are not safety-related but are very reliable for the following reasons:

1. Reference 2 concludes that the PORV circuitry meets the requirements of NUREG-0737, Item II.D.1 stating that "the PORVs were qualified under the pump and valve operability program (PVORT), the actuation transmitters are environmentally qualified, the cable is qualified (although not run as 1E), and the PIC cabinets are essentially the same hardware as the class 1E cabinets.

2. The PORVs have two diverse pneumatic supplies. One supply is the nitrogen system and the other is the instrument air system. The instrument air system 1A and 1B air compressors can be manually loaded onto the "A" and "B" train emergency diesel generators, respectively, in the event of a loss of offsite power.

The PORVs were assumed to initially open to increase the rate of pressurizer fill. The PORVs were then conservatively modeled as having exhausted their motive air supply. At this point, the PORVs are modeled in the closed position, in order to allow pressure increases that would potentially challenge the operation of the SRVs. The sequence of events and results of the analysis are given in Table 6.2.26-5. The responses of key system variables are given in Figures 16.2.26-15 through 16.2.26-21. Figure 16.2.26-15 shows that the pressurizer is filled solid at approximately 600 seconds (10 minutes). The liquid inlet pressures and temperatures for the pressurizer PORVs and SRVs remain above 2250 psia and 635 °F for approximately 950 seconds or 15.8 minutes (see Figures 16.2.26-20 and 16.2.26-21). The liquid temperature from 950 seconds to event termination at 1200 seconds remains above 564 °F.

6.2.26.5 Conclusions

Consistent with the current licensing basis, the results demonstrate that the acceptance criteria for this event are met for SGR/Uprating conditions. The analysis for this event supports operation with the Model Delta 75 replacement steam generators at core power of 2900 MWt with nominal primary T_{avg} at full power from 580.8°F to 588.8°F for steam generator tube plugging from 0% to 3%. The analysis bounds plant operation with RWST boron concentrations between the Technical Specification limits of 2400 ppm and 2600 ppm. Conclusions with respect to each of the identified acceptance criteria are provided as follows:

<u>Acceptance Criteria 1</u>

The pressure in the reactor coolant and main steam systems should be maintained below 110% of design value. This acceptance criterion is not challenged since the SRV and PORV relief capacity far exceeds the ECCS capacity to fill the pressurizer.

<u>Acceptance Criteria 2</u>

Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 limit for PWRs. This acceptance criterion is not challenged since the DNBR margin increases throughout the event, except for an insignificant decrease early in the event.

<u>Acceptance Criteria 3</u>

The results of the analysis performed for SGR/PUR indicate that conditions at the inlet to the pressurizer SRVs are well within the range of conditions previously evaluated as acceptable for compliance with NUREG-0737 II.D.1 for almost 16 minutes after event initiation. Conditions remain well within those previously evaluated for the PORVs up to event termination at 20 minutes after event initiation. The PORVs are expected to mitigate the event, thereby eliminating the challenge to the SRVs.

6.2.26.6 References

1. SPC Letter VNG:00:292, Revision 1, "Transmittal of the Letter Report and Calculation Notebook for the Evaluation of IOECCS Event for Harris at Uprated Conditions" dated November 15, 2000.
2. Letter from NRC's Richard A. Becker to CP&L's Lynn Eury dated May 31, 1989 "Evaluation of Carolina Power and Light Company's Shearon Harris Unit 1, Plant Specific Submittals in Response to NUREG-0737, TMI Action Plan Requirement, Item II.D.1 (TAC No. 63565)."
3. EPRI NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program" dated December 1982.
4. ANF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, May 1992

Table 6.2.26-1 Assumed State of Plant Systems

Parameter	BOC MDNBR	EOC MDNBR	PZR Overfill
Automatic Rod Control	Manual	Auto	Manual
Pressurizer Heaters	Disabled	Disabled	Disabled
Pressurizer Spray	Available	Available	Available
Pressurizer PORVs	Available	Available	Available at first, then assumed to lose motive power
Steam Bypass Valves	Not Modeled	Not Modeled	Not Modeled
Steam Atmospheric Dump Valves	Not Modeled	Not Modeled	Not Modeled
Main Feedwater	Auto (no consequence)	Auto (no consequence)	Auto (no consequence)
Auxiliary Feedwater	Available	Available	Available
Charging/SI Pumps	2 Pumps (Maximum Flow)	2 Pumps (Minimum Flow)	2 Pumps (Maximum Flow)
SI Boron Concentration	Maximum per Tech. Specs. (2600 ppm)	Minimum per Tech. Specs. (2400 ppm)	Maximum per Tech. Specs. (2600 ppm)
SI Unborated Purge Volume (total)	23.13 ft^3	23.13 ft^3	23.13 ft^3
SI Fluid Temperature	40 °F	40 °F	40 °F
Letdown Flow	Not Modeled	Not Modeled	Not Modeled
Turbine Control	Auto / Load Demand	Manual	Auto / Load Demand
Cycle Exposure	BOC	EOC	EOC
Initial Core Power	Maximum rated plus 2% uncertainty (2958 MWt)	Maximum rated plus 2% uncertainty (2958 MWt)	Maximum rated plus 2% uncertainty (2958 MWt)
RCS Average Temperature	588.8 °F	588.8 °F	588.8 °F
Steam Generator Tube Plugging Level	3%	3%	3%
RCS Flow	Tech. Spec. minimum (293,540 gpm)	Tech. Spec. minimum (293,540 gpm)	Tech. Spec. minimum (293,540 gpm)

Table 6.2.26-2 Input Biasing

Parameter	BOC MDNBR Case	EOC MDNBR Case	PZR Overfill Case
Scram Reactivity Worth	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
High Pressurizer Level Trip Setpoint	Not Modeled	Nominal + Uncertainty	Nominal + Uncertainty
Low Pressurizer Trip Setpoint	Nominal - Uncertainty	Nominal - Uncertainty	Nominal - Uncertainty
Moderator Temperature Reactivity Coefficient	1.2 x BOC limit	EOC Bounding Minimum	EOC Bounding Minimum
Doppler Reactivity Coefficient	0.8 x BOC	EOC Bounding Minimum	EOC Bounding Minimum
Delayed Neutron Data	Nominal BOC	Nominal EOC	Nominal EOC
Prompt Neutron Lifetime (β/l)	Nominal BOC	Nominal EOC	Nominal EOC
Pressurizer Safety Valve Open Setpoint	Nominal - Tolerance	Nominal - Tolerance	Nominal + Tolerance
Pressurizer Safety Valve Stroke Time	Nominal (Initial lift includes delay for loop seal purge)	Nominal (Initial lift includes delay for loop seal purge)	Nominal (Initial lift includes delay for loop seal purge)
Non-Compensated Pressurizer PORV Open Setpoint	Nominal	Nominal	Nominal
MS Safety Valve Open Setpoints	Nominal + Tolerance	Nominal + Tolerance	Nominal + Tolerance
Initial Pressurizer Level	Nominal	Nominal	Nominal - Uncertainty
Initial Pressurizer Pressure	Nominal	Nominal	Nominal

Table 6.2.26-3 Sequence of Events (BOC MDNBR Case)

Event	Time (seconds)
Inadvertent actuation of HHSI system (maximum flow from 2 SI pumps)	0.0
Minimum DNBR (a negligible decrease from the initial value)	2.0
Non-borated water cleared from SI lines	25.0
Core power and pressurizer level begin to decrease	27.0
Main turbine valve fully open	50.5
Reactor trip signal on low pressurizer pressure	73.25
Main turbine trip	75.275
Minimum pressurizer pressure (1889.4 psia)	83.0
Minimum pressurizer level (23.4% of span)	83.5
Pressurizer spray actuates at 2260 psia	191.0
Maximum pressurizer level (~98% of span)	600.0
Transient terminated	600.0

Table 6.2.26-4 Sequence of Events (EOC MDNBR Case)

Event	Time (seconds)
Inadvertent actuation of HHSI system (minimum flow from 2 SI pumps)	0.0
Minimum DNBR (a insignificant decrease from the initial value)	52.5
Non-borated water cleared from SI lines	~71.0
ARC rods begin to step out of core	142.5
ARC banks borated fully out	599.5
Reactor power begins to decrease	600.0
RPS trip signal on high pressurizer level	812.65
Main turbine trip	813.20
Pressurizer outsurge begins	831.0
Pressurizer insurge resumes	856.0
Maximum pressurizer level (~98.8% of span)	1200.0
Transient terminated	1200.0

Table 6.2.26-5 Sequence of Events (Pressurizer Overfill Case)

Event	Time (seconds)
Inadvertent actuation of HHSI system: (maximum flow from 2 SI pumps)	0.0
RPS trip signal on SIS signal	2.025
Main turbine trip	2.05
Pressurizer spray actuates	3.2
Compensated pressurizer PORV opens	3.5
Pressurizer spray terminated	6.5
Compensated pressurizer PORV closes	7.5
Minimum pressurizer pressure (2200.1 psia)	8.5
Minimum pressurizer level (52.87% of span)	10.0
Non-borated water cleared from SI lines	~27.0
Compensated PORV opens	578.5
Pressurizer level = 100% of span	591.5
Pressurizer liquid level = 463.15 inches	900.0
Pressurizer PORVs assumed disabled	900.0
Pressurizer pressure = 2267 psia	900.0
Pressurizer filled with liquid (level = 463.38 inches)	918.5
Pressurizer SRV opens	945.0
Pressurizer SRV closes	947.0
Pressurizer pressure ranging between 2450 and 2550 psia	950-1200
Minimum SRV inlet temperature = ~564 °F (while valves are open)	950-1200
Transient terminated	1200.0

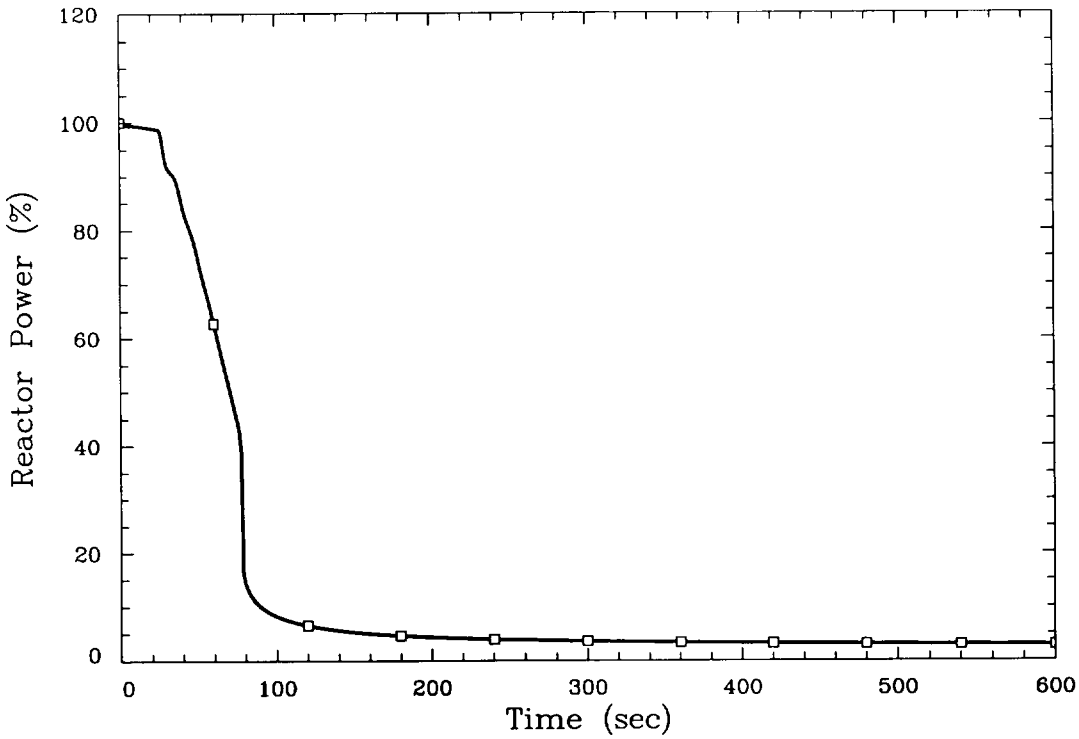

Figure 6.2.26-1 Reactor Power (BOC MDNBR Case)

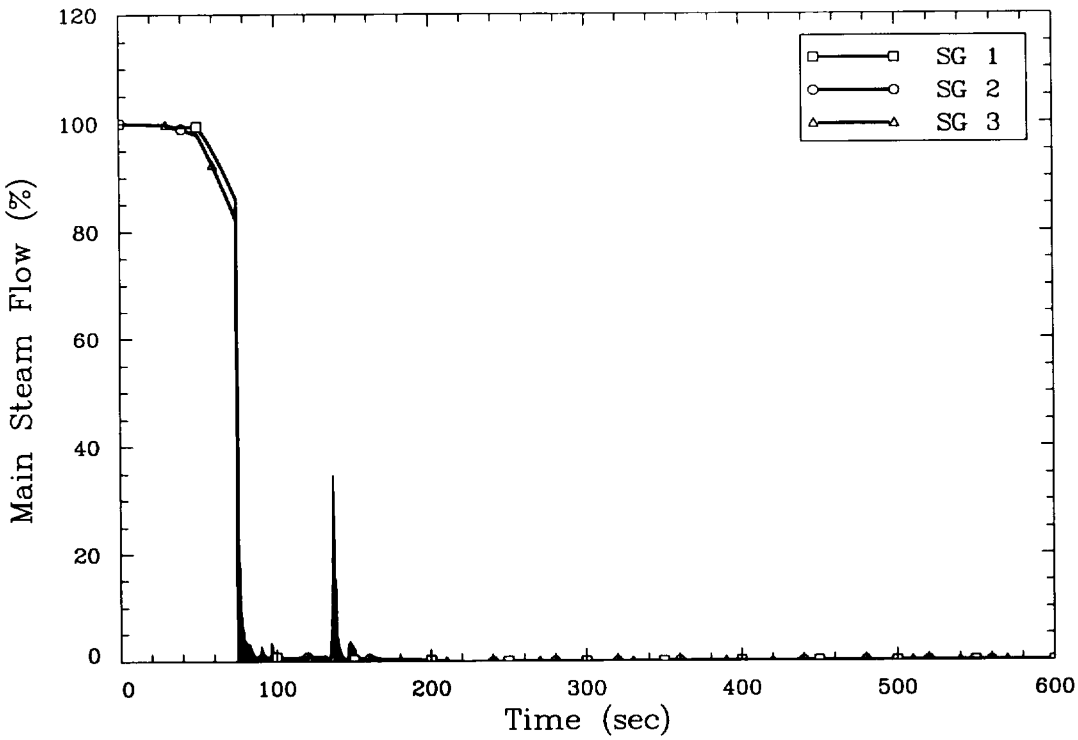

Figure 6.2.26-2 Main Steam Flow Rate (BOC MDNBR Case)

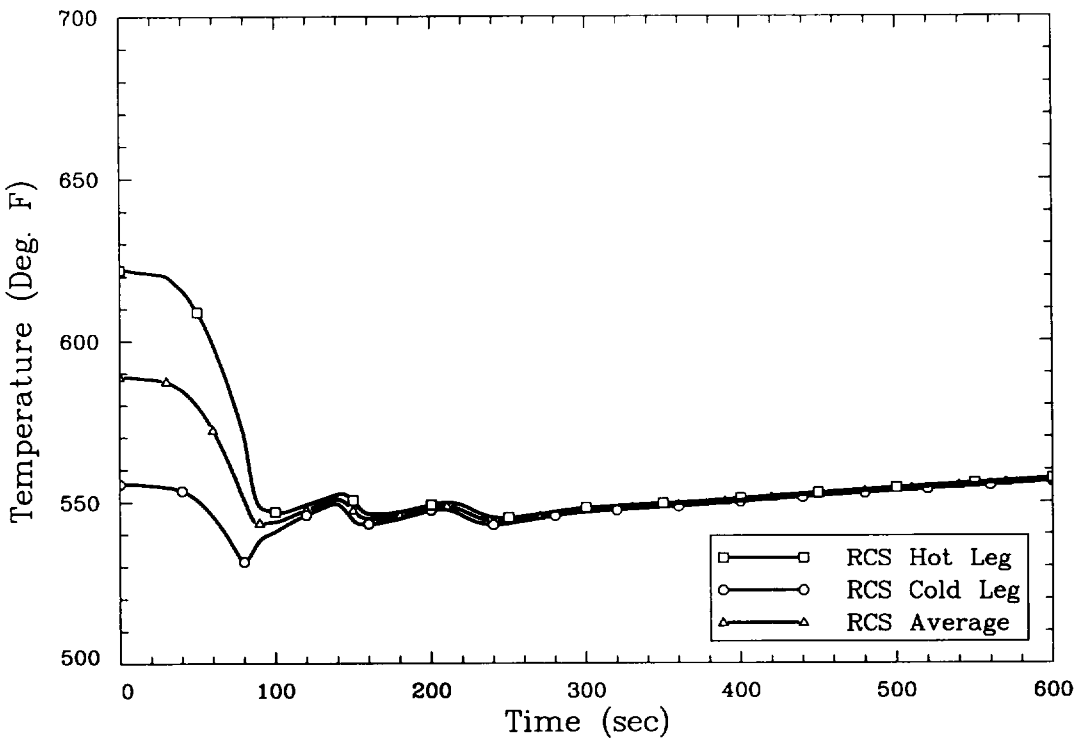

Figure 6.2.26-3 Reactor Coolant System Temperatures (BOC MDNBR Case)

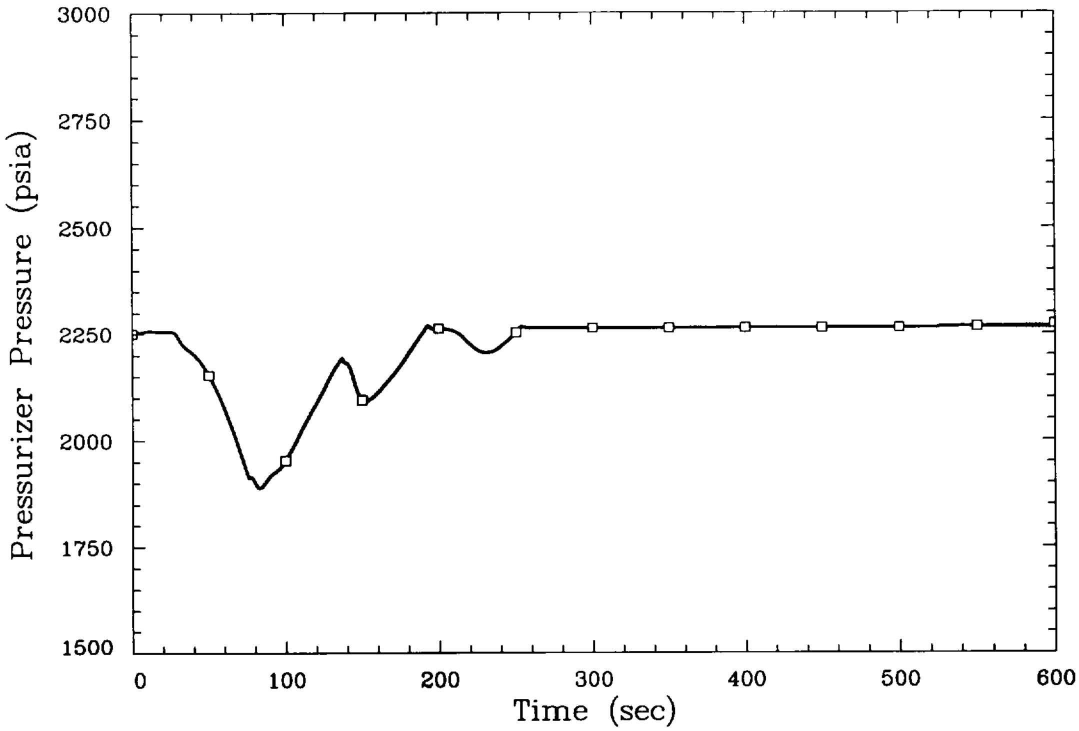

Figure 6.2.26-4 Pressurizer Pressure (BOC MDNBR Case)

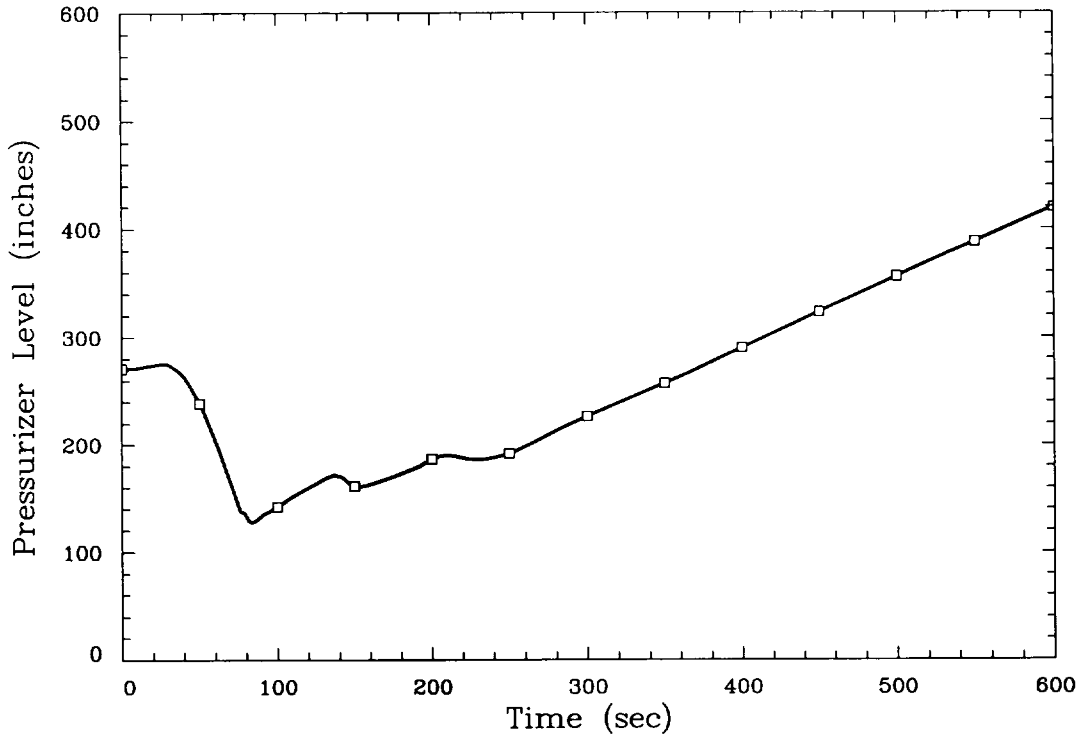

Figure 6.2.26-5 Pressurizer Level (BOC MDNBR Case)

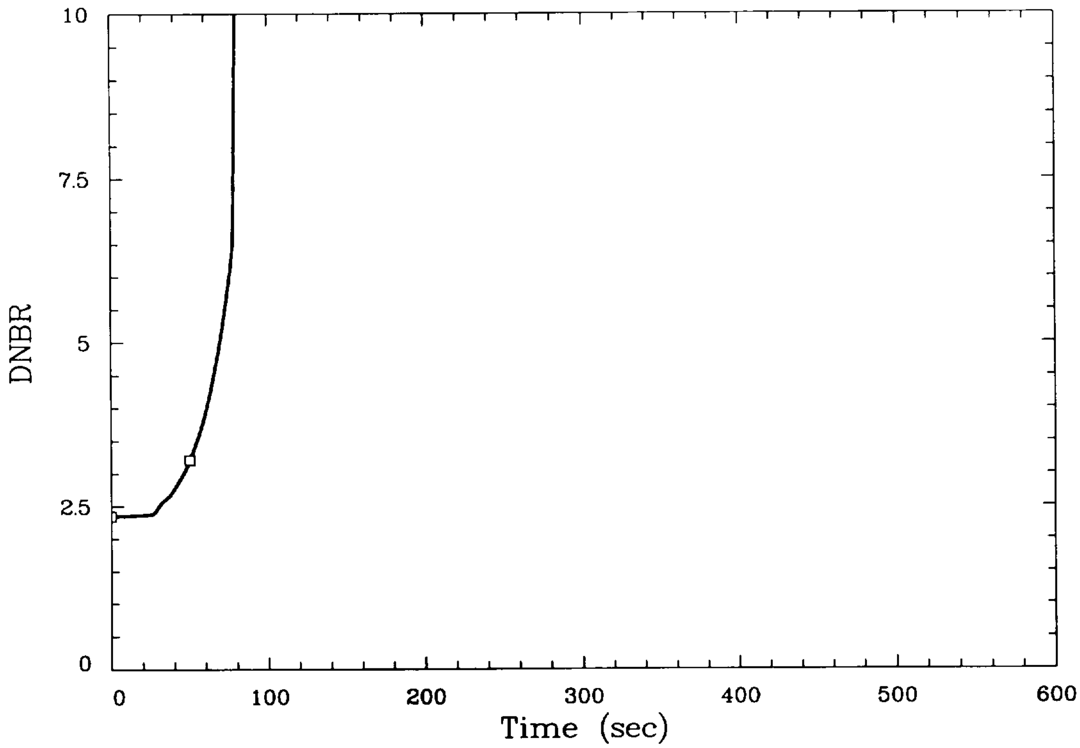

Figure 6.2.26-6 ANF-RELAP Predicted DNBR Trend (BOC MDNBR Case)

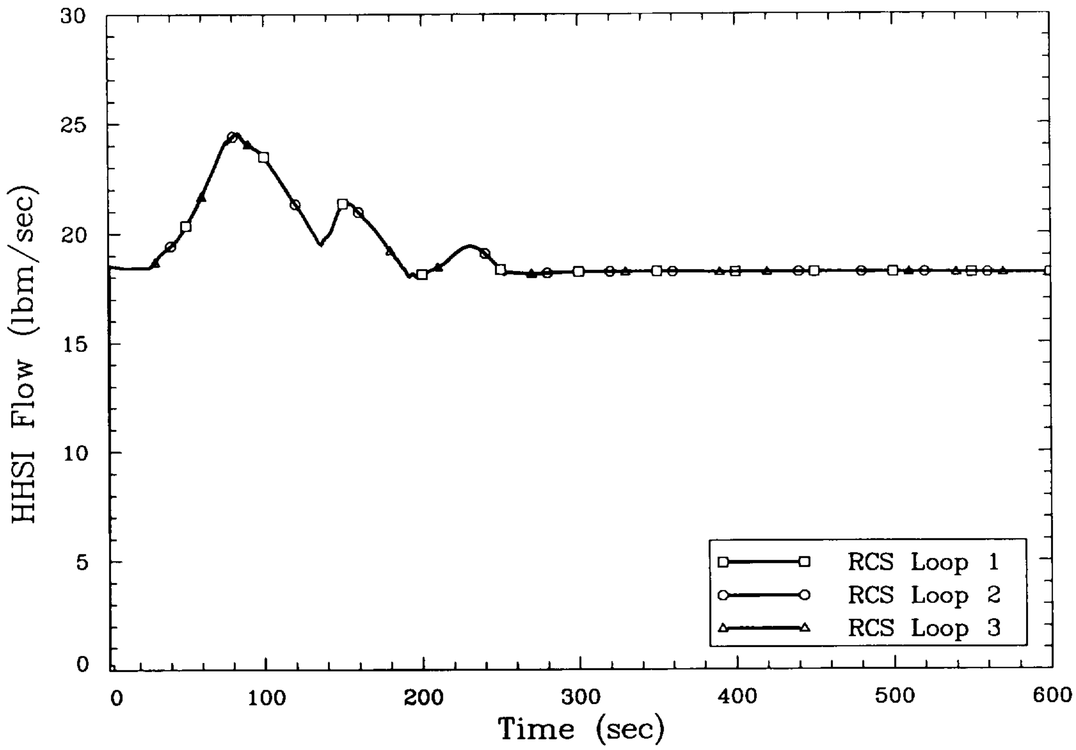

Figure 6.2.26-7 Safety Injection Flow Rate (BOC MDNBR Case)

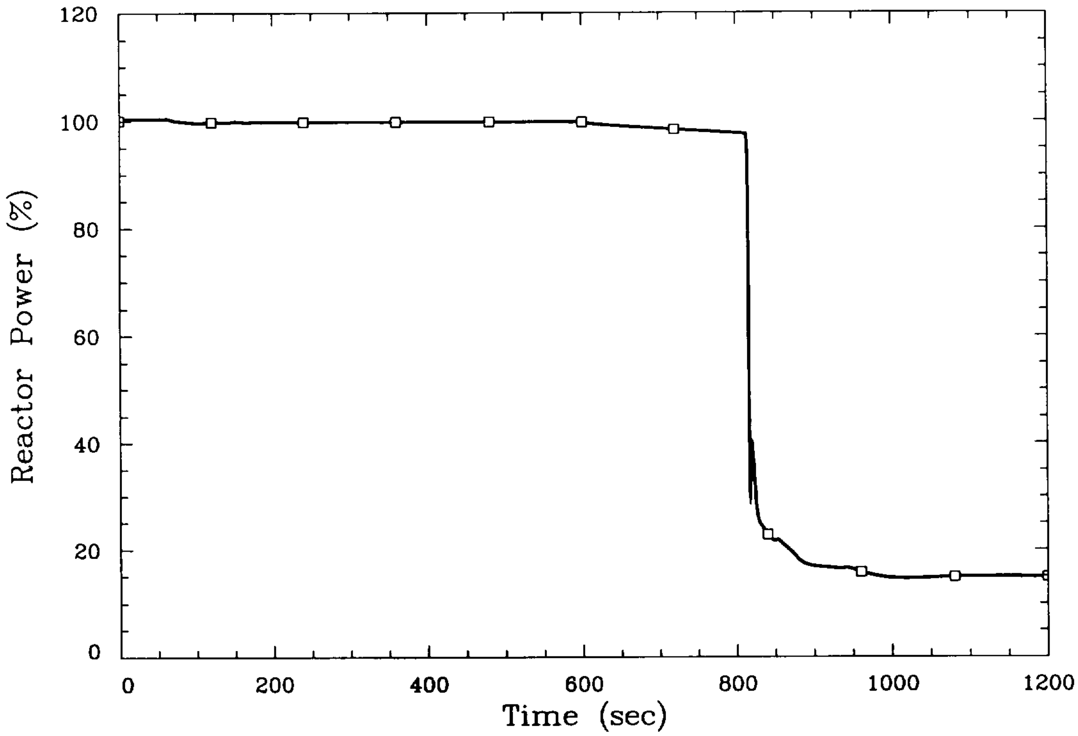

Figure 6.2.26-8 Reactor Power (EOC MDNBR Case)

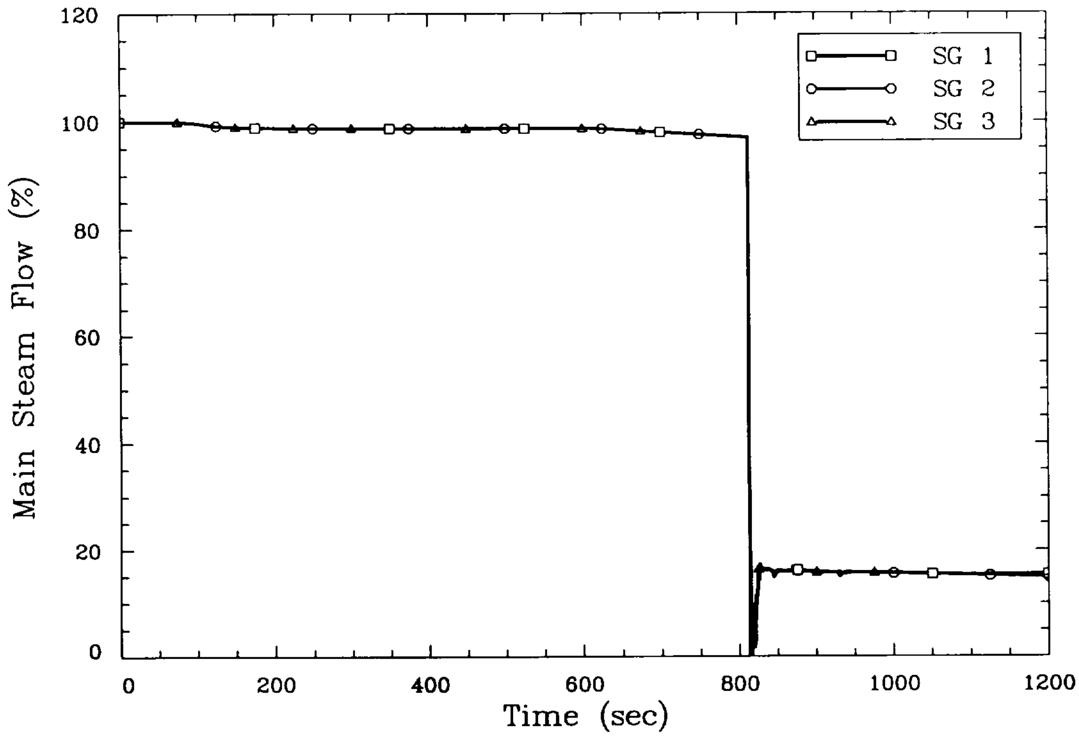

Figure 6.2.26-9 Main Steam Flow Rate (EOC MDNBR Case)

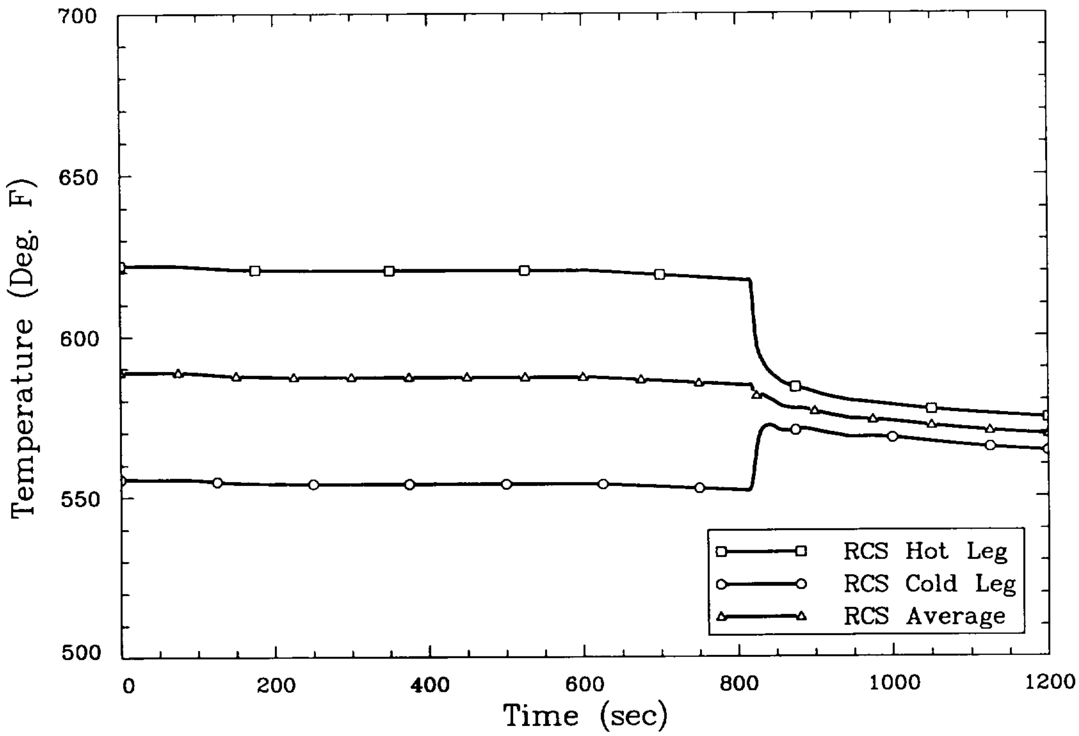

Figure 6.2.26-10 Reactor Coolant System Temperatures (EOC MDNBR Case)

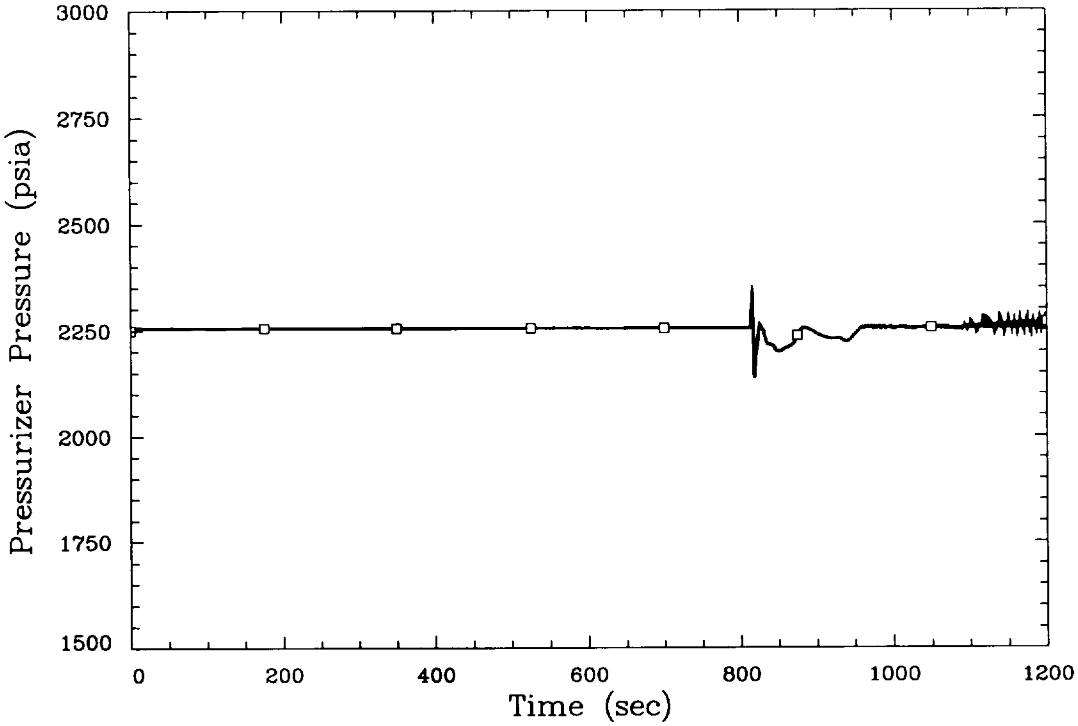

Figure 6.2.26-11 Pressurizer Pressure (EOC MDNBR Case)

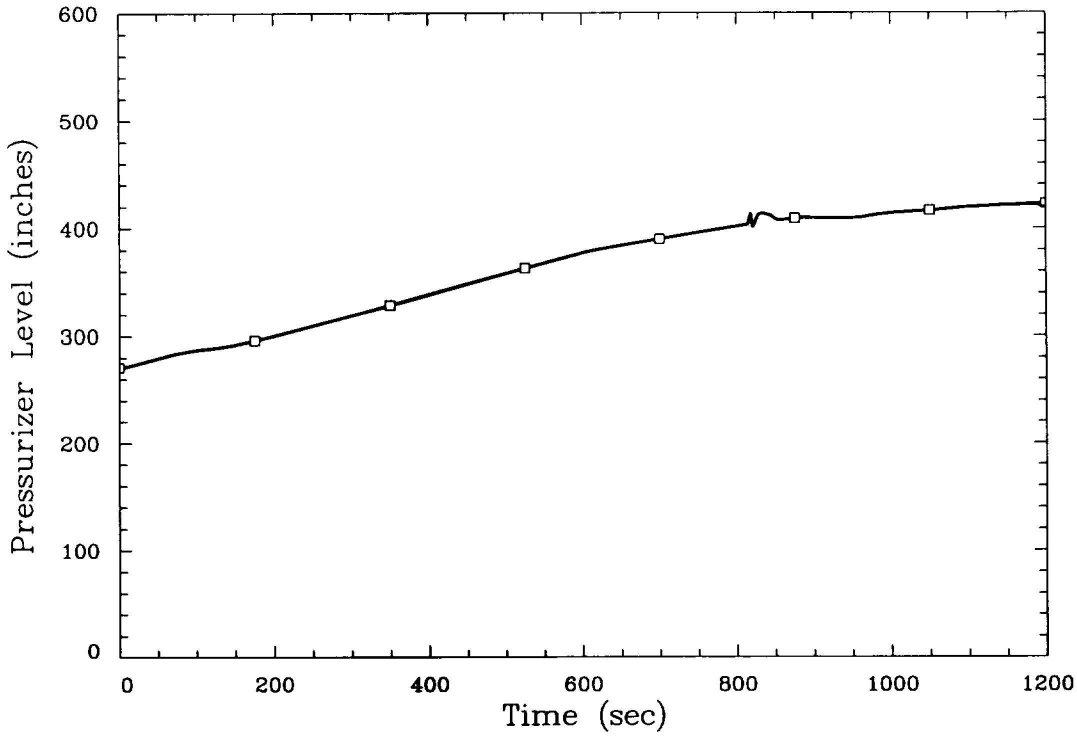

Figure 6.2.26-12 Pressurizer Liquid Level (EOC MDNBR Case)

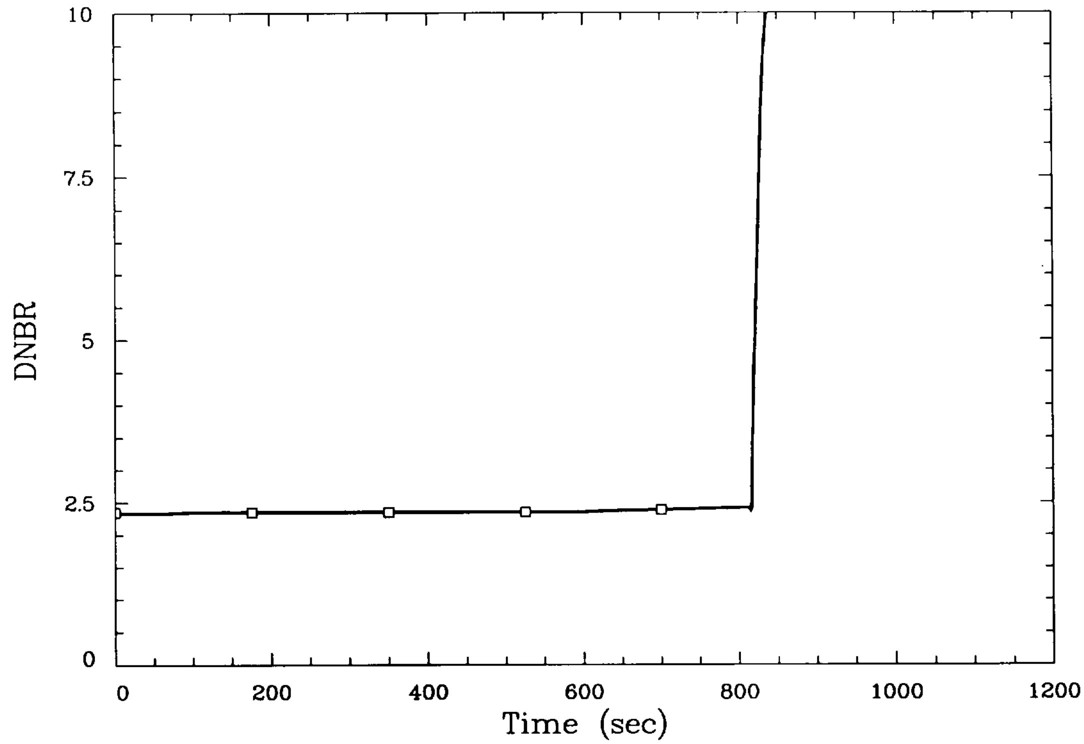

Figure 6.2.26-13 ANF-RELAP Predicted DNBR Trend (EOC MDNBR Case)

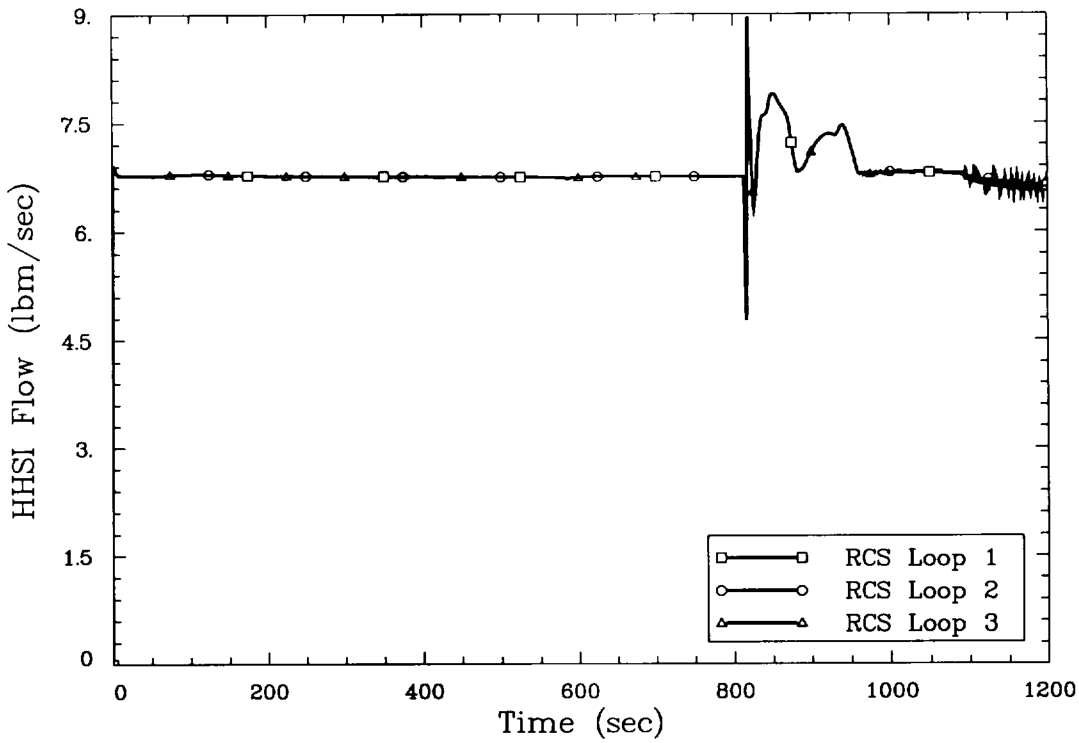

Figure 6.2.26-14 Safety Injection Flow Rate (EOC MDNBR Case)

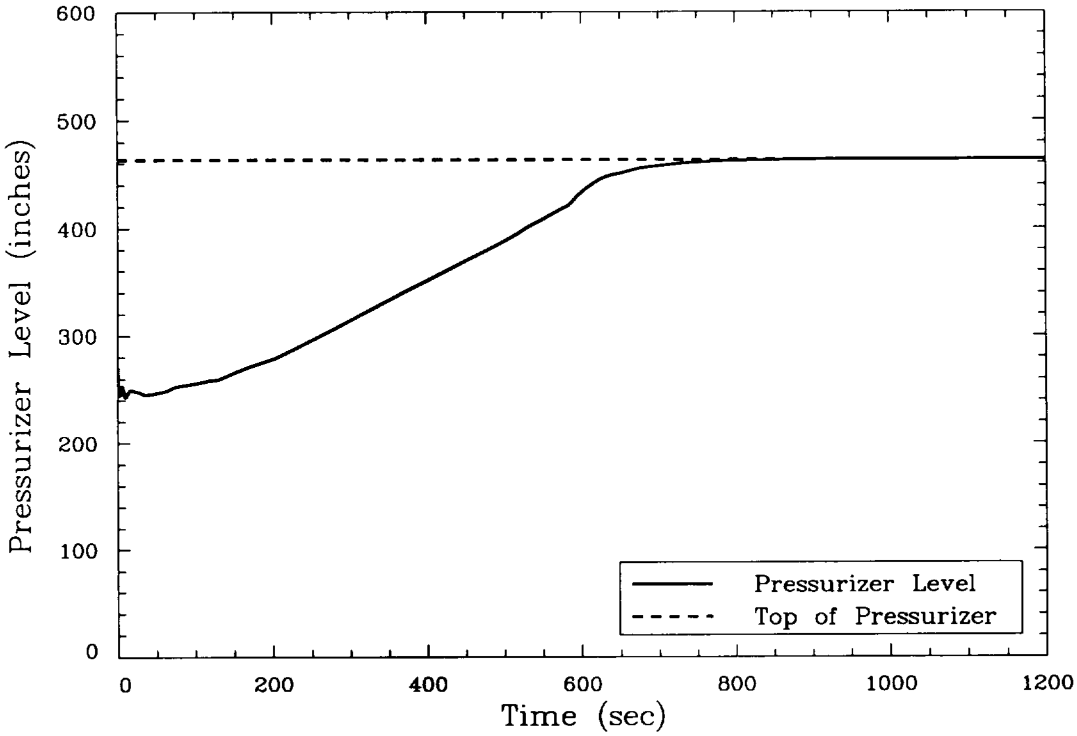

Figure 6.2.26-15 Pressurizer Liquid Level (Pressurizer Overfill Case)

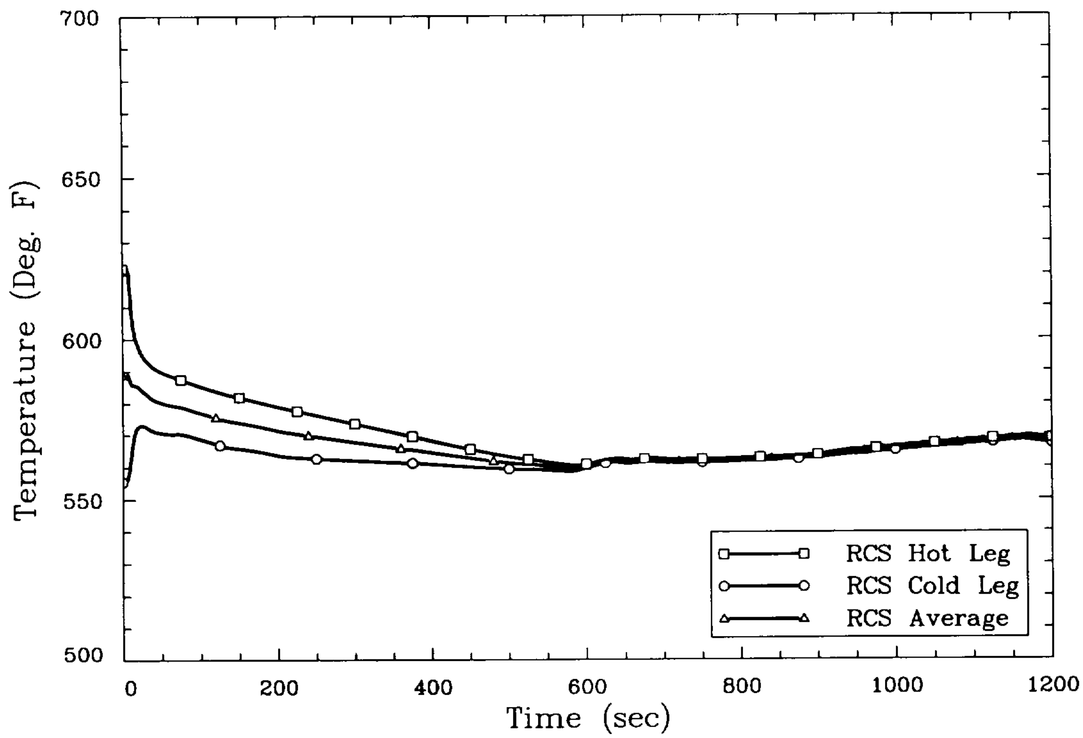

Figure 6.2.26-16 Reactor Coolant System Temperatures (Pressurizer Overfill Case)

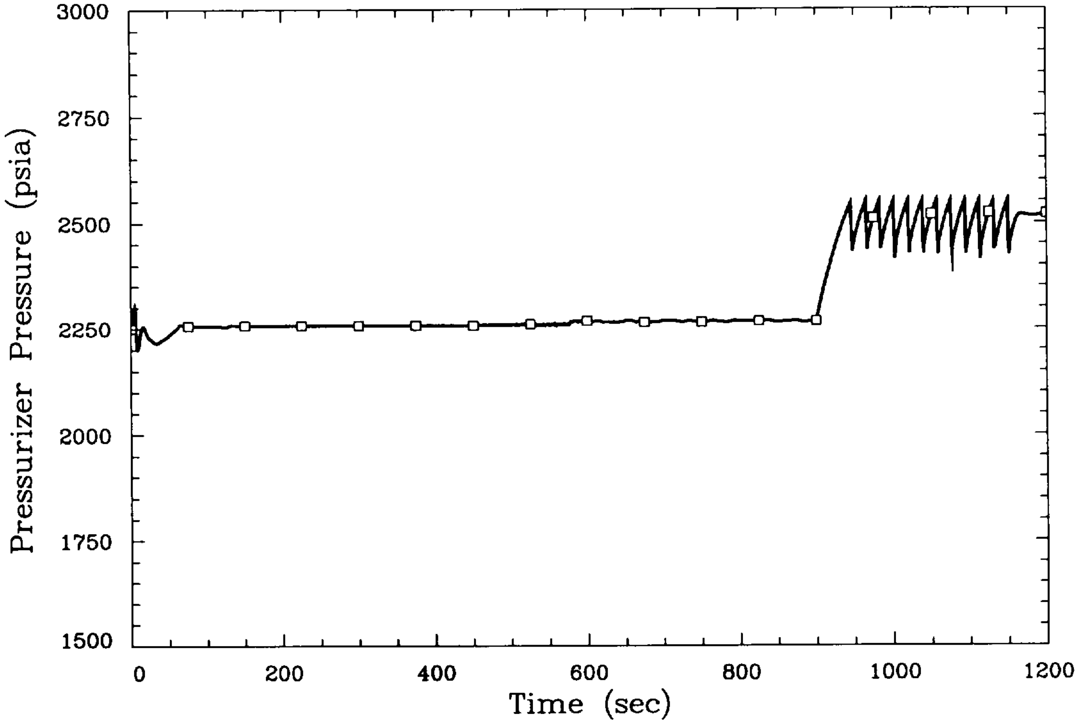

Figure 6.2.26-17 Pressurizer Pressure (Pressurizer Overfill Case)

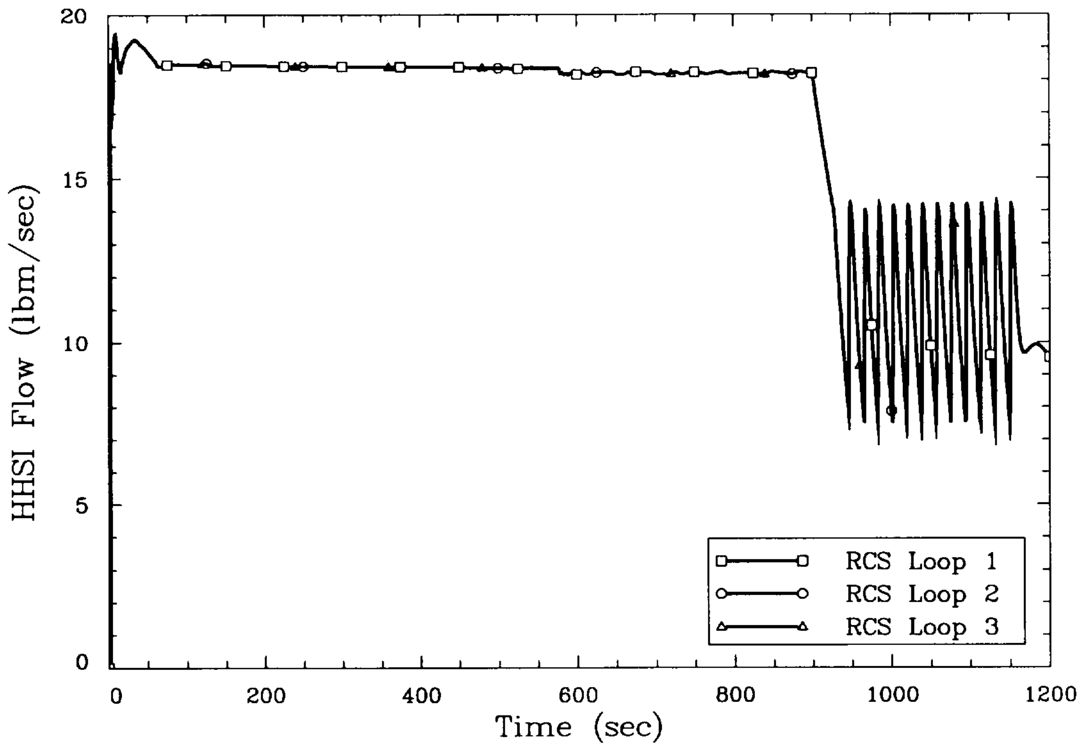

Figure 6.2.26-18 Safety Injection Flow Rate (Pressurizer Overfill Case)

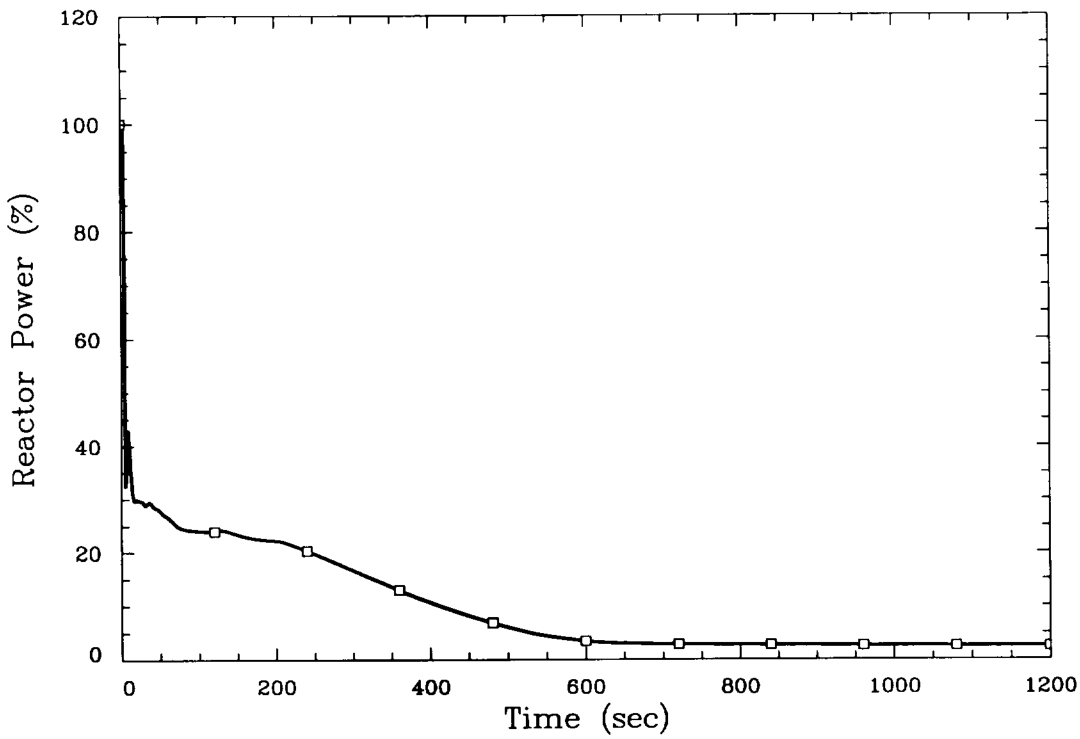

Figure 6.2.26-19 Reactor Power (Pressurizer Overfill Case)

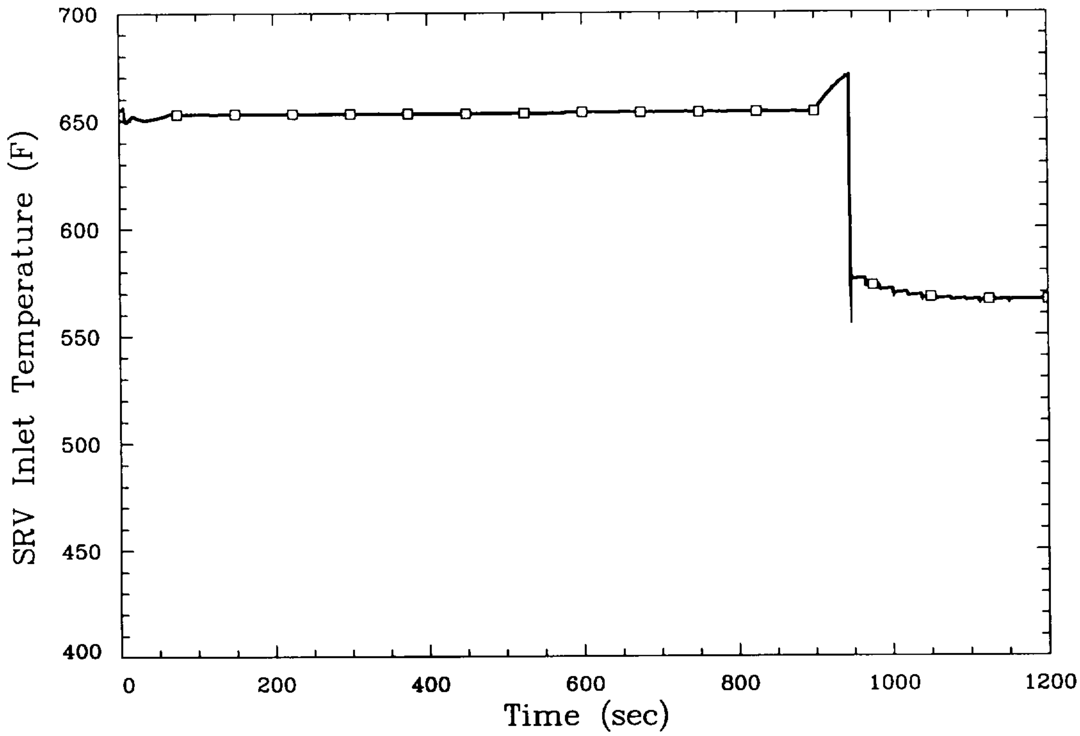

Figure 6.2.26-20 Pressurizer Safety Relief Valve Inlet Temperature (Pressurizer Overfill Case)

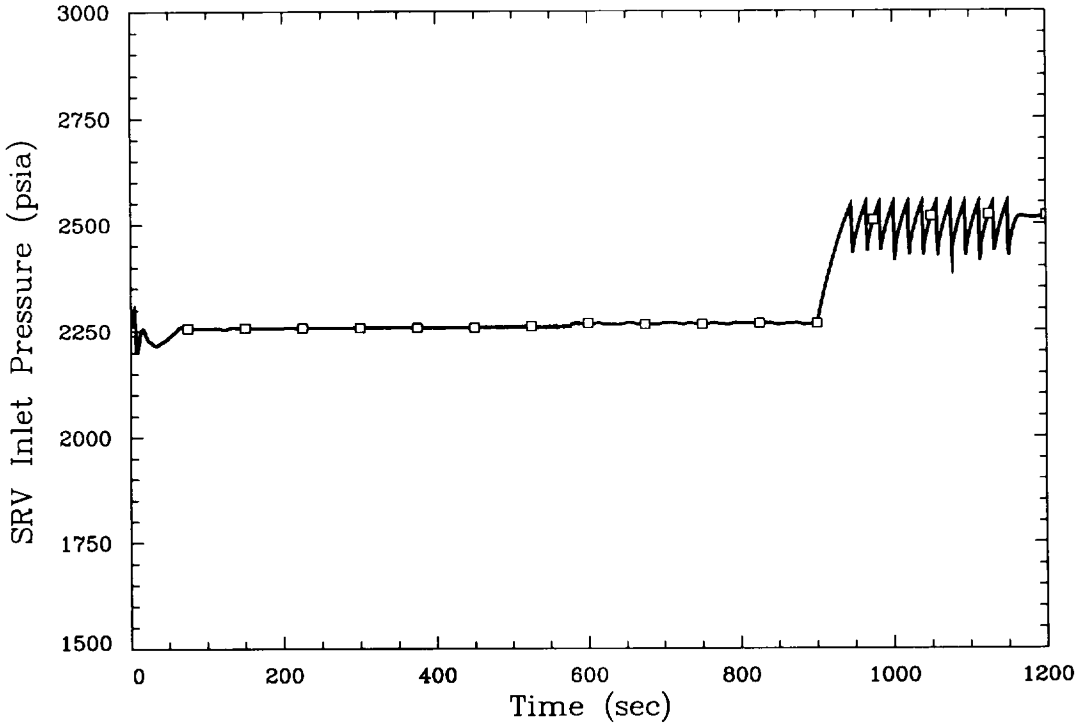

Figure 6.2.26-21 Pressurizer Safety Relief Valve Inlet Pressure (Pressurizer Overfill Case)

Power Uprate Analysis Supplement – Spent Fuel Pool Cooling

Impact of Power Uprate on Fuel Pool Cooling and Cleanup System

Introduction and Background

The Balance of Plant (BOP) Licensing Report (Enclosure 7 to CP&L Technical Specification Amendment Application for steam generator replacement (SGR), SERIAL: HNP-00-142, dated October 4, 2000) addressed the impacts of SGR on the Fuel Pool Cooling and Cleanup Systems (FPCCS). The following evaluation addresses the impact on the FPCCS due to increasing the rated thermal power of the core from 2775 MWt to 2900 MWt (herein referred to as Uprate).

This evaluation incorporates and is consistent with the data and evaluations submitted in the previous license amendment request for the SGR and the license amendment request to place Spent Fuel Pools C and D (SFP C/D) in service. The evaluation describes the impacts on all four Spent Fuel Pools (A, B, C and D, herein referred to as "SFP A/B/C/D"). The evaluation is based on the restricted decay heat load in SFP C/D of 1.0E+06 BTU/hr (1 MBTU/hr) established in the pending license amendment request (SERIAL:HNP-98-188, dated December 23, 1998) to place SFP C/D in service. This evaluation also includes the benefits of the improved Component Cooling Water (CCW) impeller discussed in Section 2.6 of the BOP License Report (LR) for SGR. CCW maximum supply temperatures are as described in the BOP LR. The overall conclusions from the following evaluation therefore bound the combined effects of SGR, Uprate and the 1 MBTU/hr decay heat load in the pending license amendment request for SFP C/D.

System Description

As described in the Final Safety Analysis Report (FSAR), the function of the FPCCS is to provide cooling to both the operating spent fuel pools while maintaining the water quality in the pools. These functions are performed by the FPCCS cooling, cleanup and skimmer subsystems. The equipment that services SFP A/B is separate from the equipment that services SFP C/D. Each cooling system is composed of two redundant cooling loops, each with a heat exchanger, cooling pump, strainers and associated piping. Each cleanup system is composed of two pumps, a single demineralizer and associated piping. Each skimmer system has skimmer service connections, a skimmer pump and a skimmer filter.

The redundant cooling loops have diverse power supplies. Each cooling loop is capable of removing 100% of the normally expected decay heat load from spent fuel stored in the associated SFPs (either SFP A/B or SFP C/D). The two cooling pumps for each pair of pools (A/B or C/D) are powered from separate electric buses, which are powered from an emergency diesel generator in the event of the interruption of the normal power supply. The heat exchangers are of the shell and straight tube type with the water from the fuel pools running through the tubes. The heat is transferred to the CCW system on the shell side. The Non-essential header of the plant's CCW system supplies all the CCW flow for

all four SFP heat exchangers. The CCW system is described in Section 2.6 of the BOP License Report and FSAR Section 9.2.2

The Uprate project does not require configuration changes to the FPCCS.

Evaluation

Decay heat loads for the Uprate analyses are presented in Table 1. These decay heat loads were conservatively calculated using a technique that is consistent with Standard Review Plan (SRP) 9.1.3. The analyzed decay heat loads represent equilibrium conditions in SFP A/B after several cycles of operation at the Uprated power level. The decay heat loads for the HNP-specific cases listed in SRP 9.1.3 for a spent fuel pool with a storage capacity greater than 1-1/3 core are not tabulated because the cases presented include all of the spent fuel stored in the HNP pools including fuel shipped from other CP&L sites. The HNP-specific cases have been shown to bound the SRP case decay heat loads.

The decay heat load for the Uprate normal operation case is the sum of two contributors. The first is the equilibrium decay heat calculated at the start of a refueling outage. This value accounts for the base heat load in the SFP A/B slowly increasing to an equilibrium value as the core design transitions to the equilibrium Uprate design. The second contributor is the decay heat from an equilibrium discharge batch. The batch is assumed to have decayed for 25 days from shutdown. The sum of these two contributors was evaluated as valid for refueling outages as short as 20 days. The difference between 20 and 25 days is accounted for by the fact that the new cycle operation does not build in equilibrium decay heat instantaneously.

Table 1
Spent Fuel Pool Heat Duty
(MBTU/hr)

Operating Condition	Pre-Uprate A/B Pools	Pre-Uprate C/D Pools[1]	Uprate A/B	Uprate C/D Pools[1]
Incore Shuffle	16.84	1.0	22.17	1.0
Full Core Offload Shuffle	35.06	1.0	40.56	1.0
Full Core Offload Post Outage	35.87	1.0	42.46	1.0
Normal Operations	16.84	1.0	16.45	1.0

[1] Proposed heat load from pending license amendment application to place SFP C/D in service.

Refueling Offload Conditions

For the Uprate analyses, an analytical limit of 140°F is used on the SFP temperature for normal refueling cases. This is an increase from the 137°F used in the current FSAR analysis. An analytical limit of 150°F is used on the SFP temperature for an emergency core offload case. As described in the FSAR, the single active failure assumption used for the normal scenarios (either an Incore shuffle or a Full Core Offload at end of cycle) is the assumed loss of one of the two cooling trains for SFP A/B (a pump and heat exchanger).

The Uprate analyses have been performed by revising the single active failure assumption to be a loss of just a single SFP cooling pump. The evaluation of SFP temperature assumes the single remaining cooling pump is aligned to both heat exchangers and the CCW flow is split between the two heat exchangers. This arrangement provides greater cooling capacity for a single cooling pump. The analysis of SFP temperature is also a function of the CCW supply temperature and CCW flow. The CCW supply flow conservatively assumes only a single CCW pump is available. Changes to the existing controls will be made to account for the cooling pump lineup being used, the increased decay heat from the Uprate core, and the increased CCW flow. The SRP acceptance criterion of 140°F will be used for the equilibrium SFP temperature.

The SRP acceptance criterion specifies that pool temperature must be predicted to be less than boiling for the abnormal case. CP&L analyses show that the pool temperature will remain ≤ 150°F during emergency conditions. As stated in the FSAR, the pool concrete design temperature is 150°F. For the case involving the Full Core Offload post-outage, a single active failure is not assumed consistent with SRP 9.1.3.

Impact of Design Basis Accident on SFP Temperature

Spent Fuel Pool conditions concurrent with a Design Basis LOCA are evaluated in the FSAR. CP&L's acceptance criterion for the evaluation is that the SFP temperatures do not exceed 150°F. In the postulated accident, CCW supply is assumed to be interrupted at the initiation of the accident. Using the decay heat for Normal Operations as shown in Table 1, the SFP A/B is conservatively calculated to heat-up at a rate of 4.26°F per hour. The starting pool temperature of SFP A/B is conservatively estimated to be 123.5°F and is conservatively postulated based on Service Water Temperatures to the CCW heat exchanger at limiting values and CCW supply temperature at 105°F. The containment analyses assume that the Non-essential header is restored at 5.0 hours. Therefore, the SFP A/B temperature will not exceed 150°F subsequent to a postulated Design Basis LOCA with the new heat loads. The predicted equilibrium SFP temperatures are below 150°F after restoration of cooling for SFP A/B and SFP C/D. The SFP A/B is the limiting pool based on decay heat. SFP C/D are also predicted to not exceed 150°F during the Design Basis LOCA. Therefore, adequate cooling is provided to SFP A/B/C/D following a Design Basis LOCA.

Impact of RCS Cooldown on SFP Temperature

The Residual Heat Removal (RHR) phase of the RCS cooldown analyses impacts the predicted SFP temperatures. When RHR is initiated, the CCW supply temperature is conservatively assumed in the FSAR to reach 120°F and the flow to the SFP heat exchangers is reduced due to the added CCW flow path to the RHR heat exchanger. The pool equilibrium temperatures during RCS cooldown were analyzed based on the new CCW flow balance, CCW supply temperature, SGR/Uprate and a bounding heat load for Normal Operations. The peak temperatures for both SFP A/B and SFP C/D were conservatively calculated to be ≤ 140 °F with only a single SFP cooling train in service for each pair of SFPs (A/B and C/D). Since the resulting temperatures are less than the listed maximum value for other SRP normal operation cases (e.g. Incore Shuffle), it is acceptable.

Conclusions

The existing FPCCS design is acceptable to support operation of SFPs A/B for SGR/Uprate conditions. The FPCCS design is also acceptable to support operation of SFPs C/D under SGR/Uprate conditions in accordance with the pending amendment request to place SFPs C/D in service. These conclusions are based on the following:

1. The results of analysis of the FPCCS under SGR/Uprate conditions with C/D fuel pool in service demonstrate the Post-LOCA Spent Fuel Pool heat-up temperatures will remain ≤ 150°F, which is less than the acceptance criteria limit of boiling during emergency conditions.

2. The Uprate conditions remain acceptable for normal operations because the predicted temperatures remain ≤140°F.