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FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4

UPRATING LICENSING REPORT

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LIST OF ACRONYMS

AC	Alternating Current
ADV	Atmospheric Dump Valves
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
BAST	Boric Acid Storage Tank
BELOCA	Best Estimate Loss-of-Coolant Accident
BOC	Beginning of Cycle
BOCREC	Bottom of Core Recovery
BOL	Beginning of Life
BOP	Balance of Plant
CCWS	Component Cooling Water System
C_D	Discharge Coefficient
CDV	Condenser Dump Valves
CFR	Code of Federal Regulations
COLR	Core Operating Limits Reports
CR	Control Room
CRDM	Control Rod Drive Mechanism
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
CW	Circulating Water
DE	Dose Equivalent
DECL	Double-Ended Cold Leg
DECLG	Double-Ended Cold Leg Guillotine
DEG	Double-Ended Guillotine
DEHL	Double-Ended Hot Leg
DEHLG	Double-Ended Hot Leg Guillotine
DEPS	Double-Ended Pump Suction
DEPSG	Double-Ended Pump Suction Guillotine
DER	Double-Ended Rupture
DF	Decontamination Factors
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DRFA	Debris Resistant Fuel Assembly
E/C	Erosion/Corrosion
ECC	Emergency Containment Coolers

LIST OF ACRONYMS (cont)

ECCFS	Emergency Containment Cooling and Filtering Systems
ECCS	Emergency Core Cooling System
ECF	Emergency Containment Filters
EFPY	Effective Full Power Years
EOL	End of Life
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
ES	Extraction Steam
ESF	Engineered Safety Features
FES	Final Environmental Statement
FHA	Fuel Handling Accident
FW	Feedwater
GDC	General Design Criteria
HELB	High Energy Line Break
HHSI	High Head Safety Injection
HLSO	Hot Leg Switchover
HP	Horsepower
HVAC	Heating, Ventilation and Air Conditioning
ICW	Intake Cooling Water
LBB	Leak Before Break
LBLOCA	Large Break Loss of Coolant Accident
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LOMF	Loss of Main Feedwater
LOOP	Loss of Offsite Power
LPZ	Lowest Population Zone
LTCC	Long Term Core Cooling
MCC	Motor Control Center
M&E	Mass & Energy Release
MFS	Main Feedwater
MS	Main Steam
MSBV	Main Steam Bypass Valve
MSCV	Main Steam Check Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSR	Moisture Separator-Reheater
MSSV	Main Steam Safety Valve

LIST OF ACRONYMS (cont)

MWD/MTU	Megawatt Days/Metric Ton Uranium
MWt	Megawatt Thermal
NCCS	Normal Containment Cooling System
NEMA	National Electrical Manufacturers Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRV	Non-Return Valve
NSSS	Nuclear Steam Supply System
OFA	Optimized Fuel Assembly
OTΔT	Overtemperature Δ Temperature
OPΔT	Overpower Δ Temperature
PCT	Peak Clad Temperature
PDP	Positive Displacement Pump
PINIT	Initial Containment Pressure
PLS	Precautions, Limitations and Setpoint
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PSV	Pressurizer Safety Valve
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RTDP	Revised Thermal Design Procedure
RV	Reactor Vessel
RVHVS	Reactor Vessel Heat Vent System
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SFP	Spent Fuel Pool
SG	Steam Generator
SGBD	Steam Generator Blowdown
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture

LIST OF ACRONYMS (cont)

SI	Safety Injection
SIS	Safety Injection System
SLB	Steamline Break
SFPCS	Spent Fuel Pit Cooling System
SRV	Safety Relief Valve
STDP	Standard Thermal Design Procedure
TDF	Thermal Design Flow
TG	Turbine Generator
TPCW	Turbine Plant Cooling Water
UFSAR	Updated Final Safety Analysis Report
VCT	Volume Control Tank
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

This report summarizes the evaluations performed to justify the acceptability of increasing the NSSS power rating from the present level of 2208 MWt to 2308 MWt (2300 MWt core power). Florida Power and Light Company has undertaken a program to uprate Turkey Point Units 3 and 4 to a maximum NSSS power level of 2308 MWt. The originally licensed maximum core power level is 2200 MWt, which corresponds to an NSSS power output of 2208 MWt when reactor coolant pump thermal output is included. Therefore, the uprating program is designed to increase licensed core power to 2300 MWt, with a total NSSS power output of 2308 MWt. Unless otherwise noted, 100% power in this report refers to a core power level of 2300 MWt. The report follows the format and contains similar content to those previously submitted to the NRC on several approved PWR uprate licensing reports. The capability of the NSSS of Turkey Point Units 3 and 4 to operate at uprated conditions was verified in accordance with guidelines contained in Westinghouse topical report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant." This WCAP methodology was followed by North Anna, Salem, Indian Point #2, Callaway and Vogtle for their core power upratings. This topical report provided the following criteria which formed the basis for the Turkey Point review:

1. The review encompassed all aspects of NSSS design and operation which are impacted by the power uprating. The scope of this review included the NSSS safety analyses, the functional capability of the systems for normal and abnormal plant operations, and the mechanical design of NSSS components and structures.

2. Safety analyses were performed to FSAR quality standards, and evaluated in accordance with criteria and standards that apply to the current Turkey Point Units 3 and 4 operating licenses.

3. Equipment structural designs were evaluated in accordance with the regulatory requirements, codes, and standards to which the equipment was originally built.

4. In general, current NRC approved analytical techniques were used wherever practical to perform analyses required during conduct of the review.

Turkey Point Units 3 and 4, like most PWR plants as originally licensed, have as-designed equipment and system capability to accommodate steam flow rates of at least 5% above the original rating. The increase to higher power is obtained by effective utilization of existing system and equipment margin.

Detailed evaluations of the Nuclear Steam Supply System, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accident analyses and previous licensing evaluations were performed. This report demonstrates that the Turkey Point Units 3 and 4 can safely operate at the requested NSSS power level of 2308 MWt.

The approach used was based on comparing the predicted uprating conditions to the original NSSS 2208 MWt licensed conditions to determine system capability and, where available, the remaining margin in the original plant design at the uprated conditions (i.e., did the original design "envelope" the uprate). To assure that the review was based on current information, the plant modifications and calculations for each system were reviewed for applicability and were included in the analysis as appropriate. Key plant personnel were consulted and current operating data was obtained to gain a perspective on plant performance and operating difficulties that could affect the capability of the plant at the uprated power level. These concerns were addressed in the various task evaluations.

Implementing the uprating at Turkey Point will only require a few minor physical modifications to the plant. Operating parameters are mainly increased in the power conversion systems (e.g., main steam, feedwater and condensate, extraction steam, etc) and then by only approximately 5%, which is within the systems and equipment capability. Where required, setpoints will be adjusted, plant procedures revised, and tests performed to ensure the safe and reliable operation of the units at the uprated conditions. In addition, the safety analyses provided in the FSAR will be updated as reflected with this licensing report.

In accordance with 10CFR50.92, this uprating evaluation has reviewed the predominant plant licensing challenges, and demonstrates that the new conditions can be supported without:

- A significant increase in the probability or consequences of an accident previously evaluated,
- Creating the possibility of a new or different kind of accident from any accident previously evaluated, or
- Resulting in a significant reduction in a margin of safety.

This thermal power uprating involves no significant hazards consideration.

CHAPTER 1

PROGRAM DESCRIPTION

1.0 PROGRAM DESCRIPTION

1.1 LICENSING PERSPECTIVE

Florida Power & Light Company has undertaken a program to uprate the Turkey Point Nuclear Units 3 and 4 to a maximum NSSS power level of 2308 MWt. The original plant was evaluated in most cases for operation at an NSSS power level of 2308 MWt, however, the plant was licensed to operate at an NSSS power level of 2208 MWt. The uprating program is intended to permit operation at the maximum original power level of 2308 MWt.

In addition to uprating, a number of other issues are incorporated in this submittal:

- Steam generator tube plugging of 20% (this would be permitted following approval of Best Estimate LOCA methodology)
- Allow operation within a ±3°F Tavg
- Increase MSSV and PSV tolerance

Turkey Point was licensed in the early 70's as a Westinghouse 3-loop PWR. The review performed shows that the plant continues to meet its licensing basis at the uprated conditions. In many cases the methods and analyses used to demonstrate compliance were upgraded to meet more stringent current NRC criteria. The licensing report clearly shows that operation at 2308 MWt will not affect the health and safety of the public.

1.2 PURPOSE AND OBJECTIVES

The purpose of this licensing report is to provide the basis for the determination that continued safe plant operation can be achieved at the uprated condition. The licensing basis assessment includes a review of the accident analyses, component design issues related to safety, emergency response guidelines, BOP Systems, Technical Specifications and appropriate sections of the Turkey Point Units 3 and 4 UFSAR.

The objective of this review was to provide the technical bases for the uprating.

1.3 DESIGN AND LICENSING CRITERIA

The analyses and evaluations performed in support of the Turkey Point Units 3 and 4 uprating program have been completed in accordance with Westinghouse quality assurance requirements defined in WCAP-8370-A/7800-A, FPL Topical Quality Assurance Report (FPLTQAR 1-76A), and Stone & Webster quality assurance requirements defined in the Stone & Webster Standard Nuclear Quality Assurance Program (SWSNQAP 1-74A), which comply with 10 CFR 50 Appendix B criteria. Equipment reviews and evaluations have been performed in accordance with Westinghouse and industry codes, standards, and regulatory requirements applicable to Turkey Point Units 3 and 4.

Assumptions and acceptance criteria for the various accident analyses are addressed in the respective sections in Chapter 3.0.

1.4 SCOPE SUMMARY

In order to support uprating of Turkey Point Units 3 and 4 to an NSSS power of 2308 MWt, the NSSS performance parameters for the uprating were calculated for a range of temperature and steam generator tube plugging conditions, as described in Chapter 2. Subsequent to development of the SG performance parameters, evaluations or analyses (depending on the extent of the uprating's impact in each area) were performed for accident analyses, NSSS and BOP systems, and NSSS and BOP components, in the areas listed below. For safety-related efforts, the analysts considered the case or cases most conservative for their respective areas. The basis for these determinations and the results of these evaluations and analyses are presented within this uprating licensing report. The listing below follows the order in which the topics addressed are presented in this report:

The following accident analyses were addressed:

- Non-LOCA
- Large and Small Break LOCA
- Steam Generator Tube Rupture
- Containment Integrity
- Equipment Qualification
- Hydrogen Generation

The NSSS and Turbine Generator components were addressed as follows:

- Reactor Vessel
- Reactor Internals
- Reactor Coolant Pumps
- Control Rod Drive Mechanisms
- Reactor Coolant Piping and Supports
- Pressurizer
- Steam Generators
- Fuel
- NSSS Auxiliary Systems Components
- Turbine Generator Components

The NSSS and Turbine Generator systems were addressed as follows:

- NSSS Fluid Systems
- Control Systems
- Protection Systems

- NSSS/BOP Interface Systems
- Turbine Generator Systems

The BOP systems and components were addressed as follows:

- Main Steam System
- Steam Dump System
- Condensate and Feedwater System
- Feedwater Heaters
- Steam Generator Blowdown System
- Condensate Polishing System
- Feedwater Heater Vent and Drain System
- Extraction Steam System
- Main Condenser
- Circulating Water System
- Turbine Plant Cooling Water System
- Intake Cooling Water System
- Control Systems
- Electrical Systems
- HVAC Systems
- Miscellaneous Systems
- BOP Components

The goal of the analyses and evaluations presented in this report is to demonstrate that Turkey Point Units 3 and 4 continue to comply with the applicable industry codes, standards, and licensing criteria at the uprated conditions.

CHAPTER 2

DETERMINATION OF NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESIGN OPERATING CONDITIONS

2.0 DETERMINATION OF NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESIGN OPERATING PARAMETERS

2.1 DISCUSSION OF DESIGN PARAMETERS

2.1.1 Introduction and Discussion of Input Parameters

Design performance capability parameters were developed for the Turkey Point Units 3 and 4 Thermal Uprate Program to encompass the following features:

- NSSS uprated power level of 2308 MWt;
- A range of primary temperatures, based on the current licensed T_{avg} value of 574.2 $\pm$ 3°F;
- A range of steam generator tube plugging level of 0-20%. (Although the LBLOCA BASH used a 5% maximum tube plugging level, following approval by the NRC of the Best Estimate LOCA (BELOCA) Methodology, FPL plans to make a submittal to the NRC to take credit for the 20% tube plugging level.)

To support the uprating for the Turkey Point units, the parameters set(s) used were the most conservative for the affected evaluations and analyses.

2.1.2 Discussion of Parameter Cases

Table 2.1-1 presents the various cases that were provided for use in the uprating analysis. These cases were developed to optimize plant operation and flexibility while at the same time maximizing electrical production. The column labeled "current" reflects the current design conditions at 0% tube plugging, and is provided for comparison only. Cases 1 and 2 provide parameters over the range of reactor vessel T_{avg} values from 571.2 - 577.2°F, with a steam generator tube plugging level of 0%, and a maximum feedwater temperature value of 443°F. Cases 3 and 4 are identical to cases 1 and 2, except that the steam generator tube plugging level assumed is 20% (the effect of this change can be seen in the steam generator parameters).

2.2 CONCLUSIONS

The design performance capability parameters which provide RCS parameters for the uprating analyses and evaluations are provided in Table 2.1-1. The set(s) of parameters which were most conservative for the particular analyses or evaluations were used, in order to bound the range of conditions specified in Table 2.1-1.

TABLE 2.1-1
Design Performance Capability Parameters for Turkey Point Units 3 and 4

THERMAL DESIGN PARAMETERS	Current		Uprated Cases 1-4		
NSSS Power, %	100		104.5		
MWt	2208		2308		
10^6 BTU/hr	7534		7875.2		
Reactor Power, MWt	2200		2300		
10^6 BTU/hr	7506.7		7847.9		
Thermal Design Flow, Loop gpm	89,500		85,000		
Reactor Coolant Pressure, psia	2250		2250		
Core Bypass, %	4.5		6.0		
		Case 1	Case 2	Case 3	Case 4
Reactor Coolant Temperature, °F					
Core Outlet	604.7	611.3	605.6	611.3	605.6
Vessel Outlet	602.3	607.8	602.0	607.8	602.0
Core Average	576.6	580.5	574.4	580.5	574.4
Vessel Average	574.2	577.2	571.2	577.2	571.2
Vessel/Core Inlet	546.2	546.6	540.4	546.6	540.4
Steam Generator Outlet	546.0	546.4	540.1	546.4	540.1
Steam Generator					
Plugging Level %	0	0	0	20	20
Steam Temperature, °F	516.0	522.8	516.3	515.2	508.6
Steam Pressure, psia	785	832	787	779	736
Steam Flow, 10^6 lb/hr total	9.60	10.17	10.16	10.16	10.14
Feed Temperature, °F	436.5	443	443	443	443
Moisture, % max.	0.25	0.25	0.25	0.25	0.25
App. Fouling Factor, hr. sq. ft. °F/BTU	0.00021	0.00005	0.00005	0.00005	0.00005
Zero Load Temperature, °F	547	547	547	547	547
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	88,500/266				
Mechanical Design Flow, gpm	100,400				
Minimum Measured Flow, gpm total		264,000	264,000	264,000	264,000
Best Estimate Flow, gpm		93,600	93,600	89,000	89,000

CHAPTER 3

ACCIDENT ANALYSES
AND
EVALUATIONS

3.1 INTRODUCTION

The accident analyses have been re-analyzed or evaluated for the Turkey Point Units to support operation at the uprated NSSS power level of up to 2308 MWt. The thermal design parameters assumed in these analyses may be found in Table 2.1-1. The computer codes and methods utilized for these analyses have all been previously approved by the NRC unless otherwise noted.

3.2 NON-LOSS OF COOLANT ACCIDENT (NON-LOCA) EVENTS AND STANDBY SAFETY FEATURES ANALYSES

All of the UFSAR Chapter 14 non-LOCA analyses applicable to the Turkey Point Units 3 and 4 were reviewed to determine their continued acceptability based upon plant operation at the uprated conditions. The following non-LOCA events were either reanalyzed or evaluated for the Turkey Point Units 3 and 4 conditions consistent with the uprated conditions identified in Table 2.1-1.

1. Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from a Subcritical Condition (Section 3.2.1)
2. Uncontrolled RCCA Withdrawal at Power (Section 3.2.2)
3. RCCA Drop (Section 3.2.3)
4. Chemical and Volume Control System (CVCS) Malfunction (Section 3.2.4)
5. Startup of an Inactive Reactor Coolant Loop (Section 3.2.5)
6. Excessive Heat Removal Due To Feedwater System Malfunctions (Section 3.2.6)
7. Excessive Load Increase Incident (Section 3.2.7)
8. Loss of Reactor Coolant Flow (Section 3.2.8)
   Partial/Complete Loss of Forced Reactor Coolant Flow (Section 3.2.8.1)
   Locked Rotor/Shaft Break (Section 3.2.8.2)
9. Loss of External Electrical Load and/or Turbine Trip (Section 3.2.9)
10. Loss of Normal Feedwater (Section 3.2.10)
11. Loss of Non-Emergency AC Power to the Plant Auxiliaries (Section 3.2.11)
12. Main Steam Line Break Core Response (Section 3.2.16)
13. Rupture of a Control Rod Drive Mechanism Housing - RCCA Ejection (Section 3.2.17)

All of the above events were reanalyzed except for those detailed in Sections 3.2.5 and 3.2.16. The evaluations of all events are detailed in their respective licensing report sections. The analyses incorporating Revised Thermal Design Procedure (RTDP) (References 1 and 2), are the current licensing basis analysis for Turkey Point Units 3 and 4. Startup of an Inactive Coolant Loop was considered in the original design bases for the plant. However, subsequent to initial plant operation, a change to the allowable plant operating conditions was made to prohibit operation at power with a loop out of service (i.e., N-1 loop operation). The current Technical Specifications require that all three (3) reactor coolant pumps be operating for reactor power operation and prohibits operation with an inactive loop. Therefore, since N-1 loop operation is prohibited at power, the startup of an inactive reactor coolant loop event as considered in the original plant design bases is precluded. The main steam line break core response limiting event was analyzed at hot zero power conditions, and is therefore not affected by uprating.

All non-LOCA licensing basis analyses have been analyzed using NRC approved methods and computer codes. The results of all of the analyses and evaluations demonstrate that applicable safety analysis acceptance criteria have been satisfied at the Uprated conditions detailed in Table 2.1-1.

References

1. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non Proprietary), April 1989.

2. NRC Letter, T. F. Plunkett (FPL) to USNRC, "Proposed License Amendments - Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint," L-95-131, dated May 5, 1995.

3.2.1 Uncontrolled RCCA Withdrawal From A Subcritical Condition

3.2.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by operator action or a malfunction of the reactor control rod drive system. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 3.2.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal or by reducing the core boron concentration. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The rods are physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod banks with the maximum combined worth at maximum speed which is well within the capability of the protection system to prevent core damage.

Should a continuous RCCA withdrawal be initiated and assuming the source and intermediate range indication and annunciators are ignored, the transient will be terminated by the following automatic protective functions.

A. Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed. It is automatically blocked when either the intermediate or power range flux channel indicates a flux level above the source range cutoff level. It is automatically reinstated when both intermediate and power range channels indicate a flux level below the source range cutoff power level and the bypass switch is returned to the normal position.

B. Intermediate range rod stop - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This rod stop may be manually bypassed when two out of the four power range channels indicate a flux level above approximately 10 percent of the full-power flux. It is automatically reinstated when three of the four power range channels are below this value.

C. Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed, when two of the four power range channels are reading above approximately 10 percent of the full-power flux and is automatically reinstated when three of the four channels indicate a flux level below this value.

D. Power range flux level trip (low setting) - actuated when two out of the four power range channels indicate a flux level above approximately 25 percent of the full-power flux. This trip function may be manually bypassed when two of the four power range channels indicate a flux level above approximately 10 percent of the full-power flux and is automatically reinstated when three of the four channels indicate a flux level below this value.

E. Power range flux level trip (high setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint, usually ≤109 percent of the full-power flux. This trip function is always active.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power increase results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup transient since it limits the power to a tolerable level prior to external control action. After the initial power increase, the nuclear power is momentarily reduced and then if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup transient by the above protection channels prevents core damage. In addition, the reactor trip from high pressurizer pressure serves as backup to terminate the event before an overpressure condition could occur.

3.2.1.2 Input Parameters and Assumptions

The accident analysis employs the Standard Thermal Design Procedure (STDP) methodology. The RTDP methodology does not apply to zero power events because the DNBR sensitivities used to define the design limit DNBR value do not extend to the zero power condition. The use of STDP methodology stipulates that the Reactor Coolant System (RCS) flow rate will be based on a fraction of the Thermal Design Flow for two RCPs operating and that the RCS pressure is at a conservatively low value which accounts for uncertainty due to instrument error. Since the event is analyzed from hot zero power, the steady-state STDP uncertainties on core power and RCS average temperature are not considered in defining the initial conditions.

In order to obtain conservative results for the analysis of the uncontrolled RCCA bank withdrawal from subcritical event, the following assumptions are made concerning the initial reactor conditions:

A. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, the least negative design value is used.

B. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Accordingly, the most-positive moderator temperature coefficient is used since this yields the maximum rate of power increase.

C. The analysis assumes the reactor to be at hot zero power conditions with a nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature (i.e., shutdown conditions). The higher initial system temperature yields a larger fuel-to-moderator heat transfer coefficient, a larger specific heat of the moderator and fuel, and a less-negative (smaller absolute magnitude) Doppler coefficient. The less-negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis assumes the initial effective multiplication factor (K_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

D. Reactor trip is assumed on power range high neutron flux (low setting). The most adverse combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release is taken into account. The analysis assumes a 10 percent uncertainty in the power range flux trip setpoint (low setting), raising it from the nominal value of 25 percent to a value of 35 percent; no credit is taken for the source and intermediate range protection. Figure 3.2.1-1 shows that the rise in nuclear power is so rapid that the effect of error in the trip setpoint on the actual time at which the rods release is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

E. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum speed (45 in/min, which corresponds to 72 steps/min).

F. The DNB analysis assumes the most-limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their highest worth position.

G. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

3.2.1.3 Description of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcriticality is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE (Reference 1), is used to calculate the core average nuclear power transient, including the various core feedback effects, i.e., Doppler and moderator reactivity. Next, the FACTRAN computer code (Reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the average heat flux calculated by FACTRAN is used in the THINC-IV computer code (References 3 & 4) for transient DNBR calculations.

3.2.1.4 Acceptance Criteria

The uncontrolled rod cluster control assembly bank withdrawal from subcritical event is considered an ANS Condition II event, a fault of moderate frequency, and is analyzed to ensure that the core and reactor coolant system are not adversely affected. This is demonstrated by showing that there is little likelihood of DNB and core damage. It must also be shown that the peak hot spot fuel and clad temperatures remain within acceptable limits, although for this event, the heat up is relatively small.

3.2.1.5 Results

The calculated sequence of events is shown in Table 3.2.1-1. The transient results are shown in Figures 3.2.1-1 through 3.2.1-4. The results of the analysis determined that the DNBR safety analysis limit was met and that the peak fuel centerline temperature was less than the temperature at which fuel melt occurs. The peak clad surface temperature is considerably less than 2700°F.

3.2.1.6 Conclusions

In the event of an RCCA withdrawal event from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature results in a minimum DNBR greater than the safety analysis limit value. Furthermore, since the maximum fuel temperatures predicted to occur during this event are much less than those required for clad damage (2700°F) or fuel (4800°F) melting to occur, no cladding or fuel damage is predicted as a result of this transient at the uprated conditions.

3.2.1.7 References

1. Barry, R. F., Jr. and Risher, D. H., "TWINKLE, a Multi-dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Non-proprietary).

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, December 1989.

3. Chelemer, H., and Hochreiter, L. E., "Application of the THINC-IV Program to PWR Design," WCAP-8195, February 1989.

4. Chelemer, H., Chu, P. T., and Hochreiter, L. E., "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, "WCAP-7956, February 1989.

Table 3.2.1-1

Sequence of Events - Uncontrolled RCCA Withdrawal from Subcritical Event

Event	Time (sec)
Initiation of Uncontrolled RCCA Withdrawal	0.0
Power Range High Neutron Flux, Low Setpoint Reached	10.31
Peak Nuclear Power Occurs	10.45
Rods Begin to Fall	10.81
Minimum DNBR occurs	12.38
Peak Average Clad Temperature Occurs	12.66
Peak Average Fuel Temperature Occurs	12.96
Peak Fuel Centerline Temperature Occurs	14.41

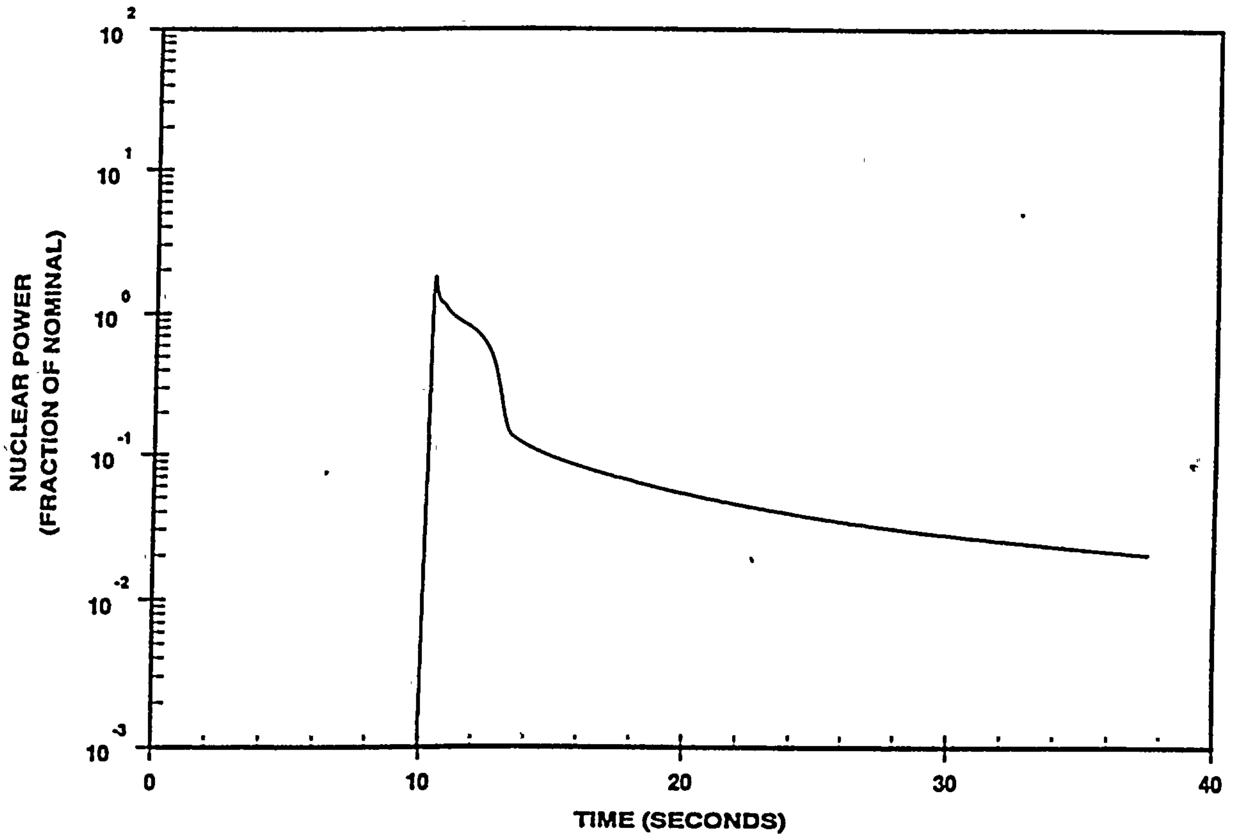


Figure 3.2.1-1 Nuclear Power Transient During Uncontrolled RCCA Withdrawal From Subcritical

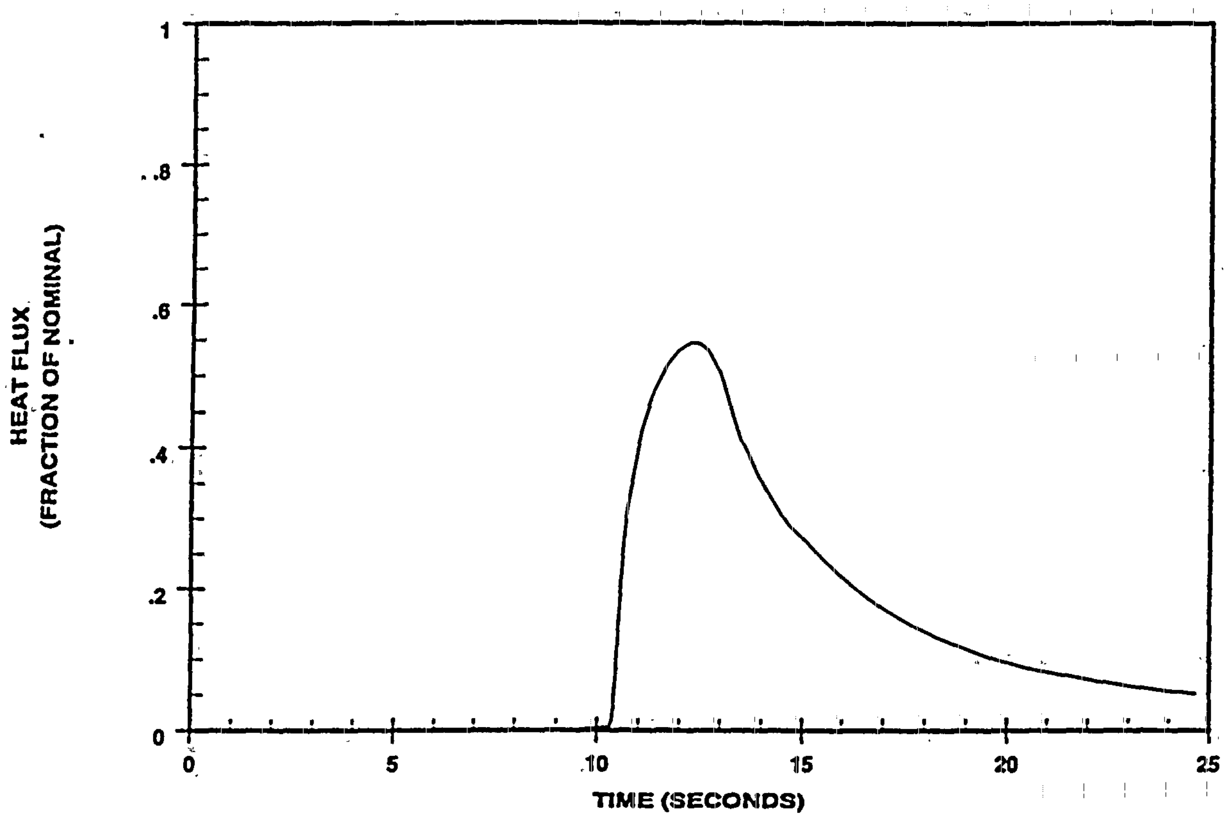


Figure 3.2.1-2 Heat Flux Transient During Uncontrolled RCCA Withdrawal From Subcritical

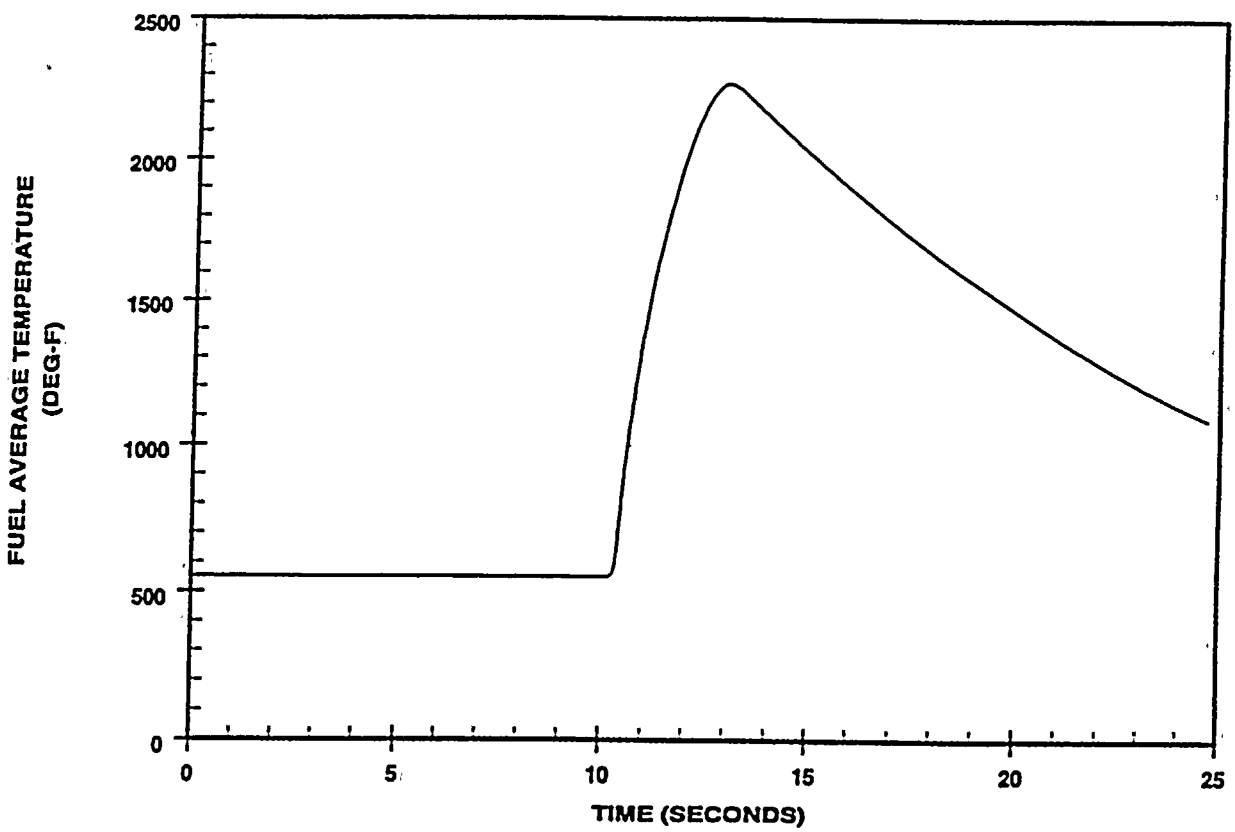


Figure 3.2.1-3 Hot Spot Average Fuel Temperature Transient During Uncontrolled RCCA Withdrawal From Subcritical

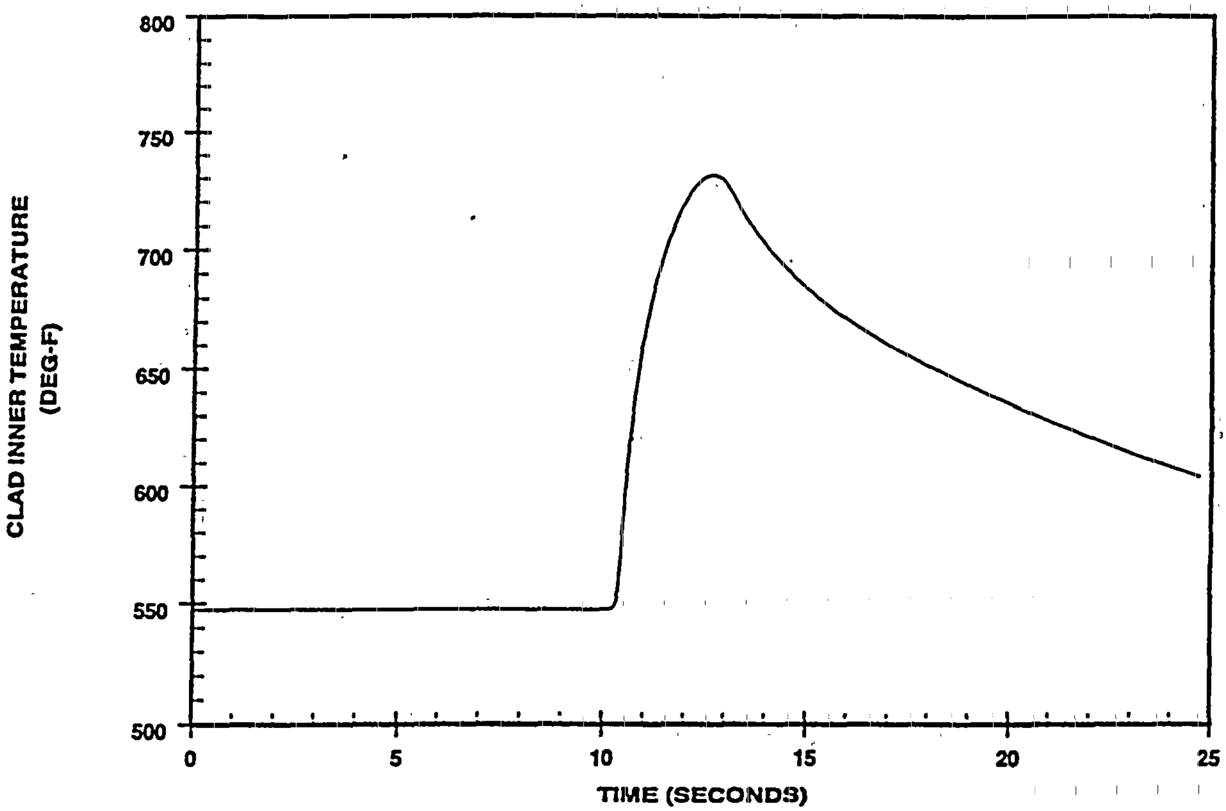


Figure 3.2.1-4 Hot Spot Clad Inner Temperature During Uncontrolled RCCA Withdrawal From Subcritical

3.2.2 Uncontrolled RCCA Bank Withdrawal At Power

3.2.2.1 Identification of Causes and Accident Description

An uncontrolled RCCA withdrawal at power which causes an increase in core heat flux may result from faulty operator action or a malfunction in the rod control system. Immediately following the initiation of the accident, steam generator heat removal rate lags behind the core power generation rate until the steam generator pressure reaches the setpoints of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in DNB and/or fuel centerline melt. Therefore, to avoid damage to the core, the reactor protection system is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value or the fuel rod linear heat generation rate (kw/ft) limit is exceeded.

The automatic features of the reactor protection system which prevent core damage in an RCCA bank withdrawal incident at power include the following.

A. Power range high neutron flux instrumentation actuates a reactor trip on neutron flux if two-out-of-four channels exceed an overpower setpoint.

B. Reactor trip actuates if any two-out-of-three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature, and coolant average pressure to protect against DNB.

C. Reactor trip actuates if any two-out-of-three ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable heat generation rate (kw/ft) is not exceeded.

D. A high pressurizer pressure reactor trip, actuated from any two-out-of-three pressure channels, is set at a fixed point. This reactor trip on high pressurizer pressure is less than the set pressure for the pressurizer safety valves.

E. A high pressurizer water level reactor trip actuates if any two-out-of-three level channels exceed a fixed setpoint.

Besides the above-listed reactor trips, there are the following RCCA withdrawal blocks. These are not credited in accident analyses.

A. High neutron flux (one-out-of-four power range)
B. Overpower ΔT (two-out-of-three)
C. Overtemperature ΔT (two-out-of-three)

3.2.2.2 Input Parameters and Assumptions

A number of cases were analyzed assuming a range of reactivity insertion for both minimum and maximum reactivity feedback at various power levels. The cases presented in Section 3.2.2.5 are representative for this event.

For an uncontrolled RCCA bank withdrawal at power accident, the analysis assumes the following conservative assumptions:

A. This accident is analyzed with the Revised Thermal Design Procedure (Reference 2). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

B. For reactivity coefficients, two cases are analyzed.

   1. <u>Minimum Reactivity Feedback</u> A +7 pcm/°F moderator temperature coefficient and a least-negative Doppler-only power coefficient form the basis of the beginning-of-life minimum reactivity feedback assumption.

   2. <u>Maximum Reactivity Feedback</u> A conservatively large positive moderator density coefficient of 0.5 Δk/gm/cc (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler-only power coefficient form the basis of the end-of-life maximum reactivity feedback assumption.

C. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

D. The RCCA trip insertion characteristic is based on the assumption that the highest-worth assembly is stuck in its fully withdrawn position.

E. A range of reactivity insertion rates are examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at a conservative speed (45 in/min, which corresponds to 72 steps/min).

F. Power levels of 10%, 60%, 80%, and 100% are considered.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

3.2.2.3 Description of Analysis

The purpose of this analysis is to demonstrate the manner in which the protection functions described above actuate for various combinations of reactivity insertion rates and initial conditions. Insertion rate and initial conditions determine which trip function occurs first.

The rod withdrawal at power event is analyzed with the LOFTRAN computer code (Reference 1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The program computes pertinent plant variables including temperatures, pressures, power level, and departure from nucleate boiling ratio (DNBR).

3.2.2.4 Acceptance Criteria

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at power accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event.

The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. With respect to peak pressure, the uncontrolled RCCA bank withdrawal at power accident is bounded by the loss of load/turbine trip analysis. The loss of load/turbine trip analysis is described in Section 3.2.9.

The protection features presented in Section 3.2.2.1 provide mitigation of the uncontrolled RCCA bank withdrawal at power transient such that the above criteria are satisfied.

3.2.2.5 Results

Figures 3.2.2-1 and 3.2.2-2 show the transient response for a rapid RCCA bank withdrawal incident (75 pcm/sec) starting from 60% power with minimum feedback. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because of the rapid reactor trip with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result in the margin to DNB being maintained.

The transient response for a slow RCCA bank withdrawal (1 pcm/sec) from 60% power with minimum feedback is shown in Figures 3.2.2-3 and 3.2.2-4. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the safety analysis limit value.

Figure 3.2.2-5 shows the minimum DNBR as a function of reactivity insertion rate from 100% power for both minimum and maximum reactivity feedback. It can be seen that the two reactor trip functions (high neutron flux and overtemperature ΔT) provide DNB protection over the whole range of reactivity insertion rates. The minimum DNBR is never less than the safety analysis limit value.

Figures 3.2.2-6, 3.2.2-7, and 3.2.2-8 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 80%, 60%, and 10% power, respectively. The results are similar to the 100% power case; however, as the initial power decreases, the range over which the overtemperature ΔT trip is effective is increased. In none of these cases does the DNBR fall below the safety analysis limit value (typical cell 1.43, thimble cell 1.42).

A typical calculated sequence of events for two cases is shown on Table 3.2.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

3.2.2.6 Conclusions

The high neutron flux and overtemperature ΔT reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates (i.e., the minimum value of DNBR is always larger than the safety analysis limit value). The RCS and main steam systems are maintained below 110% of the design pressures. Therefore, the results of the analysis show that an uncontrolled RCCA withdrawal at power does not adversely affect the core, the RCS, or the main steam system and all applicable criteria are met.

3.2.2.7 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

2. Friedland, A.J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-proprietary), April 1989.

Table 3.2.2-1

Sequence of Events - Uncontrolled RCCA Bank Withdrawal at Power Analysis

Case	Event	Time (sec)
60% Power	Initiation of Withdrawal	0.0
Minimum Feedback 75 pcm/sec	High Neutron Flux Setpoint Reached	5.11
	Rods Begin to Fall	5.61
	Minimum DNBR Reached	7.20
60% Power	Initiation of Withdrawal	0.0
Minimum Feedback 1 pcm/sec	Overtemperature ΔT Setpoint Reached	100.14
	Rods Begin to Fall	102.14
	Minimum DNBR Reached	103.2

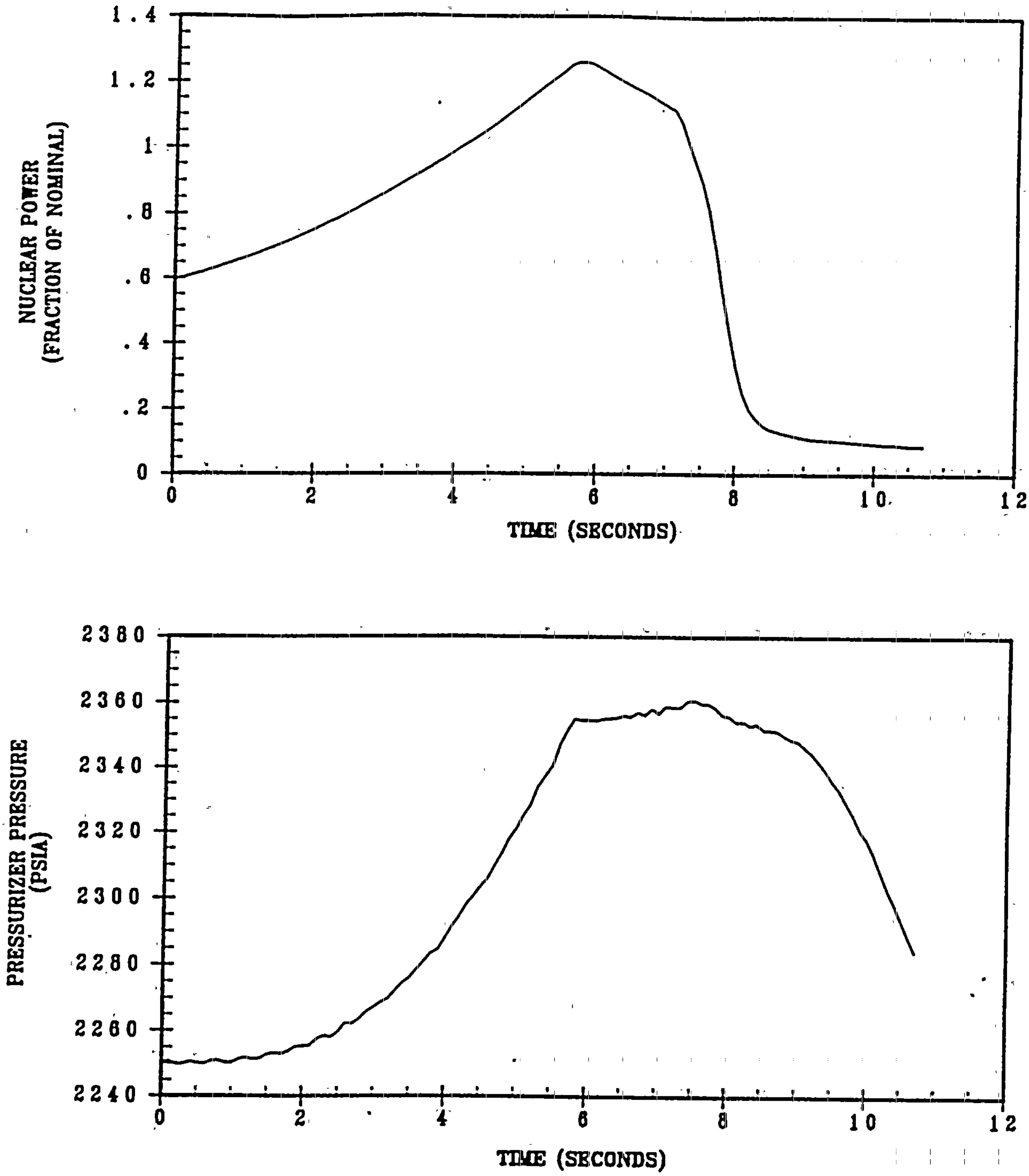


Figure 3.2.2-1 Uncontrolled RCCA Withdrawal 60% Power, Minimum Feedback
(75 pcm/sec withdrawal rate)

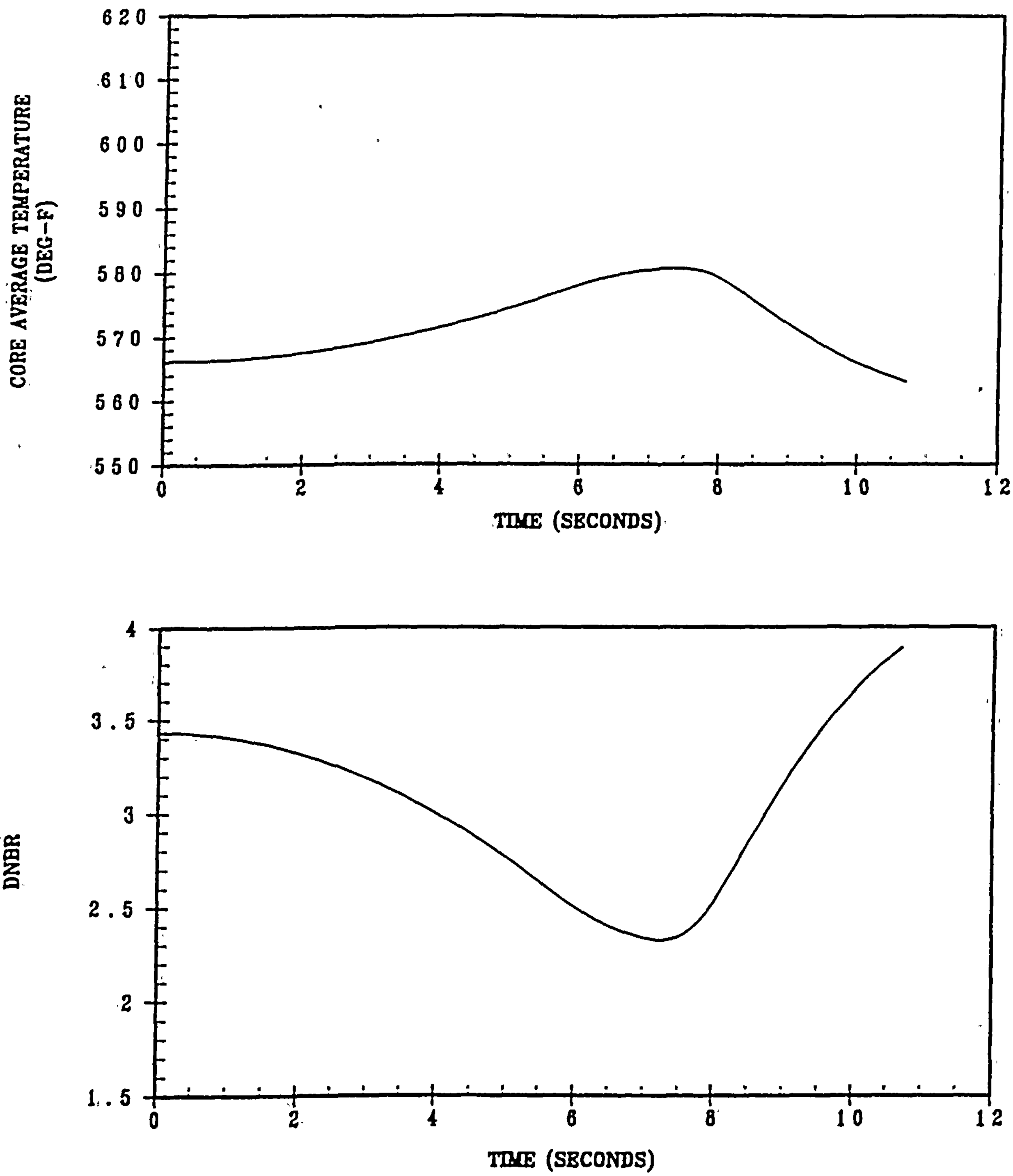
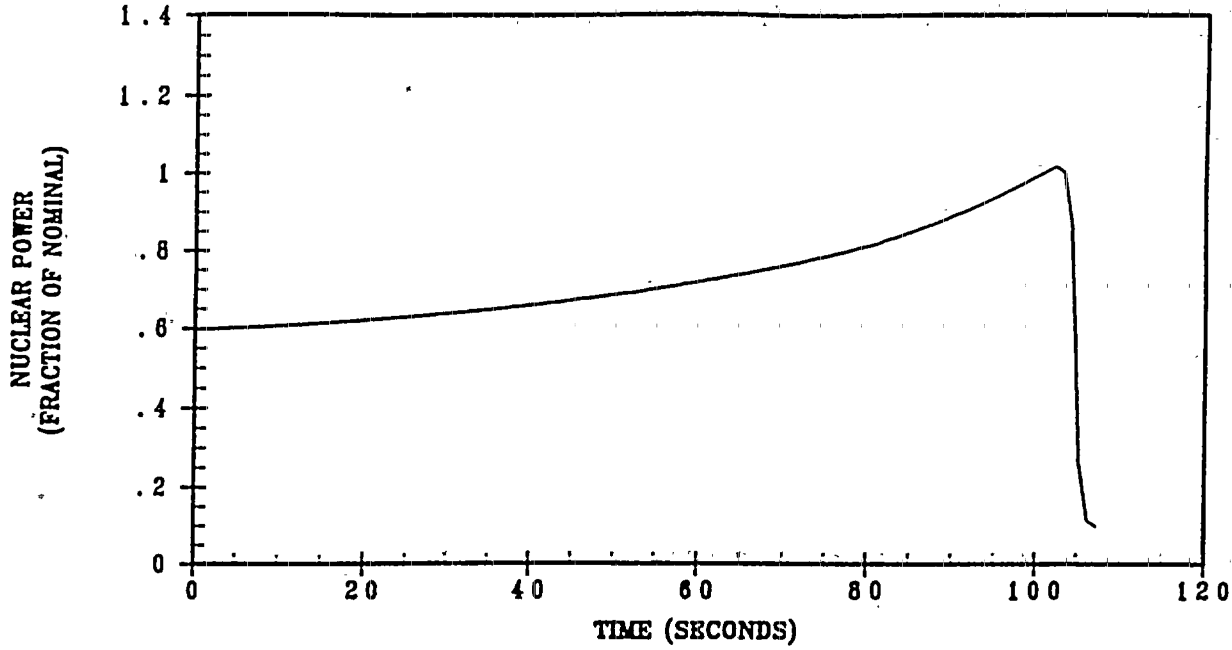


Figure 3.2.2-2 Uncontrolled RCCA Withdrawal 60% Power, Minimum Feedback
(75 pcm/sec withdrawal rate)



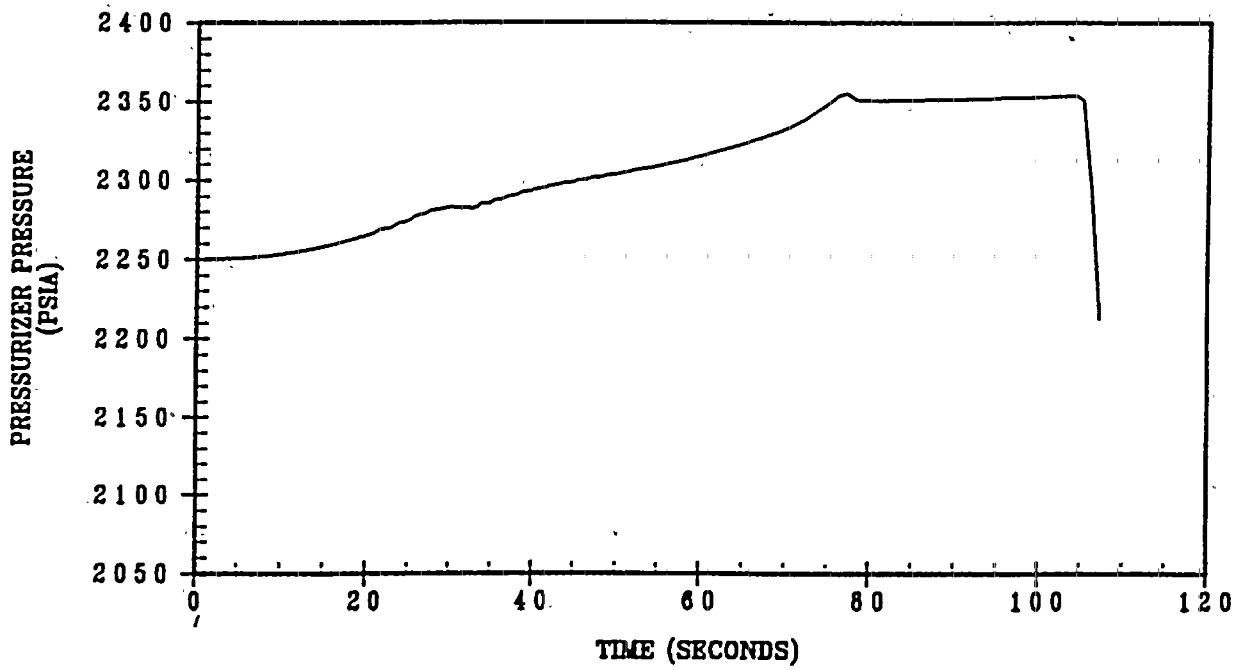


Figure 3.2.2-3 Uncontrolled RCCA Withdrawal 60% Power, Minimum Feedback (1 pcm/sec withdrawal rate)

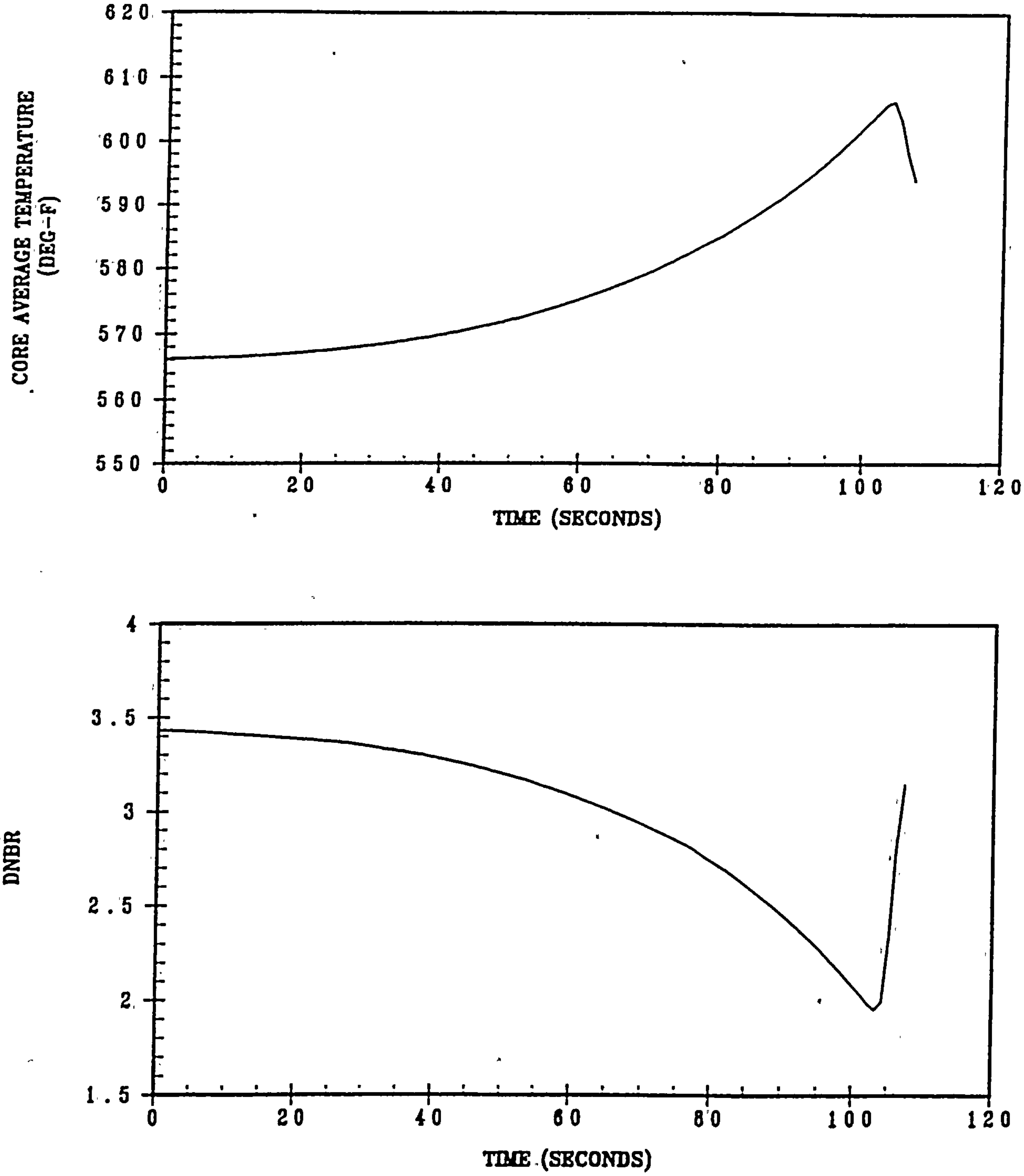
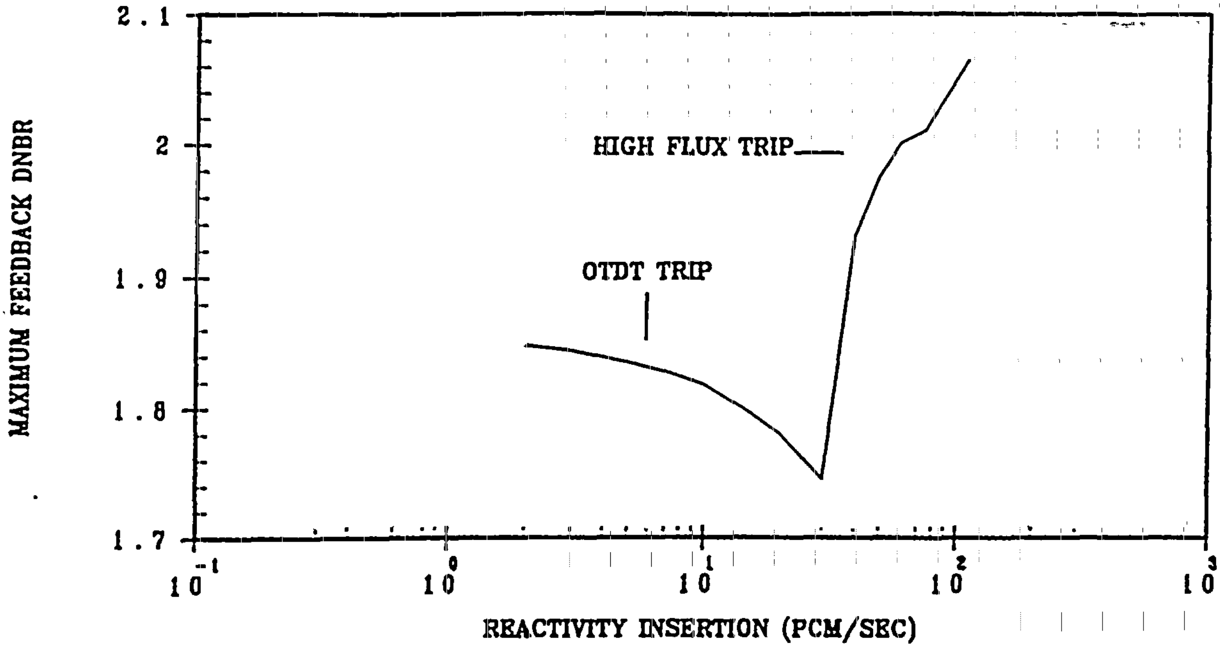


Figure 3.2.2-4 Uncontrolled RCCA Withdrawal 60% Power, Minimum Feedback
(1 pcm/sec withdrawal rate)



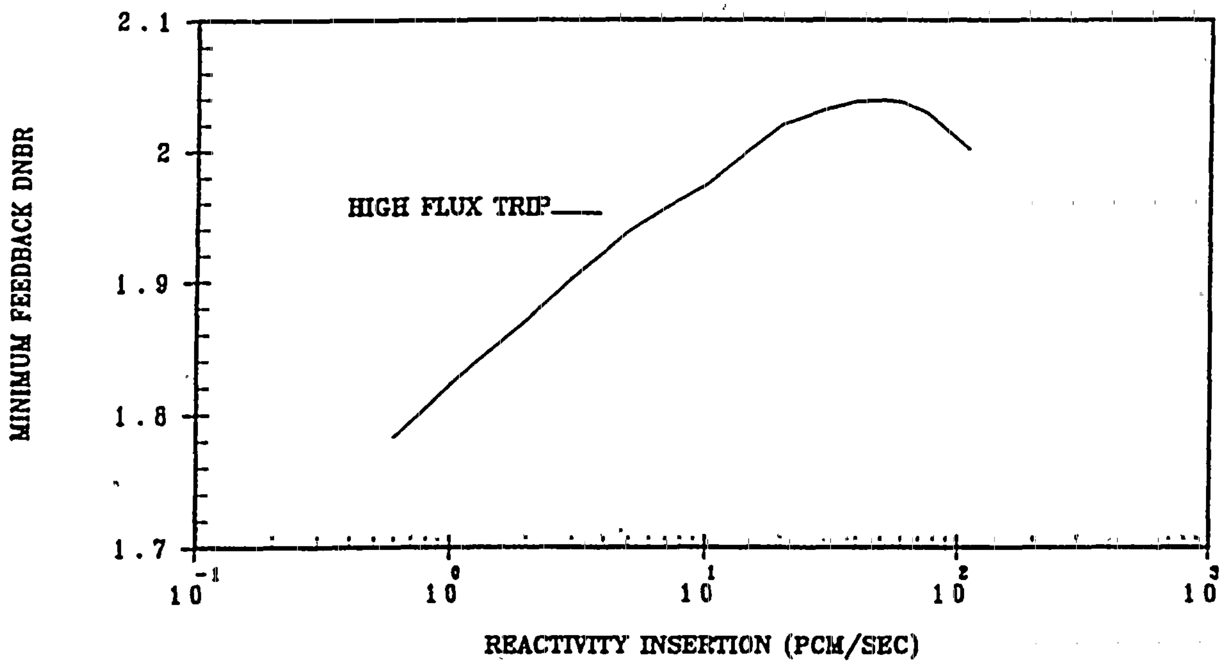


Figure 3.2.2-5 Minimum DNBR vs Reactivity Insertion Rate For Rod Withdrawal at 100% Power

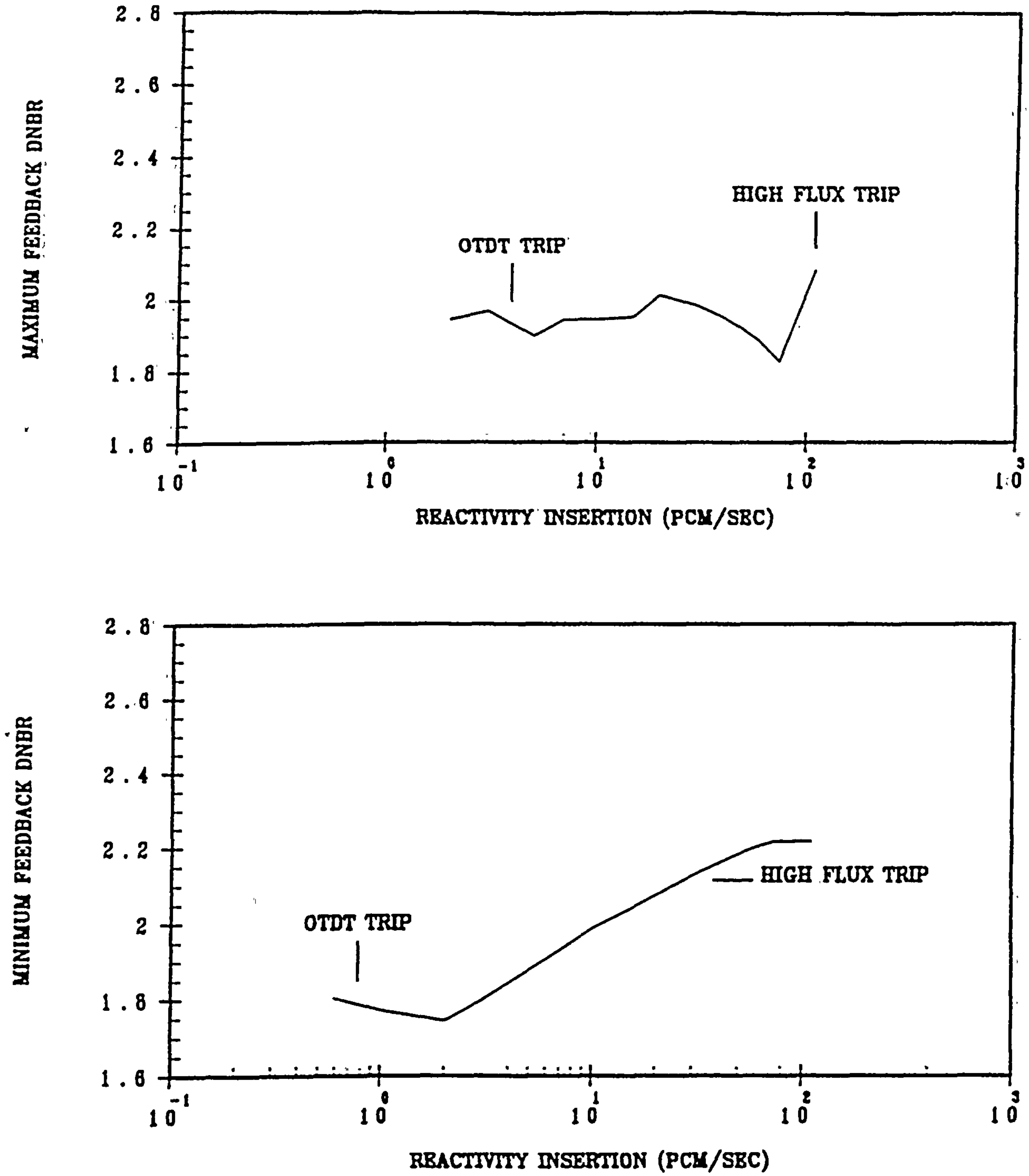
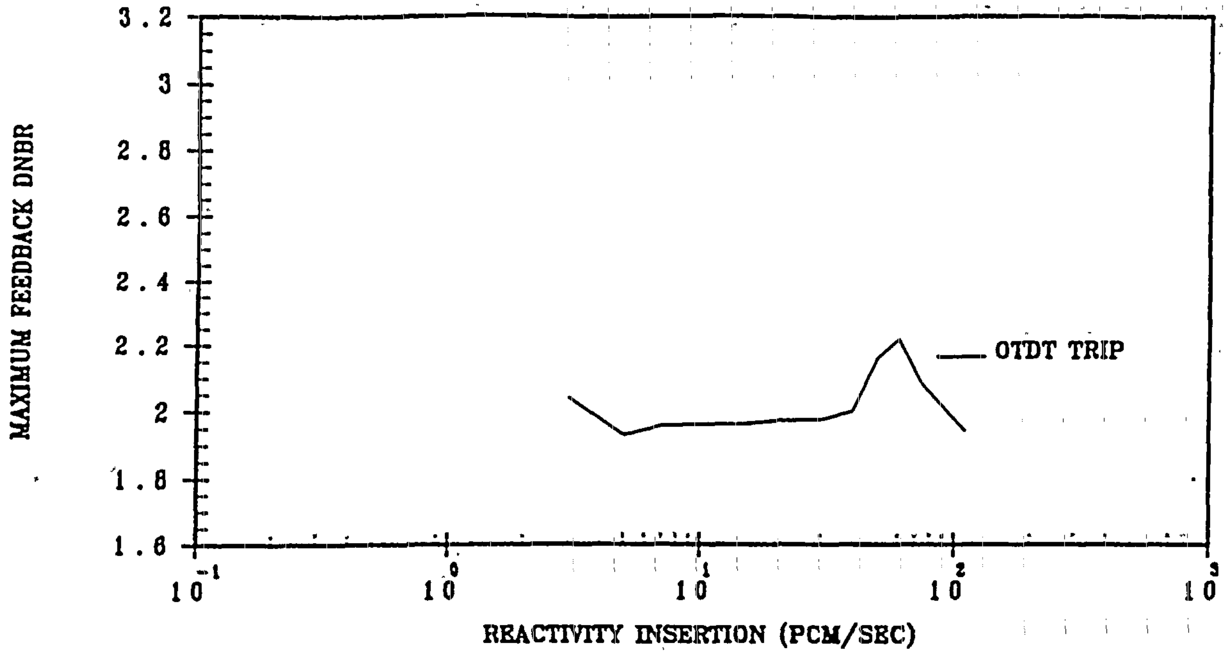


Figure 3.2.2-6 Minimum DNBR vs Reactivity Insertion Rate For Rod Withdrawal at 80% Power



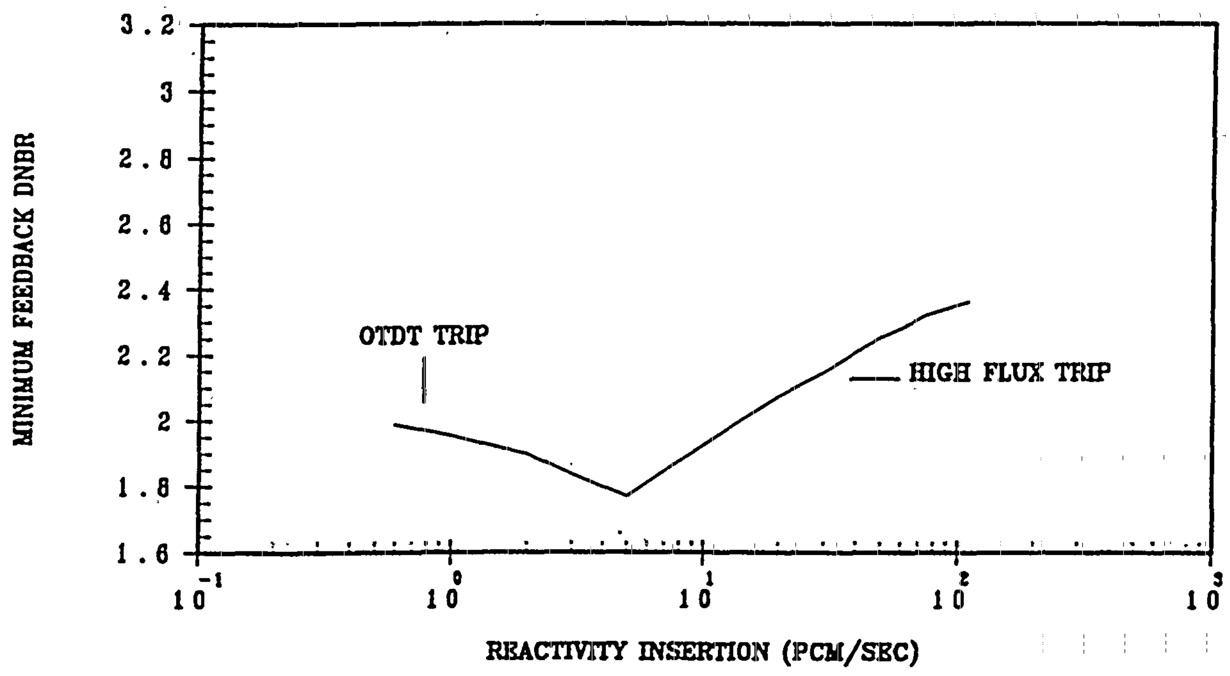
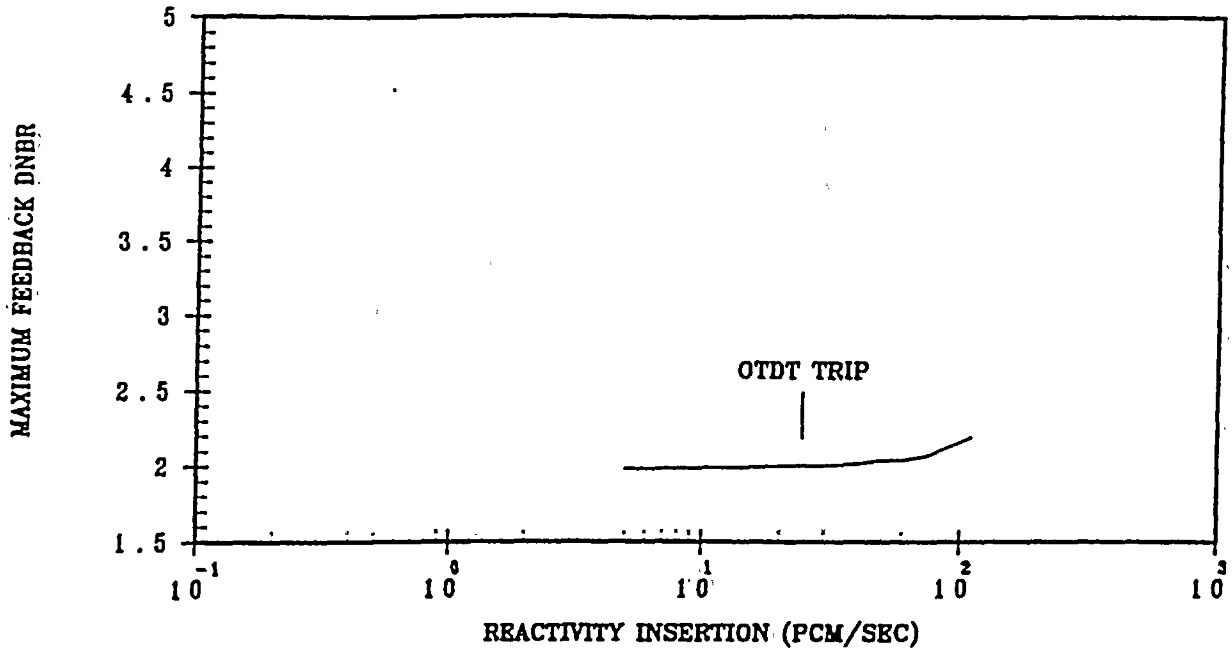


Figure 3.2.2-7 Minimum DNBR vs Reactivity Insertion Rate For Rod Withdrawal at 60% Power



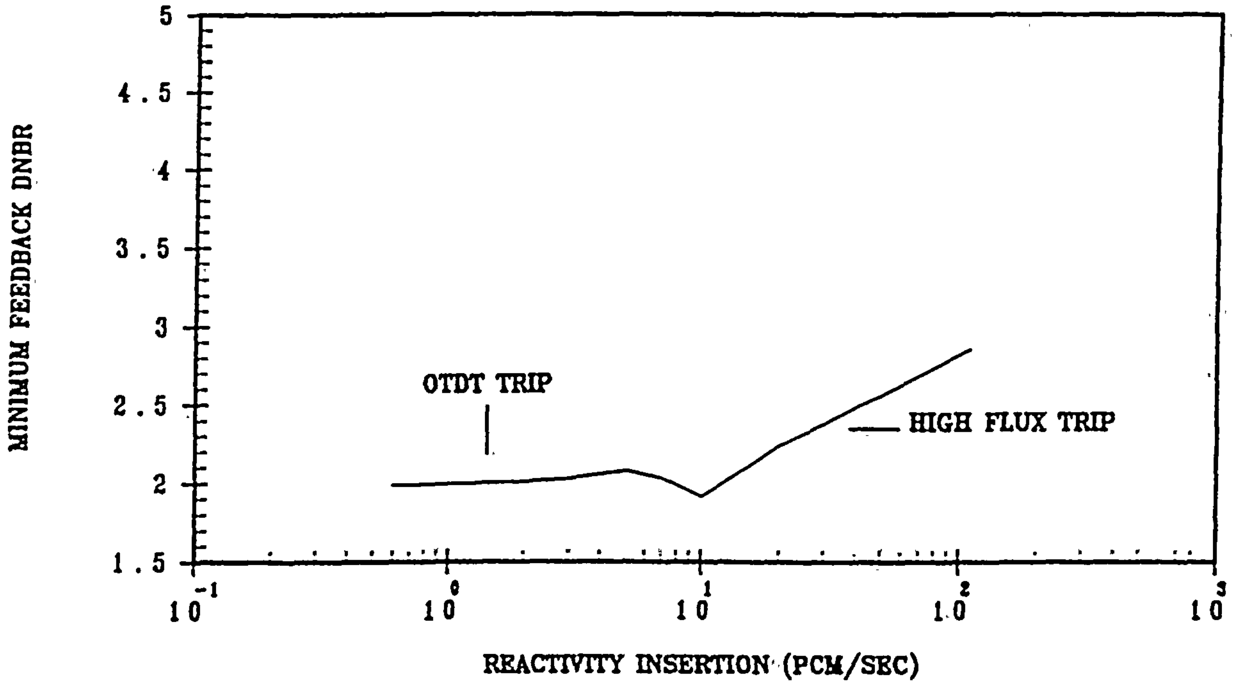


Figure 3.2.2-8 Minimum DNBR vs Reactivity Insertion Rate For Rod Withdrawal at 10% Power

3.2.3 Rod Cluster Control Assembly (RCCA) Drop

3.2.3.1 Identification of Causes and Accident Description

Dropping of a full-length RCCA is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given control bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor may occur due to the skewed power distribution representative of a dropped rod configuration. For this event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following dropped rod.

If an RCCA drops into the core during power operation, it may be detected by a rod bottom signal, an excore detector, a rod position indication, or the NIS instrumentation. The rod bottom signal device provides an indication signal for each RCCA. The other independent indication of a dropped RCCA is obtained by using the out-of-core power range channel signals. This rod drop detection circuit is actuated upon sensing a rapid decrease in local flux and is designed such that normal load variations do not cause it to be actuated.

3.2.3.2 Input Parameters and Assumptions

For a RCCA(s) Drop, the analysis assumes the following conservative assumptions.

A. This event is analyzed with the Revised Thermal Design Procedure (Reference 3). Therefore, initial reactor power, pressure, and RCS temperature are assumed at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

B. A range of moderator temperature coefficients from 0 pcm/°F to -35 pcm/°F was analyzed. An evaluation was performed to bound a +1 pcm/°F MTC at hot full power conditions.

C. A range of negative reactivity insertions from 100 pcm to 1000 pcm are assumed to simulate the Dropped RCCA event.

D. Automatic rod withdrawal is disabled at Turkey Point Units 3 and 4. Therefore, the RCCA drop event for Turkey Point is analyzed assuming manual rod control.

3.2.3.3 Description of the Analysis

The transient following a dropped RCCA event is determined by a detailed digital simulation of the plant. The dropped rod causes a step decrease in reactivity and the core power generation is determined using the LOFTRAN code (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. Since LOFTRAN employs a point neutron kinetics model, a dropped rod event is modeled as a negative reactivity insertion corresponding to the reactivity worth of the dropped rod(s) regardless of the actual configuration of the rod(s) that drop. The system transient is calculated by assuming a constant turbine load demand at the initial value (no turbine runback) and no control bank withdrawal. Because the plant is assumed to be in manual rod control (i.e., automatic rod withdrawal is disabled), the plant will establish a new equilibrium condition. The equilibrium process is monotonic in that there is no significant power overshoot without control bank withdrawal.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the dropped rod methodology described in WCAP-11394 (Reference 2).

3.2.3.4 Acceptance Criteria

Based on its frequency of occurrence, the RCCA(s) drop event is considered a Condition II event as defined by the American Nuclear Society. The primary acceptance criterion for the RCCA(s) drop event is that the critical heat flux should not be exceeded. This is demonstrated by precluding Departure from Nucleate Boiling (DNB).

3.2.3.5 Results

For the dropped RCCA event, with no automatic rod withdrawal, power may be reestablished by reactivity feedback.

Following a dropped RCCA(s) event, with no automatic rod withdrawal, the plant will establish a new equilibrium condition. Figures 3.2.3-1 and 3.2.3-2 show the transient response for representative dropped RCCA(s) case. Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 2. In all cases, the minimum DNBR remains greater than the limit value, therefore the acceptance criteria is met.

3.2.3.6 Conclusions

Following a dropped RCCA(s) event, without automatic rod withdrawal, the plant will return to a stabilized condition at less than or equal to the initial power. Results of the analysis show that a dropped RCCA event does not adversely affect the core, since the DNBR remains above the limit value for a range of dropped RCCA worths.

3.2.3.7 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984

2. Haessler, R.L., et al, "Methodology for the Analysis of the Dropped Rod Event", WCAP-11394 (Proprietary) and WCAP-11395 (Non-Proprietary), April 1987.

3. Friedland, A.J., and Ray, S., "Revised Thermal Design Procedure", WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

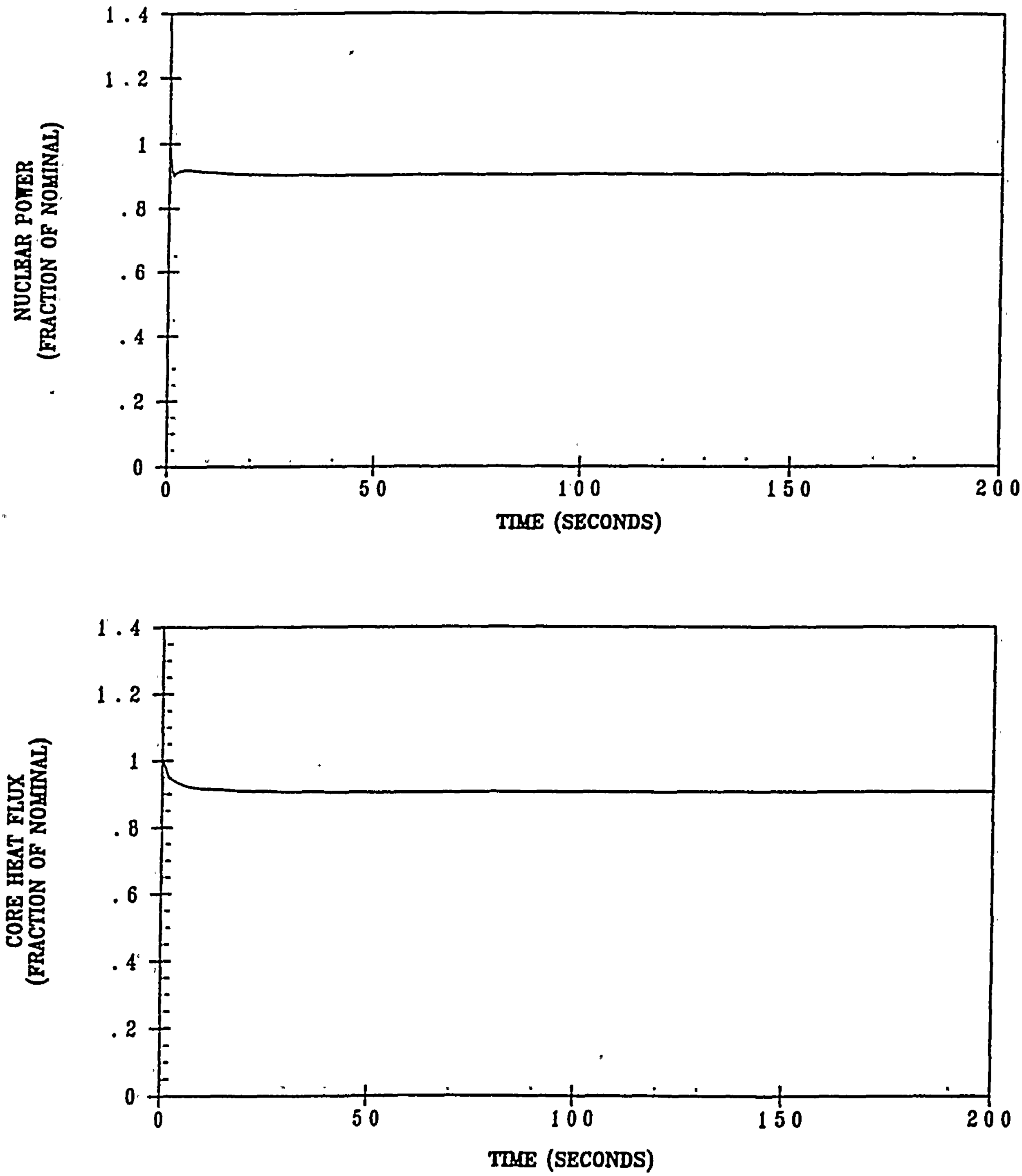
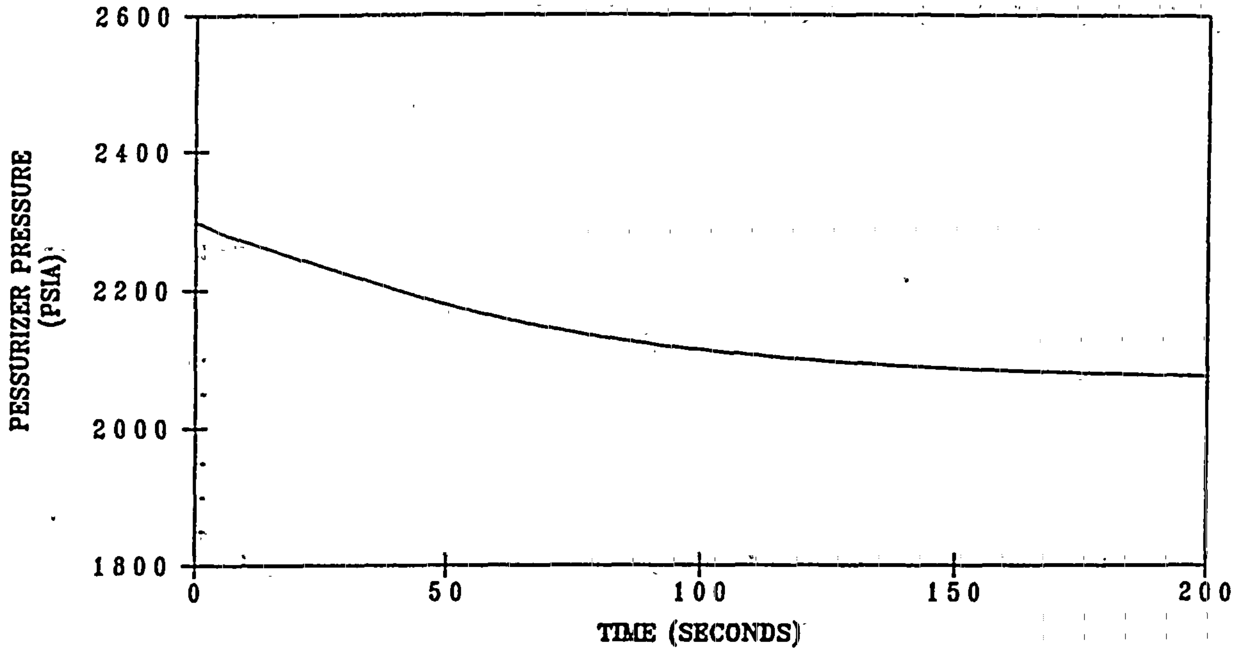


Figure 3.2.3-1 Dropped RCCA Nuclear Power and Core Heat Flux



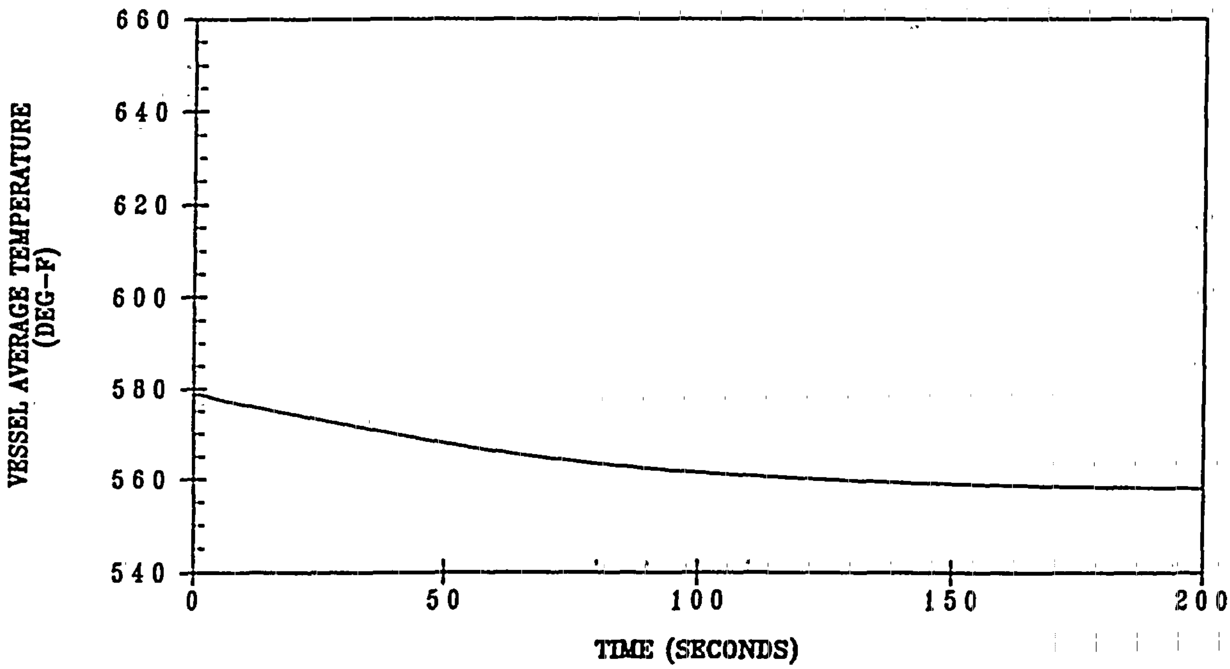


Figure 3.2.3-2 Dropped RCCA Pressurizer Pressure and Vessel Average Temperature

3.2.4 Chemical And Volume Control System (CVCS) Malfunction

3.2.4.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System (RCS) via the reactor makeup portion of the Chemical and Volume Control System (CVCS). Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to the RCS boron concentration. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of primary water makeup to the RCS from the primary water makeup system, and inadvertent dilution can be readily terminated by isolating this single source. The operation of pumps which take suction from the primary water makeup tank provides the only supply of makeup water to the RCS. In order for makeup water to be added to the RCS, the charging pumps must be running in addition to the primary water makeup pumps. The primary water makeup pumps are operating continuously.

The rate of addition of unborated water makeup to the RCS is assumed to be equal to the capacity of the three charging pumps.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required. The operator must switch from the automatic makeup mode to the dilute or alternate dilute mode, and the start switch must be placed in the start position. Omitting either step would prevent dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or makeup water flow rates deviate from preset values as a result of system malfunction.

3.2.4.2 Input Assumptions and Description of Analysis

3.2.4.2.1 Dilution During Refueling

During refueling, the following assumptions are made.

A. One residual heat removal (RHR) pump is operating to ensure continuous mixing in the reactor vessel.

B. The dilute mode adds water in the Volume Control Tank where the primary water is mixed with letdown before it is pumped back into the system. The alternate dilute mode adds water in the Volume Control Tank and to the charging pump suction header. Either mode can be assumed in the analysis.

C. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid.

D. The boron concentration in the refueling water is assumed to be 1950 ppm corresponding to a shutdown margin of at least 5% $\Delta k/k$ with all RCCAs in; periodic sampling ensures that this concentration is maintained.

A minimum RCS water volume is considered. The value assumed corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the RHR loop. A maximum dilution flow and uniform mixing are assumed.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. The high count rate alarm is actuated in the reactor containment and the control room. The count rate increase is proportional to the inverse multiplication factor.

For dilution during refueling, the boron concentration must be reduced from greater than 1950 ppm to approximately 1400 ppm before the reactor will go critical. It must be shown that there is at least 30 minutes from event initiation to when criticality is reached. Within this time, the operator must recognize the high count rate signal and isolate the primary water makeup source by closing any one of several valves and stopping the reactor makeup water pumps.

3.2.4.2.2 Dilution During Startup

In this mode, the plant is being taken from one long-term mode of operation, hot standby, to another, power. Typically, the plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions assumed for the analysis are:

A. The dilution flow is the maximum capacity of the primary water makeup pumps;

B. A minimum RCS water volume, corresponding to the active RCS volume minus the pressurizer;

C. The Mode 2 initial boron concentration is assumed to be 2000 ppm which is a conservative maximum value for the conditions of hot zero power, rods at the insertion limits and no xenon. The minimum change in boron concentration following a reactor trip, 200 ppm, results in the maximum critical concentration for the conditions of hot zero power, all rods inserted except the most-reactive RCCA, and no xenon. The critical concentration at hot-zero-power conditions is thus 1800 ppm.

The startup mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to achieve criticality. During this mode, the rods are in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods. The operator determines the estimated critical position of the control rods prior to approaching criticality, thus ensuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range (nominally at 10^5 cps). Too fast of a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip, and the reactor would immediately shut down.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power until the power range high neutron flux low setpoint is reached and a reactor trip occurs. From the initiation of the event, there is greater than 15 minutes available for operator action prior to return to criticality.

3.2.4.2.3 Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions assumed for this analysis are the following.

A. With the units at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging pumps. Although less charging pumps are normally in operation, the analysis is performed assuming the dilution flow is the maximum capacity of the charging pumps.

B. A minimum RCS water volume, corresponding to the active RCS volume minus the pressurizer, is assumed.

C. The Mode 1 initial boron concentration is assumed to be 1900 ppm which is a conservative maximum value for the conditions of hot full power, rods at the insertion limits and no xenon. The minimum change in boron concentration following a reactor trip, 350 ppm, results in the maximum critical concentration for the conditions of hot zero power, all rods inserted except the most-reactive RCCA, and no xenon. The critical concentration at hot-zero-power conditions is thus 1550 ppm.

With the reactor in automatic rod control the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in available shutdown margin. The rod insertion limit alarms (Low and Low-Low settings) alert the operator to the dilution. This is sufficient time to determine the cause of dilution, isolate the reactor makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be 3.1 pcm/sec, which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis (Section 3.2.2 of this report). Following reactor trip, there is greater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shut down margin is lost.

3.2.4.3 Acceptance Criteria

A CVCS malfunction is classified as an ANS Condition II event, a fault of moderate frequency. Criteria established for Condition II events are as follows.

- The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures.

- Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value which would cause fuel centerline melt.

This event is analyzed to ensure that there is sufficient time for mitigation of an inadvertent boron dilution prior to the complete loss of shutdown margin. A complete loss of plant shutdown margin results in a return of the core to the critical condition causing an increase in the RCS temperature and heat flux. This could violate the safety analysis limit DNBR value and challenge the fuel and fuel cladding integrity. A complete loss of plant shutdown margin could also result in a return of the core to the critical condition causing an increase in RCS pressure. This could challenge the pressure design limit for the reactor coolant system.

If the minimum allowable shutdown margin is shown not to be lost, the condition of the plant at any point in the transient is within the bounds of those calculated for other Condition II transients. By showing that the above criteria are met for those Condition II events, it can be concluded that they are also met for the boron dilution event.

To preclude a complete loss of plant shutdown margin, operator action is relied upon. The analysis of the boron dilution event is only performed, in accordance with Regulatory Guide 1.70 Rev. 1, in Modes 1, 2, and 6 (plant modes of full-power operation, plant startup, and refueling, respectively). The required operator action times are:

Mode 1: 15 minutes from time of alarm
Mode 2: 15 minutes from time of dilution
Mode 6: 30 minutes from time of dilution

3.2.4.4 Results

Plant operation during refueling, startup, and power operation is considered in this analysis. Table 3.2.4-1 contains the time sequence of events of the boron dilution analysis for refueling, startup and power operation. Table 3.2.4-2 presents results of the boron dilution analysis for refueling, startup, and power operation. Also included in this table are pertinent analysis assumptions. Perfect mixing is assumed in the analysis. This assumption results in a conservative rate of RCS boron dilution.

3.2.4.5 Conclusions

If an unintentional dilution of boron in the reactor coolant system does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator sufficient time to determine the cause of the addition and take corrective action before shutdown margin is lost. The acceptance criteria as specified in Section 3.2.4.3 are met.

Table 3.2.4-1

Sequence of Events - Uncontrolled Boron Dilution

Mode of Operation	Event	Time (seconds)
During Refueling	Dilution begins	0
	Shutdown margin lost (if dilution continues)	>1800.0
During Startup	Power range - low setpoint reactor trip due to dilution	0
	Shutdown margin lost (if dilution continues)	>900
During Full-Power Operation		
a. Automatic Rod Control	Operator receives low-low rod insertion limit alarm due to dilution	0
	Shutdown margin lost (if dilution continues)	>900
b. Manual Rod Control	Reactor trip on OTΔT due to dilution	0
	Shutdown margin is lost (if dilution continues)	>900

Table 3.2.4-2

Summary of Boron Dilution Analysis Results and Analysis Assumptions

Mode of Operation	Dilution Flow Rate (gpm)	Active Volume (cubic feet)	Assumed Initial Boron Conc. (ppm)	Assumed Critical Boron Conc. (ppm)	Average Core Coolant Temperature (°F)	Operation Action time (minutes)
Power Operation						
Auto Rod Control	252	7308.2	1900	1550	583.2	31.5
Manual Rod Control	252	7308.2	1900	1550	583.2	30.3
Startup	252	7308.2	2000	1800	554.5	17.0
Refueling	252	3204.6	1950	1400	140.0	31.0

3.2.5 Startup of an Inactive Reactor Coolant Loop

The current Turkey Point Technical Specifications preclude operation with an inactive loop. This event was originally included in the UFSAR licensing basis when operation with a loop out of service was considered. Based on the current Technical Specifications which prohibit at power operation with a loop out of service as indicated above, it is concluded that this event should be deleted from the current UFSAR licensing basis.

3.2.6 Excessive Heat Removal Due To Feedwater System Malfunctions

3.2.6.1 Identification of Causes and Accident Description

Reductions in feedwater temperature or excessive feedwater additions are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the RCS and the secondary side of the plant. The overpower/overtemperature protection functions (neutron high flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the limit value.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. Continuous excessive feedwater addition is prevented by the steam generator high-high water level trip.

A second example of excess heat removal is the transient associated with the accidental opening of the low-pressure heater bypass valve which diverts flow around the low-pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost; e.g., following a large load decrease. At power, this increased subcooling will create a greater load demand on the RCS.

3.2.6.2 Input Parameters and Assumptions

The reactivity insertion rate following a feedwater system malfunction, attributed to the cooldown of the RCS, is calculated with the following assumptions.

A. This accident is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A (Reference 1). Therefore, the initial reactor power, pressure, and RCS average temperature are assumed to be at the nominal values. Uncertainties in initial conditions are included in the DNBR limit calculated using the methodology described in Reference 1.

B. For the feedwater control valve accident at full-power conditions, one feedwater control valve is assumed to malfunction resulting in a step increase to 200% of nominal feedwater flow to one steam generator.

C. The initial water level in all the steam generators is a conservatively low level.

D. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

E. The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal that closes all feedwater main control and feedwater control-bypass valves, indirectly closes all feedwater pump discharge valves, and trips the main feedwater pumps and turbine generator.

The reactor protection systems, including Power-Range High Neutron Flux, Overpower ΔT, and Turbine Trip on High-High Steam Generator Water Level features are available to provide mitigation of the feedwater system malfunction transient.

Normal reactor control systems and engineered safety systems (e.g., SI) are not assumed to function. The reactor protection system may actuate to trip the reactor due to an overpower condition. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

3.2.6.3 Description of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN (Reference 2) computer code. This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate acceptable consequences in the event of a feedwater system malfunction. Feedwater temperature reduction due to low-pressure heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump is considered. Additionally, excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully is considered.

The excessive feedwater flow event assumes an accidental opening of one feedwater control valve with the reactor at full-power conditions with both automatic and manual rod control. Both the automatic and manual rod control cases assume a conservatively large moderator density coefficient characteristic of EOL conditions.

The plant conditions representative of zero-load operation are not affected by the power uprating at Turkey Point. Therefore, the analysis of the Feedwater Malfunction event with the reactor just critical at zero-load conditions was not performed in support of the plant change to the uprated power level. The results and conclusions presented in Section 14.1.7 of the UFSAR remain valid for the zero-load excessive feedwater addition transient.

3.2.6.4 Acceptance Criteria

Based on its frequency of occurrence, the feedwater system malfunction event is considered a Condition II event as defined by the American Nuclear Society. Even though DNB is the primary concern in the analysis of the Feedwater Malfunction event, the following 3 items summarize the criteria associated with this transient.

- The critical heat flux shall not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

- Pressure in the reactor coolant and main steam systems shall be maintained below 110% of the design pressures.

- Fuel temperature and fuel clad strain limits shall not be exceeded. The peak linear heat generation rate should not exceed a value which would cause fuel centerline melt.

3.2.6.5 Results

Opening of a low-pressure heater bypass valve and trip of the heater drain pumps causes a reduction in the feedwater temperature which increases the thermal load on the primary system. The reduction in the feedwater temperature is less than 60°F, resulting in an increase in the heat load on the primary system of less than 10 percent of full power. The increased thermal load due to the opening of the low-pressure heater bypass valve would result in a transient very similar (but of reduced magnitude) to the Excessive Load Increase incident presented in Section 3.2.7. Thus, the results of this event are bounded by the Excessive Load Increase event and, therefore, not presented here.

The full-power case (EOL maximum reactivity feedback with automatic rod control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in the affected steam generator reaches the high-high level setpoint. Assuming the reactor to be in manual rod control results in a slightly less-severe transient. The rod control system is not required to function for this event; however, assuming that the rod control system is operable yields a slightly more limiting transient.

For all cases of excessive feedwater flow, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. In addition, the feedwater discharge isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Following turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in Section 3.2.9 (Loss of External Electrical Load and/or Turbine Trip). If the reactor was in automatic rod control, the control rods would be inserted at the maximum rate following the turbine trip, and the resulting transient would not be limiting in terms of peak RCS pressure.

Transient results (see Figures 3.2.6-1 through 3.2.6-3) show the core heat flux, pressurizer pressure, core average temperature, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator water level rises until the feedwater addition is terminated as a result of the high-high steam generator water level trip. The DNBR does not drop below the limit value at any time.

Since the power level rises during this event, the fuel temperature will also rise until the reactor trip occurs. The core heat flux lags behind the neutron flux due to the fuel rod thermal time constant and, as a result, the peak core heat flux value does not exceed 118% of nominal. Thus, the peak fuel melting temperature will remain well below the fuel melting point.

The calculated sequence of events is shown in Table 3.2.6-1. The transient results show that the DNBR does not fall below the limit value at any time during the feedwater flow increase transient; thus, the ability of the primary coolant to remove heat from the fuel rods is not reduced. Therefore, the fuel cladding temperature does not rise significantly above its initial value during the transient.

3.2.6.6 Conclusions

The decrease in feedwater temperature transient due to an opening of the low-pressure heater bypass valve is less severe than the excessive load increase event (see Section 3.2.7). Based on the results presented in Section 3.2.7, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

For the excessive feedwater addition at power transient, the results show that the DNBRs encountered are above the limit value; hence, no fuel damage is predicted.

The protection features presented in Section 3.2.6.2 provide mitigation of the feedwater system malfunction transient such that the above criteria are satisfied.

As documented in Section 14.1.7 of the UFSAR, the analysis at hot zero power demonstrated that the minimum DNBR remained greater than the limit value for a maximum reactivity insertion rate corresponding to an excessive feedwater addition at no-load conditions. This conclusion is unaffected by the uprated power conditions.

3.2.6.7 References

1. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-proprietary), April 1989

2. Burnett, T. W. T. et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-proprietary)", April 1984

Table 3.2.6-1
Time Sequence of Events
Excessive Feedwater Flow at Full Power (Automatic Rod Control)

Event	Time (sec)
One main feedwater control valve fails fully open	0.0
High-high SG water level signal generated	35.0
Minimum DNBR occurs	37.0
Turbine trip occurs due to high-high SG water level	37.5
Reactor trip due to turbine trip (rod motion begins)	39.5
Feedwater control valves fully closed	44.0

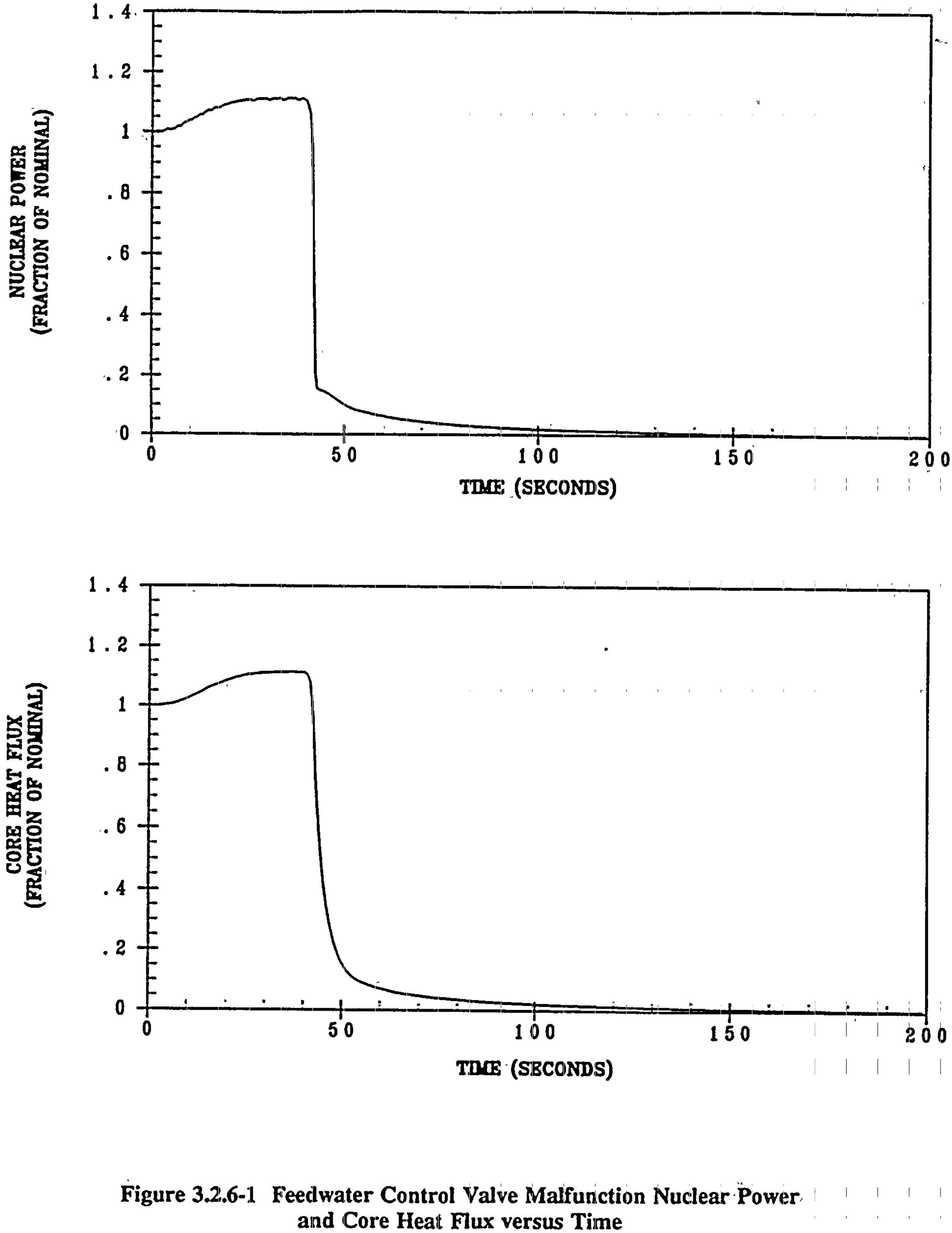
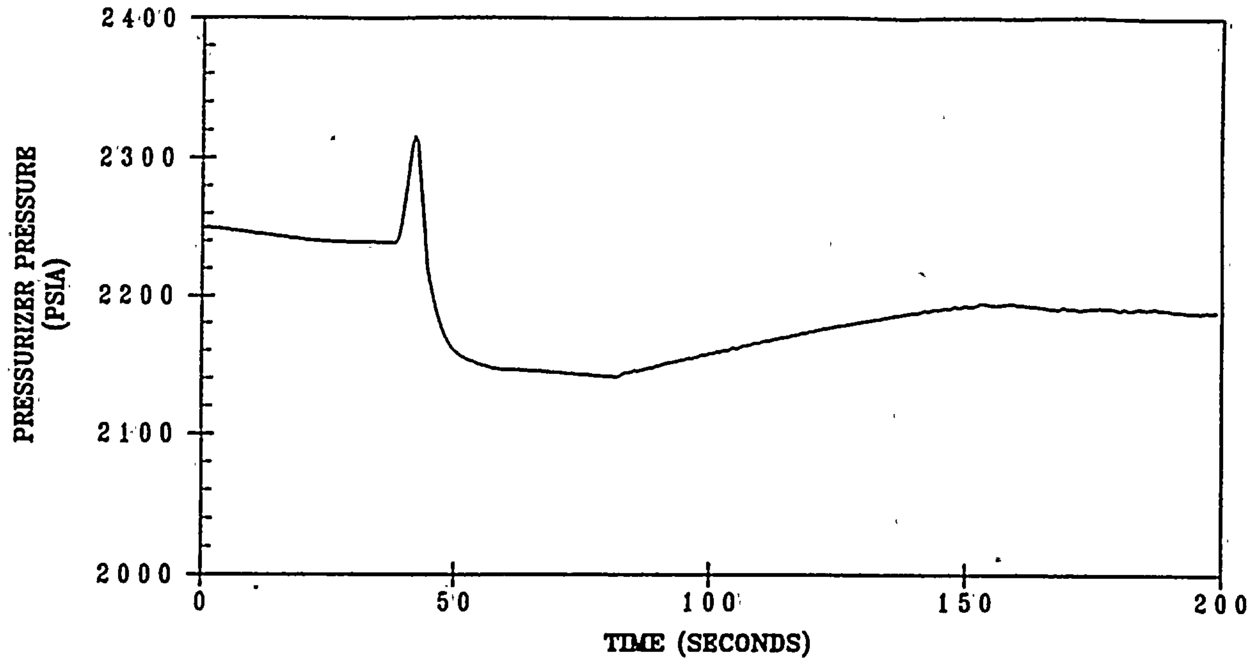


Figure 3.2.6-1 Feedwater Control Valve Malfunction Nuclear Power and Core Heat Flux versus Time



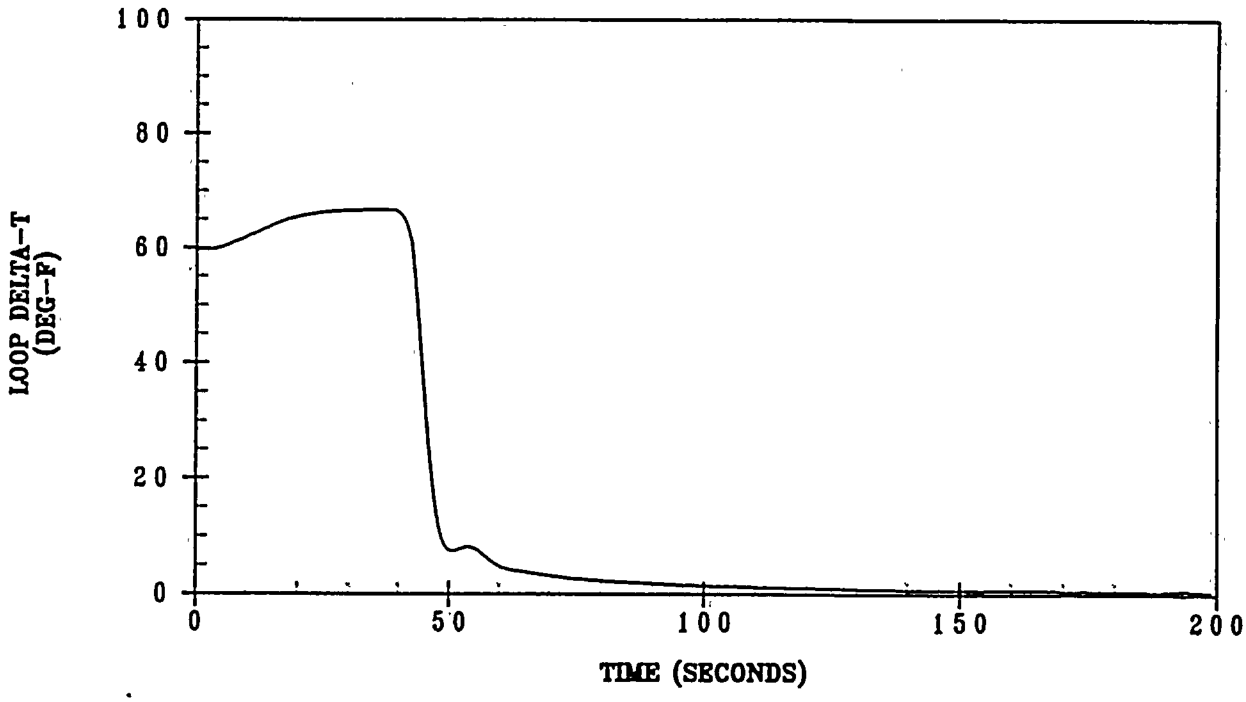


Figure 3.2.6-2 Feedwater Control Valve Malfunction Pressurizer Pressure and Loop Delta-T versus Time

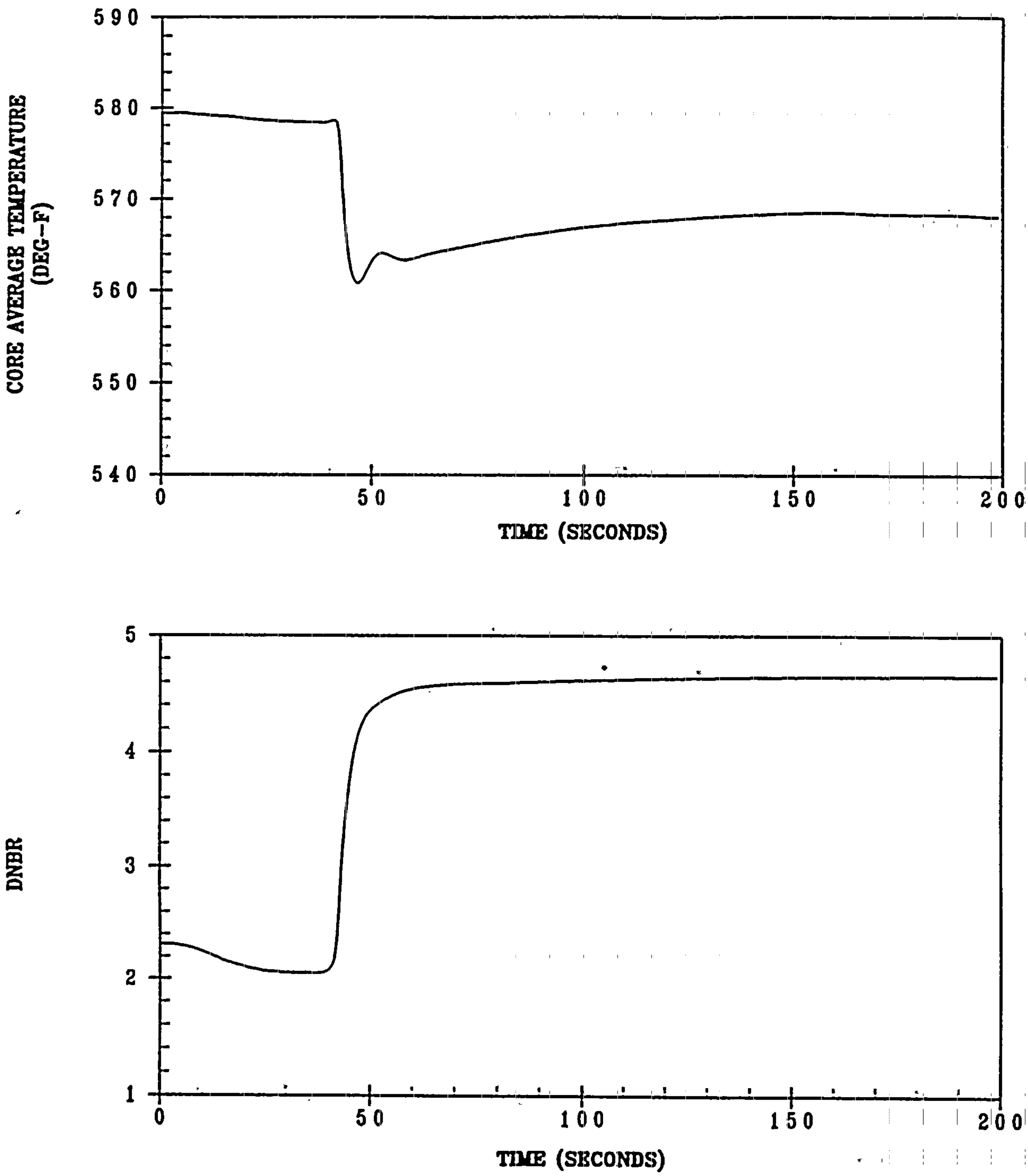


Figure 3.2.6-3 Feedwater Control Valve Malfunction Core Average Temperature and DNBR versus Time

3.2.7 Excessive Load Increase Incident

3.2.7.1 Identification of Cause and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp-load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient would be terminated in sufficient time to prevent the DNB design basis from being violated.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control or turbine speed control.

During power operation, steam dump to the condenser is controlled by comparing the RCS temperature to a reference temperature based on turbine power, where a high temperature difference in conjunction with a loss of load or turbine trip indicates a need for steam dump. A single controller malfunction does not cause steam dump valves to open. Interlocks are provided to block the opening of the valves unless a large turbine load decrease or a turbine trip has occurred. In addition, the reference temperature and loss of load signals are developed by independent sensors.

Regardless of the rate of load increase, the reactor protection system will trip the reactor in time to prevent the DNBR from going below the limit value. Increases in steam load to more than design flow are analyzed as the steam line rupture event in Section 3.2.16.

Protection against an excessive load increase accident is provided by the following reactor protection system signals.

- Overtemperature ΔT
- Overpower ΔT
- Power range high neutron flux
- Low pressurizer pressure

3.2.7.2 Input Parameters and Assumptions

- This accident is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A (Reference 1). Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit as described in Reference 1.

- The evaluation is performed for a step load increase of 10 percent steam flow from 100 percent of Rated Thermal Power.

- This event is analyzed in both automatic and manual rod control.

- The excessive load increase event is analyzed for both the beginning-of-life (minimum reactivity feedback) and end-of-life (maximum reactivity feedback) conditions. A small (zero) moderator density coefficient at beginning of life and a large value at end of life are used. A positive moderator temperature coefficient is not assumed since this would provide a transient benefit. For all cases, a small (absolute value) Doppler coefficient of reactivity is assumed.

3.2.7.3 Description of Analysis

Four cases are analyzed to demonstrate the plant behavior following a 10% step-load increase from rated load. These cases are as follows.

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback
- Reactor in automatic rod control with BOL (minimum moderator) reactivity feedback
- Reactor in automatic rod control with EOL (maximum moderator) reactivity feedback

This accident is analyzed using the LOFTRAN (Reference 2) computer code to determine the plant transient conditions following the excessive load increase. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system. The code computes pertinent plant variables including DNBR, temperatures, pressures, and power level.

At BOL, minimum moderator feedback cases, the core has the least-negative moderator temperature coefficient of reactivity and the least-negative Doppler only power coefficient curve; therefore, the least-inherent transient response capability. Since a positive moderator temperature coefficient would provide a transient benefit, a zero moderator temperature coefficient was assumed in the minimum feedback cases. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its most-negative value and the most-negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature. Normal reactor control systems and engineered safety systems are not required to function. A 10% step increase in steam demand is assumed and the analysis does not take credit for the operation of the pressurizer heaters. The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for the cases analyzed. No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

3.2.7.4 Acceptance Criteria

Based on its frequency of occurrence, the excessive load increase accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event.

The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. With respect to peak pressure, the excessive load increase accident is bounded by the loss of electrical load/turbine trip analysis. The loss of electrical load/turbine trip analysis is described in Section 3.2.9.

Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate (expressed in kw/ft) should not exceed a value which would cause fuel centerline melt.

The protection features presented in Section 3.2.7.1 provide mitigation of the excessive load increase transient such that the above criteria are satisfied.

3.2.7.5 Results

Figures 3.2.7-1 through 3.2.7-4 illustrate the transient with the reactor in the manual rod control mode. As expected, for the BOL case, there is a slight power increase and the average core temperature shows a decrease. This results in a DNBR which increases (after a slight decrease) above its initial value. For the EOL manual rod control case, there is a larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the safety analysis limit value.

Figures 3.2.7-5 through 3.2.7-8 illustrate the transient assuming the reactor is in the automatic rod control mode. Both the BOL and EOL cases show that core power increases. The BOL case shows the core average temperature to stabilize, due to the action of the rod control system, at a slightly higher value from the initial temperature. The EOL case shows that after a slight increase the core average temperature stabilizes, again due to the action of the rod control system, at a value approximately equal to the initial temperature. For both of these cases the DNBR remains above the safety analysis limit value.

The calculated time sequence of events for the excessive load increase incident is shown on Table 3.2.7-1. Note that a reactor trip signal was not generated for any of the four cases.

3.2.7.6 Conclusions

It has been demonstrated that for an excessive load increase, the minimum DNBR during the transient will not go below the safety analysis limit value thus ensuring the applicable acceptance criteria for critical heat flux and fuel centerline melt are met. Following the initial load increase, the plant reaches a stabilized condition. In addition, RCS pressure and main steam system does not exceed 110% of design as described in Section 3.2.9.

3.2.7.7 References

1. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure", WCAP-11397-P-A, (Proprietary), WCAP-11397-A (Nonproprietary), April 1989.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

Table 3.2.7-1

Time Sequence of Events for
Excessive Load Increase Incident

Accident	Event	Time (sec)
Manual Rod Control (minimum moderator feedback)	10% step-load increase	0.0
	Equilibrium conditions reached (approx. time)	170.0
Manual Rod Control (maximum moderator feedback)	10% step-load increase	0.0
	Equilibrium conditions reached (approx. time)	90.0
Automatic Rod Control (minimum moderator feedback)	10% step-load increase	0.0
	Equilibrium conditions reached (approx. time)	140.0
Automatic Rod Control (maximum moderator feedback)	10% step-load increase	0.0
	Equilibrium conditions reached (approx. time)	40.0

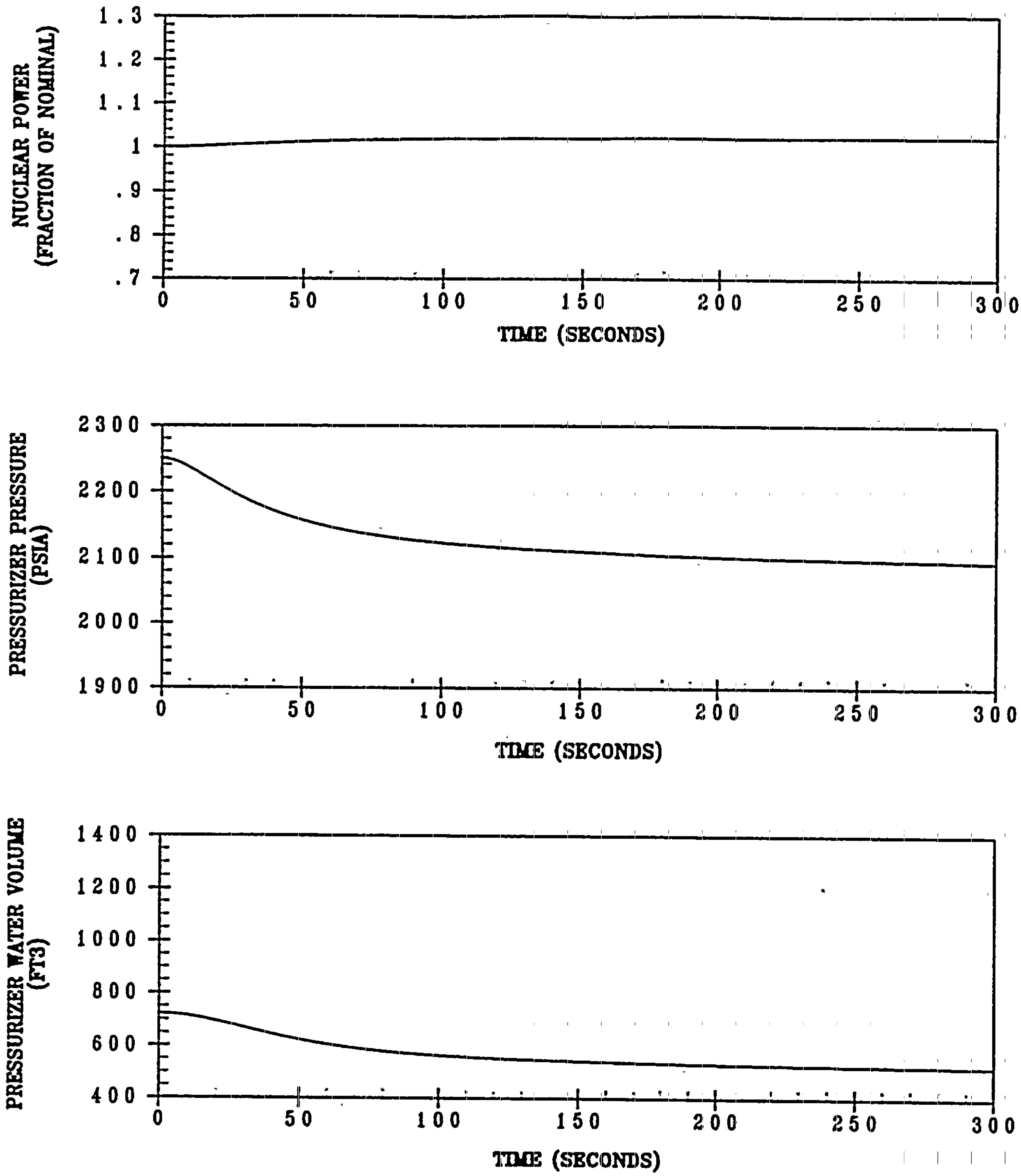
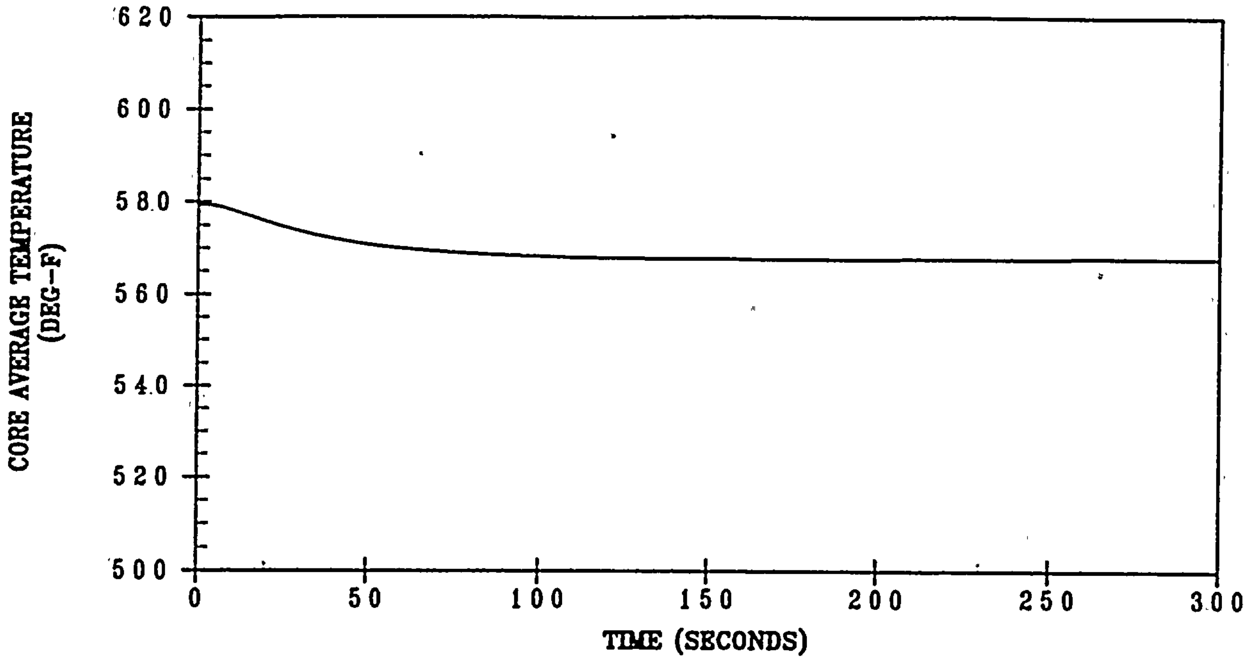


Figure 3.2.7-1 10% Step Load Increase Minimum Moderator Feedback, Manual Rod Control



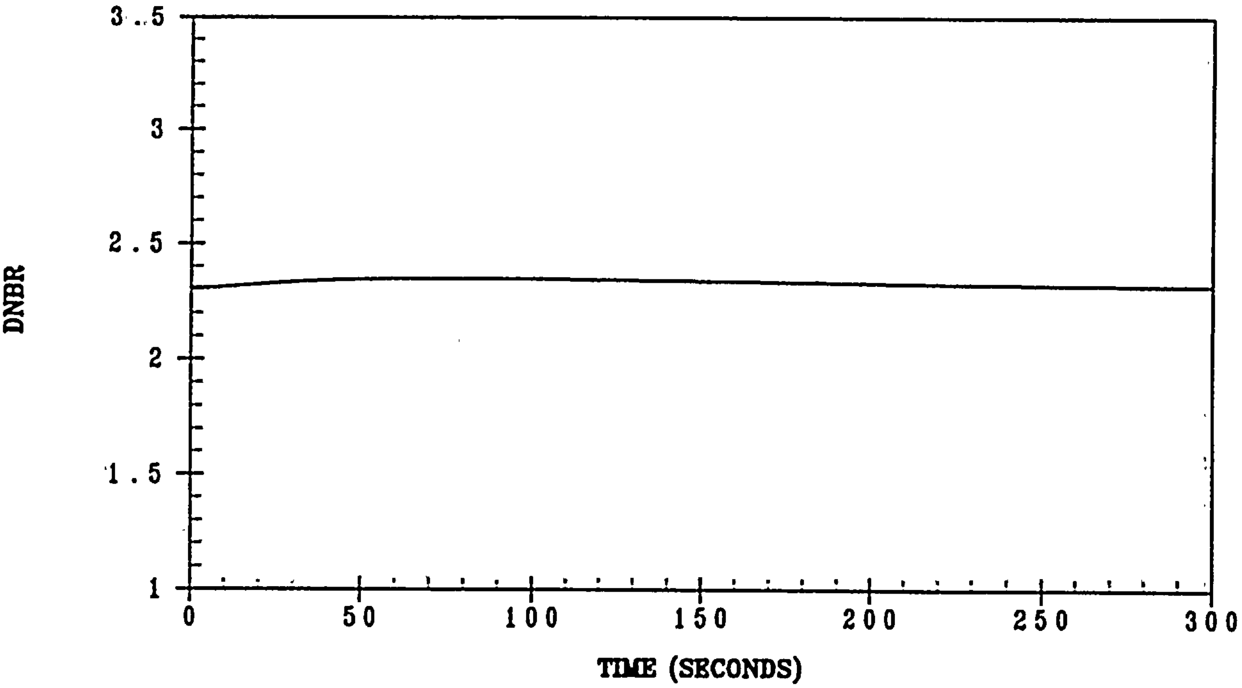


Figure 3.2.7-2 10% Step Load Increase Minimum Moderator Feedback
Manual Rod Control

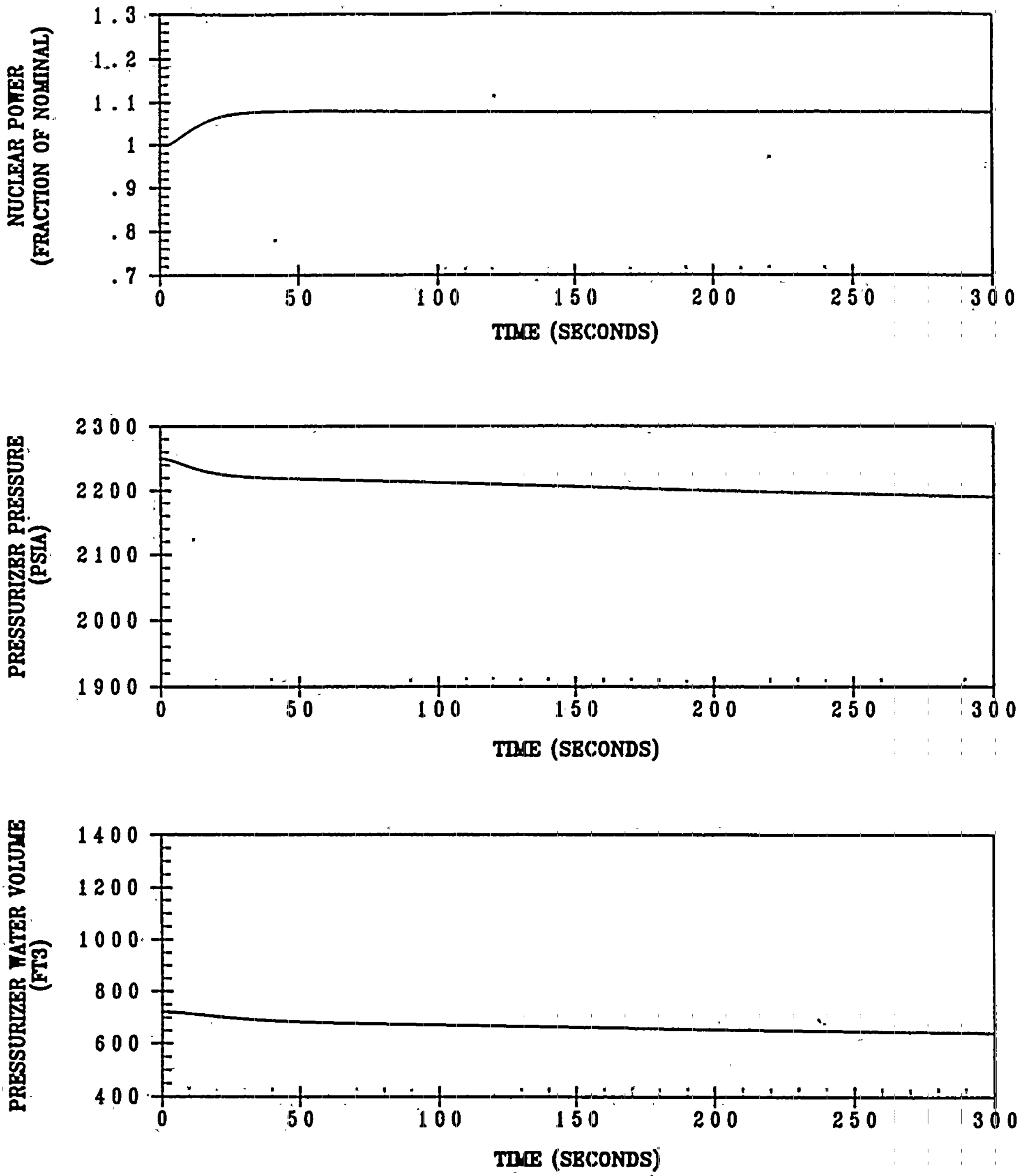
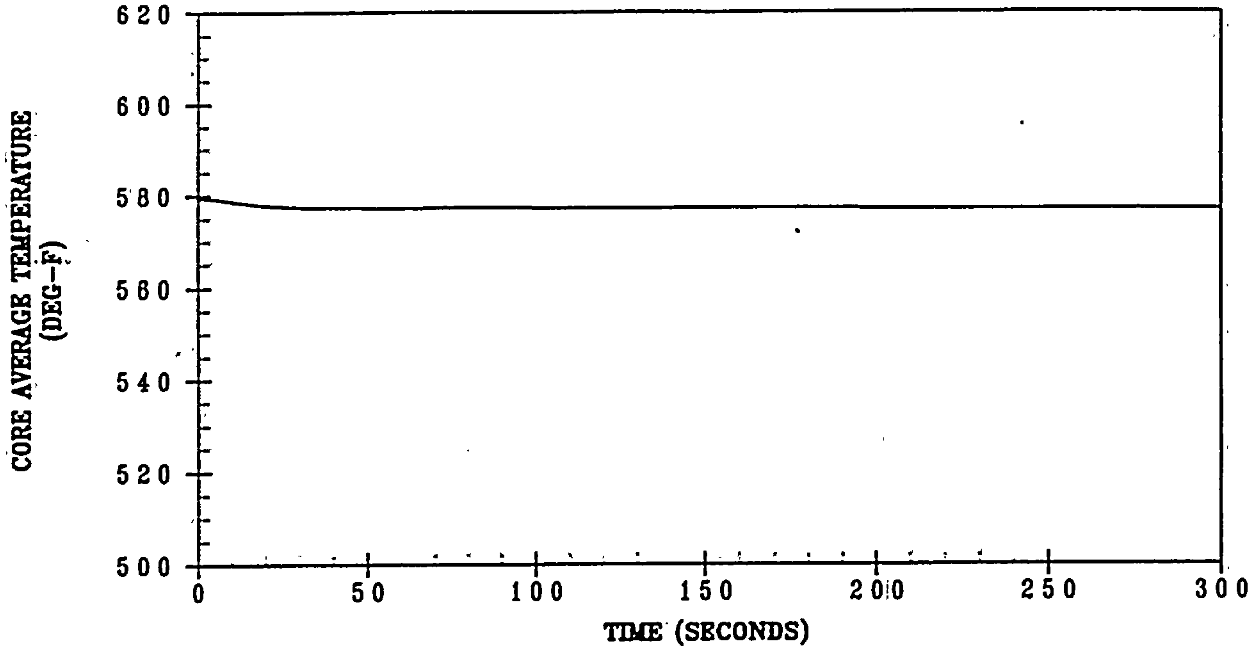


Figure 3.2.7-3 10% Step Load Increase Maximum Moderator Feedback Manual Rod Control



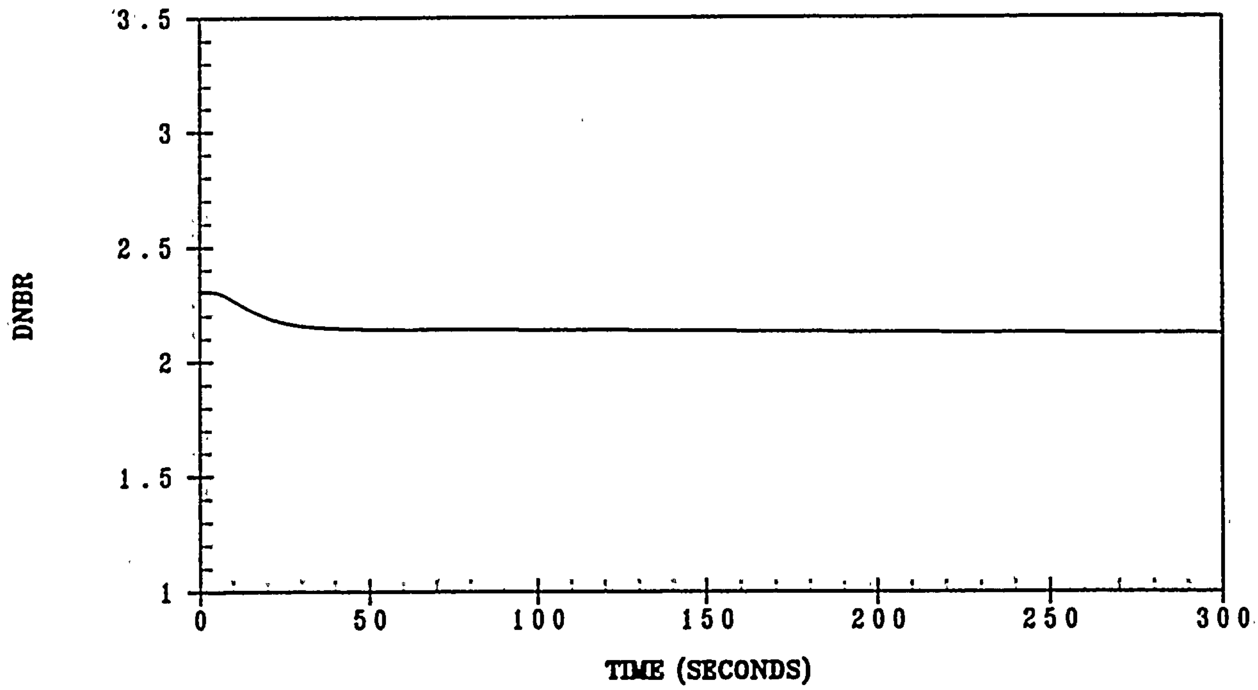


Figure 3.2.7-4 10% Step Load Increase Maximum Moderator Feedback Manual Rod Control

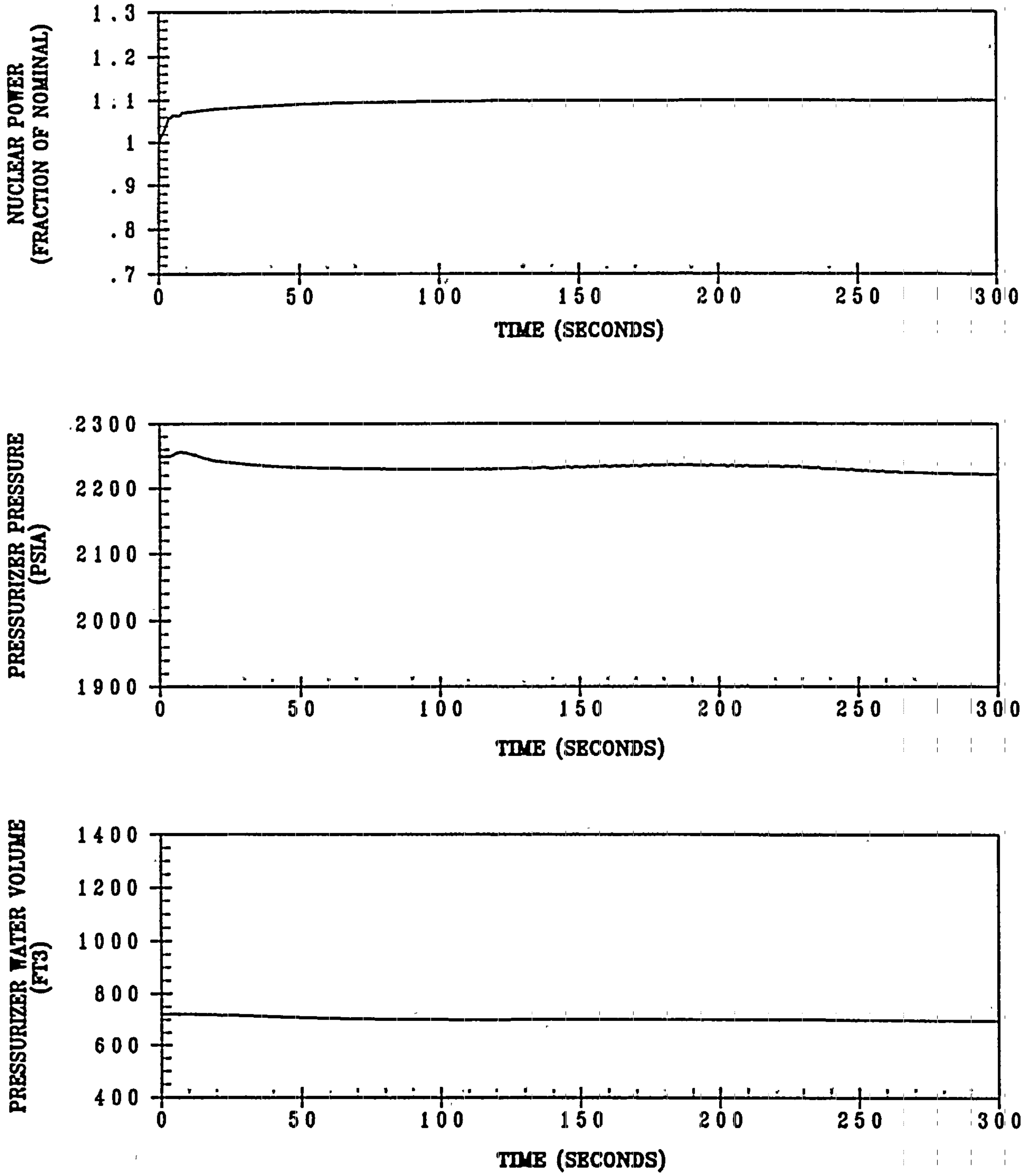


Figure 3.2.7-5 10% Step Load Increase Minimum Moderator Feedback Automatic Rod Control

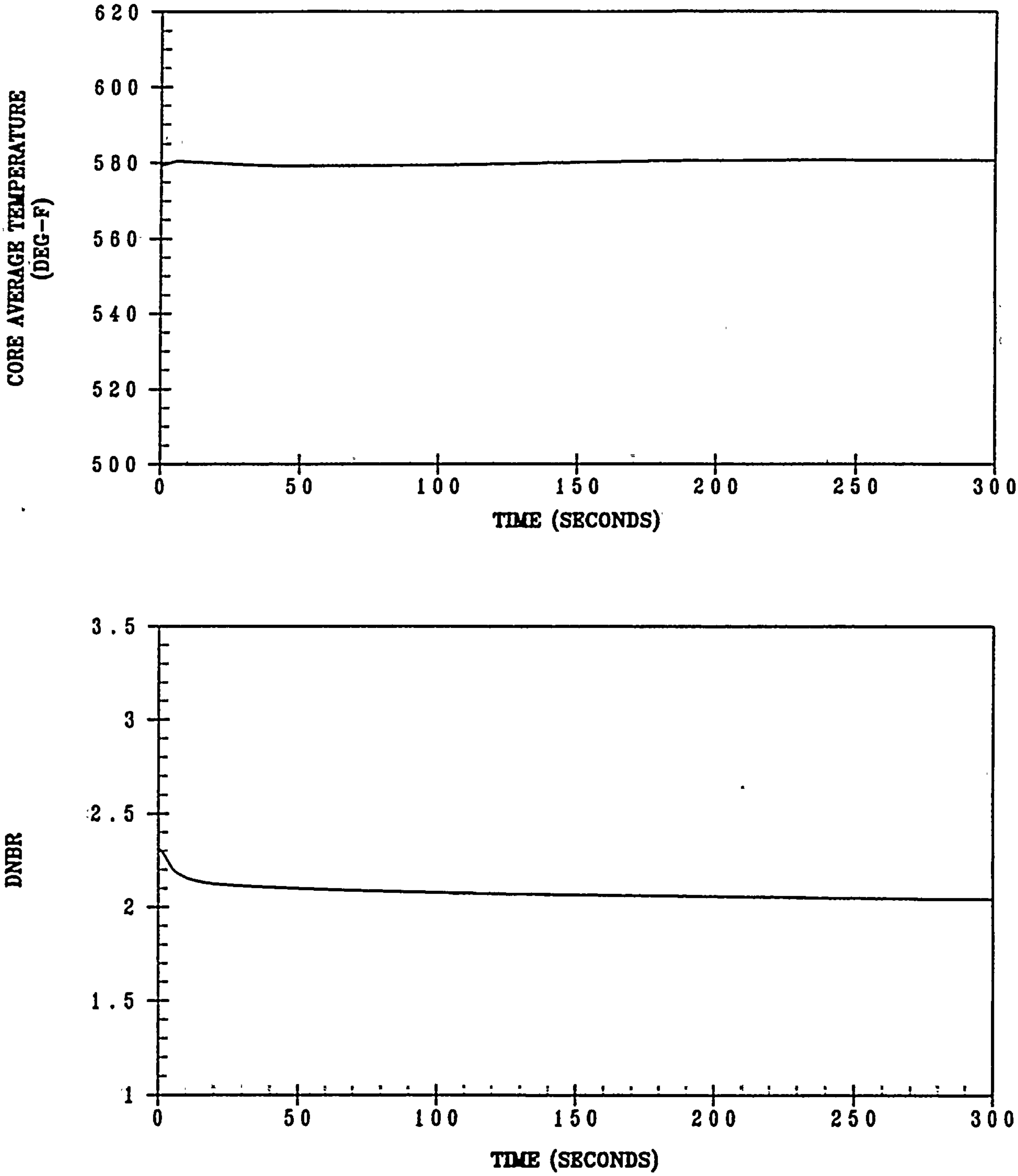


Figure 3.2.7-6 10% Step Load Increase Minimum Moderator Feedback Automatic Rod Control

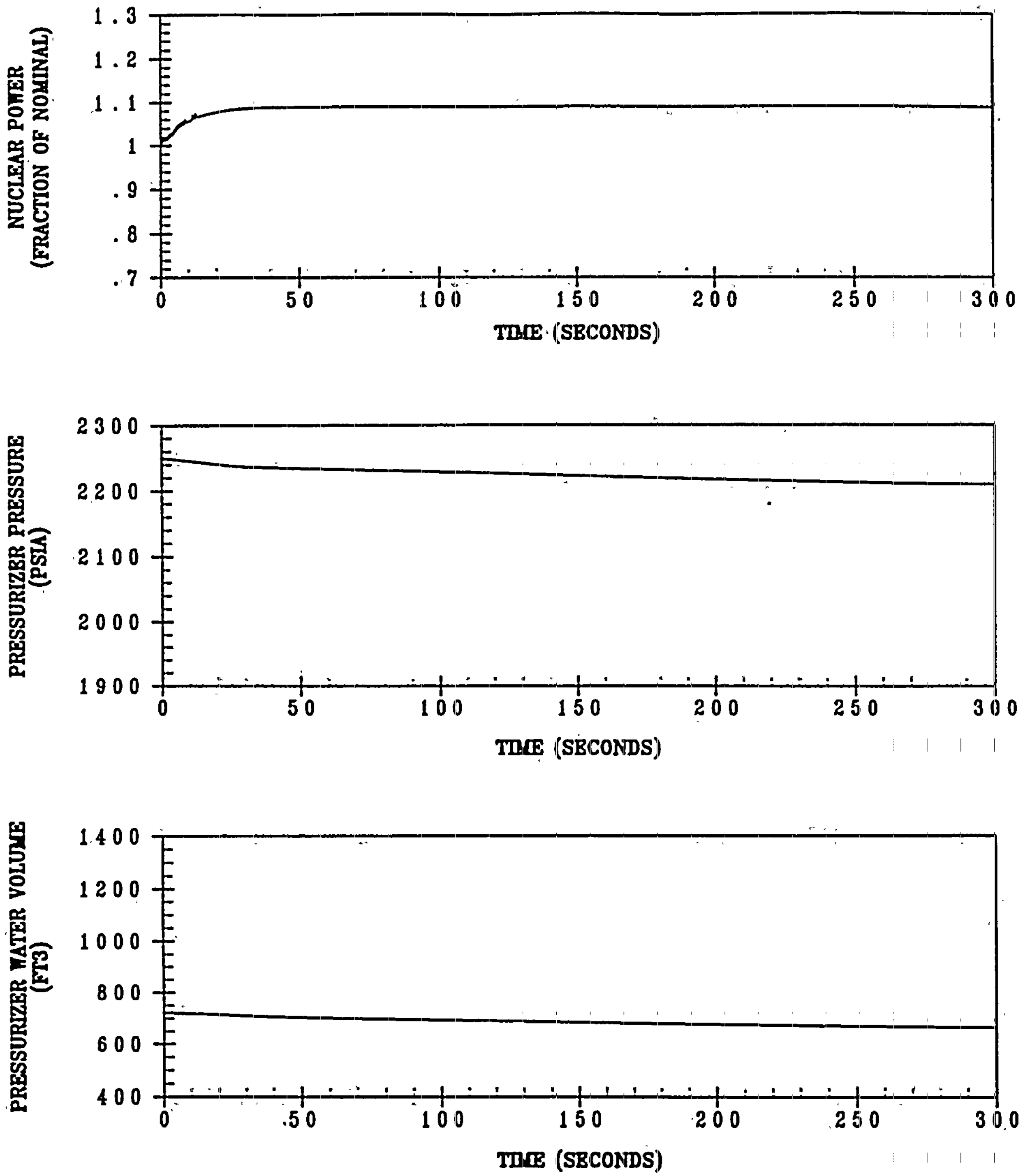


Figure 3.2.7-7 10% Step Load Increase Maximum Moderator Feedback Automatic Rod Control

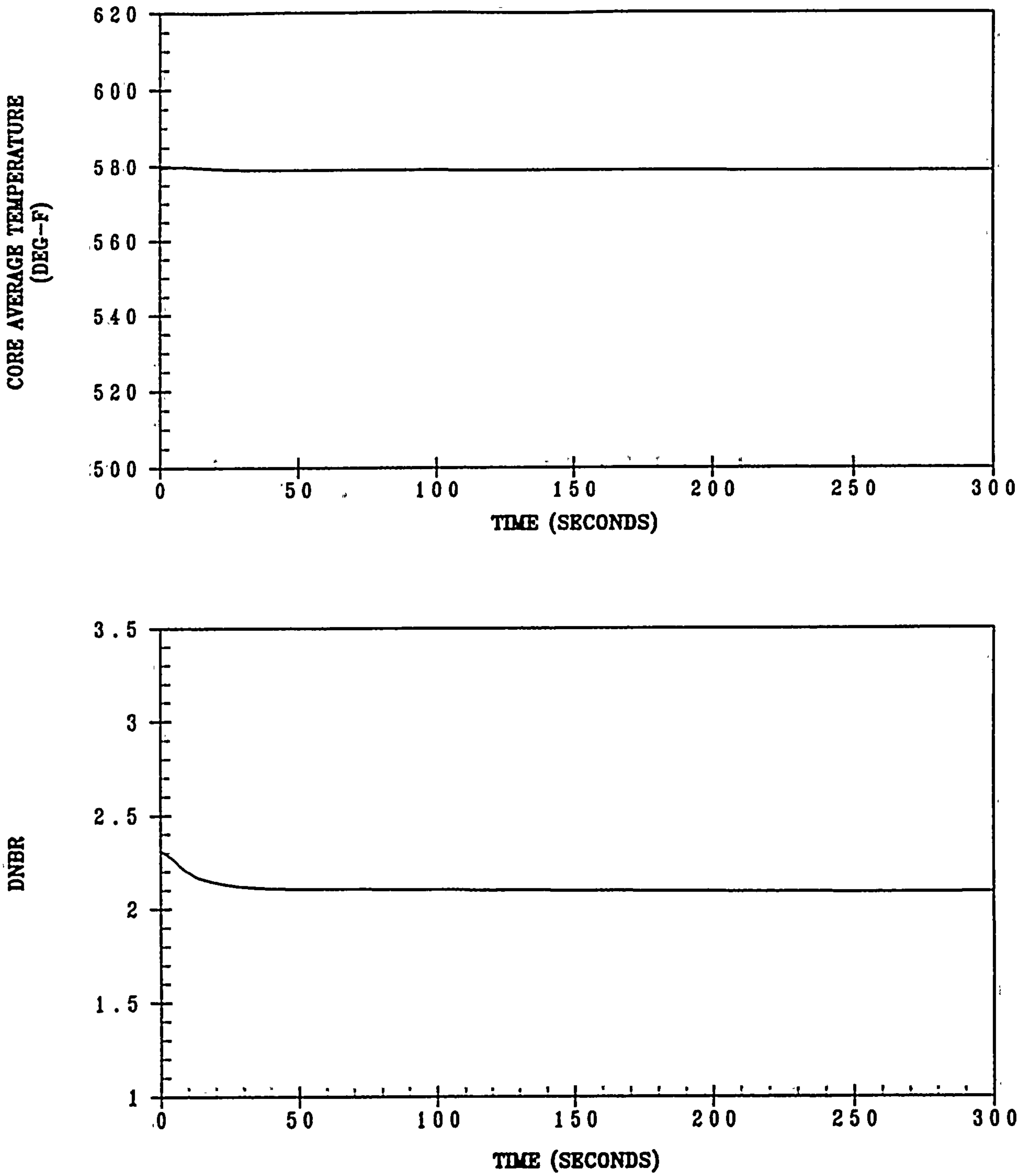


Figure 3.2.7-8 10% Step Load Increase Maximum Moderator Feedback
Automatic Rod Control

3.2.8 Loss of Reactor Coolant Flow

3.2.8.1 Partial / Complete Loss of Forced Reactor Coolant Flow

3.2.8.1.1 Identification of Causes and Accident Description

A loss of forced coolant flow incident may result from a mechanical or electrical failure in one or more reactor coolant pumps (RCPs), or from a fault in the power supply to these pumps. If the reactor is at power at the time of the event, the immediate effect of loss of forced coolant flow is a rapid increase in the coolant temperature. Promptly tripping the reactor ensures that this rapid increase in coolant temperature does not violate DNB.

Normal power supplies for the RCP pumps are A and B 4.16 kV buses supplied from the auxiliary transformer, one of which supplies power to one of the three pumps and the other of which supplies power to two of the three pumps. When a generator trip occurs, the buses automatically fast transfer to the startup transformer supplied from external power lines so that the pumps will continue to provide forced coolant flow to the core.

The following signals provide the necessary protection against a loss of coolant flow incident:

- Undervoltage (4.16 kV bus A or B) or underfrequency on reactor coolant pump power supply buses

- Underfrequency RCP breaker trips

- Low reactor coolant loop flow

- Pump circuit breaker opening

The reactor trip on undervoltage of 4.16 kV bus A or B is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of offsite power. This function is blocked below approximately 10 percent power (Permissive P-7).

The underfrequency RCP breaker trip is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. The reactor coolant pump underfrequency reactor trip function is blocked below P-7. In addition, the underfrequency function will open all RCP breakers whenever an underfrequency condition occurs (no P-7 or P-8 interlock) to ensure adequate RCP coastdown.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect one or two reactor coolant loops. It also serves as a backup to the undervoltage and underfrequency trips for the loss of all three reactor coolant pumps case. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above Permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive P-7) and the power level corresponding to Permissive P-8 (which is ~ 45% RTP), low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive P-7.

A reactor trip from pump breaker position is provided as a backup to the low flow signal. Similar to the low flow trip, above P-8, a breaker open signal from any pump will actuate a reactor trip, and between P-7 and P-8, a breaker open signal from any two pumps will actuate a reactor trip. Reactor trip on RCP breakers open is blocked below Permissive P-7.

3.2.8.1.2 Input Parameters and Assumptions

This accident is analyzed using the Revised Thermal Design Procedure (Reference 1). Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit value as described in Reference 1.

A conservatively large absolute value of the Doppler only power coefficient is used. The most-positive moderator temperature coefficient is assumed since this results in the maximum core power and hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Normal reactor control systems and engineered safety systems (e.g., SI) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

3.2.8.1.3 Description of Analysis

The following loss of flow cases are analyzed:

1. Loss of all three reactor coolant pumps with three loops in operation.

2. Loss of two reactor coolant pumps with three loops in operation.

These transients are analyzed by three digital computer codes. First, the LOFTRAN code (Reference 2) is used to calculate the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The flow coastdown analysis performed by LOFTRAN is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the as-built pump characteristics and is based on high estimates of system pressure losses.

The FACTRAN code (Reference 3) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC (Reference 6) code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical and thimble cells.

3.2.8.1.4 Acceptance Criteria

Partial Loss of Flow is an ANS Condition II event and Complete Loss of Flow is an ANS Condition III event. Both are analyzed to Condition II criteria. The immediate effect of either a partial or complete loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature and subsequent increase in reactor coolant system (RCS) pressure. The following 3 items summarize the criteria associated with this event.

- The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures.
- Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value which would cause fuel centerline melt.

3.2.8.1.5 Results

The complete loss of flow event is the most DNB limiting of the two cases presented in Reference 4. The reactor is assumed to trip on an undervoltage reactor trip signal for the complete loss of flow case resulting from a loss of power to the RCPs. Reactor trip for the partial loss of flow case occurs on a low flow signal. The THINC-IV (Reference 7) analyses for these scenarios confirm that the minimum DNBR values are greater than the safety analysis limit value. Fuel clad damage criteria are not challenged in either the partial or complete loss of forced reactor coolant flow events, since the DNB criterion is met.

The analyses of the partial and complete loss of flow events also demonstrate that the peak RCS and Main Steam system pressures are well below acceptable limits.

The calculated sequence of events for the cases presented in Section 14.1.9 of the UFSAR (Reference 4) is shown in Table 3.2.8-1. Figures 3.2.8-1 through 3.2.8-4 show the transient response for the loss of power to all reactor coolant pumps. Figures 3.2.8-5 through 3.2.8-8 show the transient response for the loss of two reactor coolant pumps with three loops initially in operation.

3.2.8.1.6 Conclusions

The analyses performed at the uprated conditions demonstrate that for the above loss of flow incidents, the DNBR does not decrease below the safety analysis limit value at any time during the transient; thus, no fuel or clad damage is predicted. The peak primary and secondary pressure remain below 100% of design at all times. All applicable acceptance criteria are therefore met.

The protection features presented in Section 3.2.8.1.1 provide mitigation for the loss of forced reactor coolant flow transients such that the above criteria are satisfied.

3.2.8.2 Locked Rotor/Shaft Break

3.2.8.2.1 Identification of Causes and Accident Description

The event postulated is an instantaneous seizure of a reactor coolant pump rotor or the sudden break of the shaft of the reactor coolant pump (RCP). Flow through the affected reactor coolant loop is rapidly reduced, leading to initiation of a reactor trip on a low Reactor Coolant System (RCS) flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip due to turbine trip on reactor trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the event. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in the analysis.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. The analysis considers only one of the two scenarios; it represents the most-limiting condition for the locked rotor and pump shaft break event.

3.2.8.2.2 Input Parameters and Assumptions

Two cases are evaluated in the analysis. Both assume one locked rotor/shaft break with a total of three loops in operation. The first case is aimed at maximizing the RCS pressure transient. This is done using the Standard Thermal Design Procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their maximum values consistent with the uprated full-power conditions including allowances for calibration and instrument errors. This assumption results in a conservative calculation of the coolant insurge into the pressurizer which in turn results in a maximum calculated peak RCS pressure.

The second case is an evaluation of DNB in the core during the transient. This case is analyzed using the Revised Thermal Design Procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit value as described in Reference 1.

The reactivity coefficients assumed in both cases include a positive moderator temperature coefficient and a conservatively large (absolute value) of the Doppler-only power coefficient. For this analysis, the negative reactivity insertion upon trip is based on a 4% trip reactivity from full power.

The transient is evaluated with no loss of offsite power. The two unaffected RCPs continue to operate through the duration of the event.

Normal reactor control systems and engineered safety systems (e.g., SI) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

The offsite doses following a locked rotor event reflect the uprated power level of 2346 MWt (102% of 2300 core power), 10% failed fuel, and a pre-accident iodine spike (Reference 8). The assumptions used for the locked rotor analysis are summarized in Table 3.2.8.3.

3.2.8.2.3 Description of Analysis

The pressure case is analyzed using two digital computer codes. The LOFTRAN code (Reference 2) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and the peak RCS pressure. The reactor coolant flow coastdown analysis performed by LOFTRAN is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, the as-built pump characteristics, and is based on high estimates of system pressure losses. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code (Reference 3) which uses the core flow and the nuclear power values calculated by LOFTRAN. The FACTRAN code includes a film boiling heat transfer coefficient.

The case analyzed to evaluate core DNB uses LOFTRAN, FACTRAN and THINC (Reference 6). The LOFTRAN and FACTRAN codes are used in the same manner as in the previous case. The THINC code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN (Reference 6).

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 60 psi above the nominal pressure of 2250 psia to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 3.2.8-10 is at the point in the RCS having the maximum pressure (i.e., the outlet of the faulted loop's RCP).

For a conservative analysis of fuel rod behavior, the hot spot evaluation assumes that DNB occurs at the initiation of the transient and continues throughout the event. This assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux rises due to the temperature increase and positive MTC and then is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reaches 84.5 percent of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these systems are expected to function and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are modelled including the effects of the pressurizer safety valve loop seals using WOG methodology (Reference 5). The pressurizer safety valve includes a 4% uncertainty (1% set pressure shift and a 3% set pressure tolerance) over the nominal setpoint of 2500 psia. Additionally, no steam flow is assumed until the valve loop seals are purged.

Evaluation of DNB in the Core During the Event

For this event, DNB is assumed to occur in the core and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the value at the initial core power level. The number of rods-in-DNB are conservatively calculated for use in dose consequence evaluations.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation (Reference 3). The fluid properties are evaluated at the film temperature (average between the wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to the clad temperature response. As indicated earlier, DNB was assumed to start at the beginning of the transient.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between the fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and clad. For the initial portion of the transient, a high gap coefficient produces higher clad temperatures since the heat stored and generated in the fuel redistributes itself in the cooler cladding. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperatures to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation (Reference 3) shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(W^2)}{dt} = 33.3 \times 10^6 \, e^{-(45500/1.986T)}$$

where: W = amount Zr reacted, mg/cm^2
t = time, sec
T = temperature, °K

The reaction heat is 1510 cal/gm. The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

3.2.8.2.4 Acceptance Criteria

An RCP locked rotor is an ANS Condition IV event. An RCP locked rotor results in a rapid reduction in forced reactor coolant loop flow which increases the reactor coolant temperature and subsequently causes the fuel cladding temperature and RCS pressure to increase. The following items summarize the criteria associated with this event.

- Fuel cladding damage (including melting), due to increased reactor coolant temperatures and the Zirconium-water reaction, must be shown not to occur.

- Pressure in the reactor coolant system should be maintained below 110% of the design pressures.

- Fuel temperature and fuel clad strain limits should not be exceeded even for rods experiencing DNB. The peak linear heat generation rate should not exceed a value which would cause fuel centerline melt.

- Rods-in DNB (dose calculation) should be less than or equal to 10%.

- Dose limit for a locked rotor is a "small fraction of " or 10% of the 10 CFR 100 guideline values.

The protection features described in Section 3.2.8.2.3 provide mitigation for a locked rotor transient such that the above criteria are satisfied.

3.2.8.2.5 Results

The calculated sequence of events is shown in Table 3.2.8-1. The transient results are shown in Figures 3.2.8-9 through 3.2.8-12. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. The results of these calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are also summarized in Table 3.2.8-2. The rods-in-DNB design criteria of less than 10% has been met.

The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary as follows:

	EB (0-2 Hr)	LPZ (0-24 Hr)
Thyroid	1.0 E0	4.0 E-1
γ-Body	9.9 E-2	1.5 E-2

3.2.8.2.6 Conclusions

The analysis performed at the uprated conditions demonstrates that for the above locked rotor event, since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F and the amount of zirconium-water reaction is small, the core will remain in place and intact with no loss of core cooling capability.

The analysis also confirms that the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered. The rods-in-DNB design criteria is also met. The offsite dose criterion were met and the locked rotor event does not present unacceptable risk to the public.

The offsite thyroid and γ-body doses are within the acceptance criteria of 10 CFR 100.

3.2.8.3 References

1. Friedland, A. J. and Ray, S.,"Revised Thermal Design Procedure", WCAP-11397-P-A, April 1989.

2. Burnett, T.W.T et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

3. Hargrove, H.G., "FACTRAN -- A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod", WCAP-7908-A, December 1989.

4. Turkey Point Plant Units 3 and 4 Updated Final Safety Analysis Report, Revision 12.

5. Barrett, G.O., et al.,"Pressurizer Safety Valve Set Pressure Shift", WCAP-12910, March 1991.

6. Shefchek, J., "Application of the THINC Program to PWR Design," WCAP-7359-L, August 1969.

7. Chelemer, H., Chu, P. T., Hochreiter, L. E., "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, February 1989.

Table 3.2.8-1

Sequence of Events - Loss of Flow Events

Case	Event	Time (sec)
Complete Loss of Forced Reactor Coolant Flow	Reactor coolant pump undervoltage trip setpoint reached, all pumps lose power and begin coasting down	0.0
	Rods begin to drop	2.0
	Minimum DNBR occurs	3.8
	Maximum RCS pressure	5.1
Partial Loss of Forced Reactor Coolant Flow	Two reactor coolant pumps lose power and begin coasting down	0.0
	Low flow reactor trip setpoint reached	2.0
	Rods begin to drop	3.0
	Minimum DNBR occurs	4.7
	Maximum RCS pressure	5.8
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Rotor on one pump locks	0.0
	Low flow reactor trip setpoint reached	0.05
	Rods begin to drop	1.05
	Maximum clad temperature occurs	3.5
	Maximum RCS pressure occurs	3.8

Table 3.2.8-2

Summary of Results for the Locked Rotor Transient

Criteria	3 Loops Initially Operating, One Locked Rotor
Maximum RCS Pressure (psia)	2690
Maximum Clad Temperature at Core Hot Spot (°F)	1906
$Zr\text{-}H_2O$ Reaction at Core Hot Spot (wt. %)	0.4

Table 3.2.8-3
Assumptions Used for Locked Rotor Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 μCi/gm of DE I-131
Activity Released to Reactor Coolant from Failed Fuel (Noble Gas & Iodine)	10.0% of Core Gap
Fraction of Core Activity in Gap (Noble Gas & Iodine)	0.10
Secondary Coolant Activity Prior to Accident	0.10 μCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	1.0 gpm
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	24 hr
Offsite Power	Lost*
Steam Release from SGs to Environment	521,000 lb (0-2 hr) 448,400 lb (2-8 hr) 1,196,000 lb (8-24 hr)

* Assumption of a loss of offsite power is conservative for the locked rotor dose analysis.

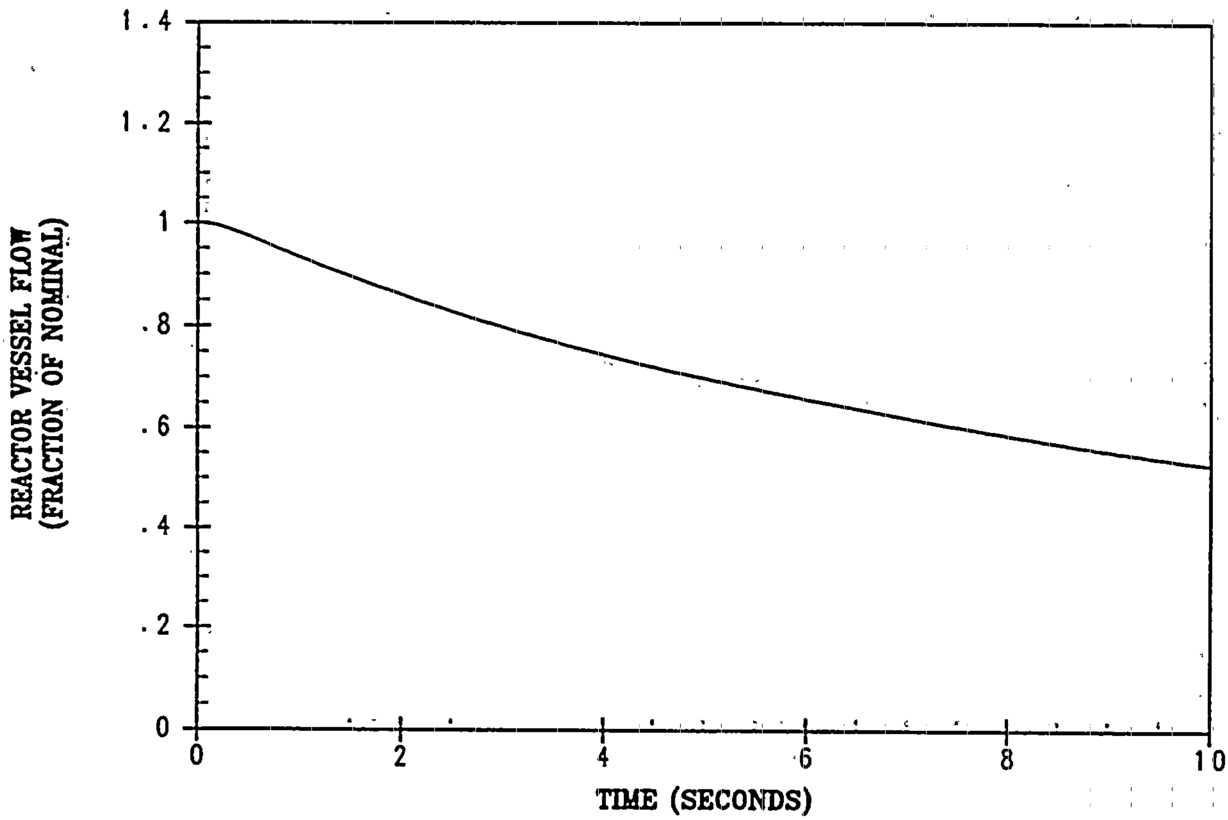


Figure 3.2.8-1 Core Flow vs. Time

Complete Loss of Forced Reactor Coolant Flow

(All loops operating, all loops coasting down)

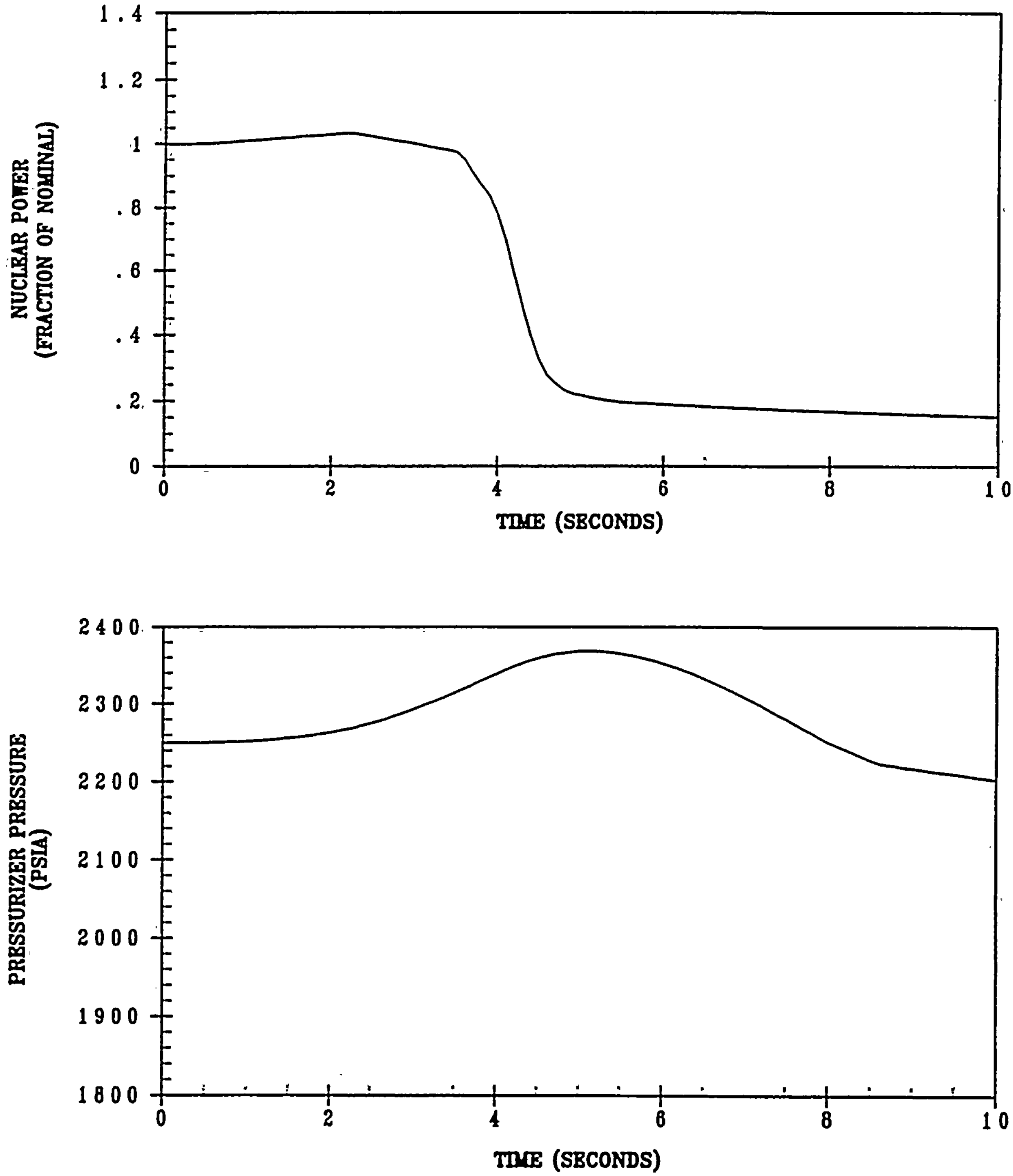
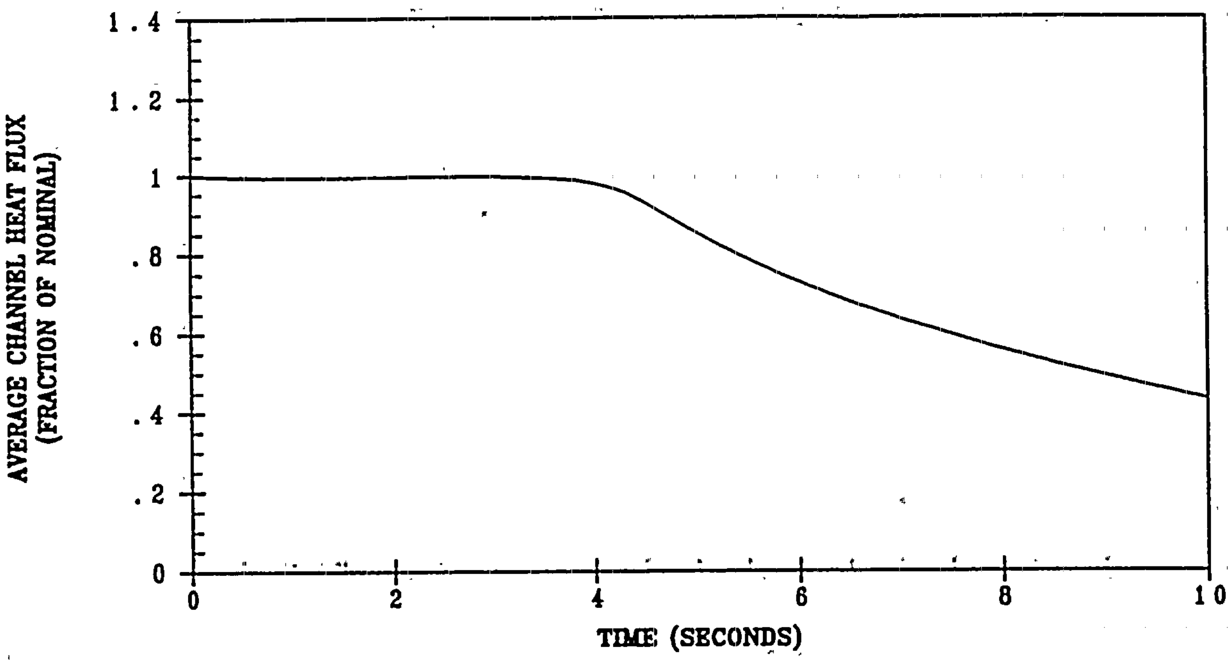


Figure 3.2.8-2 Nuclear Power and Pressurizer Pressure Transients
Complete Loss of Forced Reactor Coolant Flow
(All loops operating, all loops coasting down)



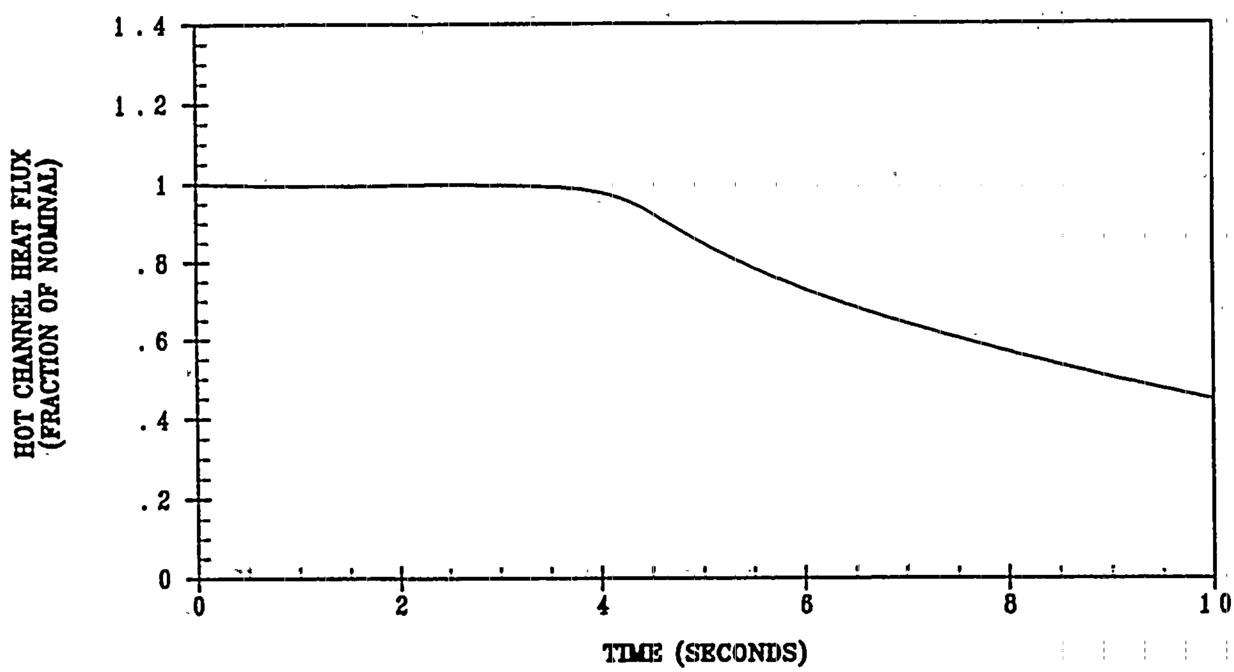


Figure 3.2.8-3 Average and Hot Channel Heat Flux Transients
Complete Loss of Forced Reactor Coolant Flow
(All loops operating, all loops coasting down)

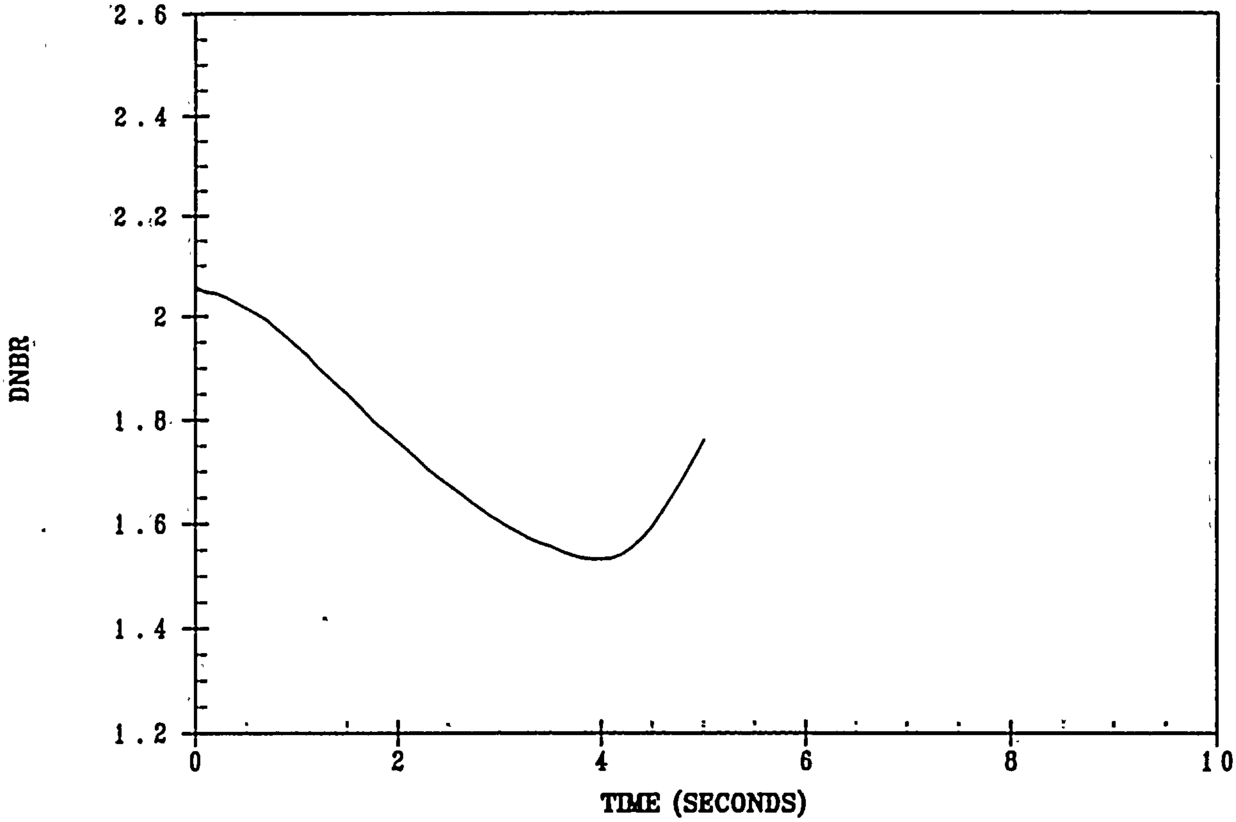
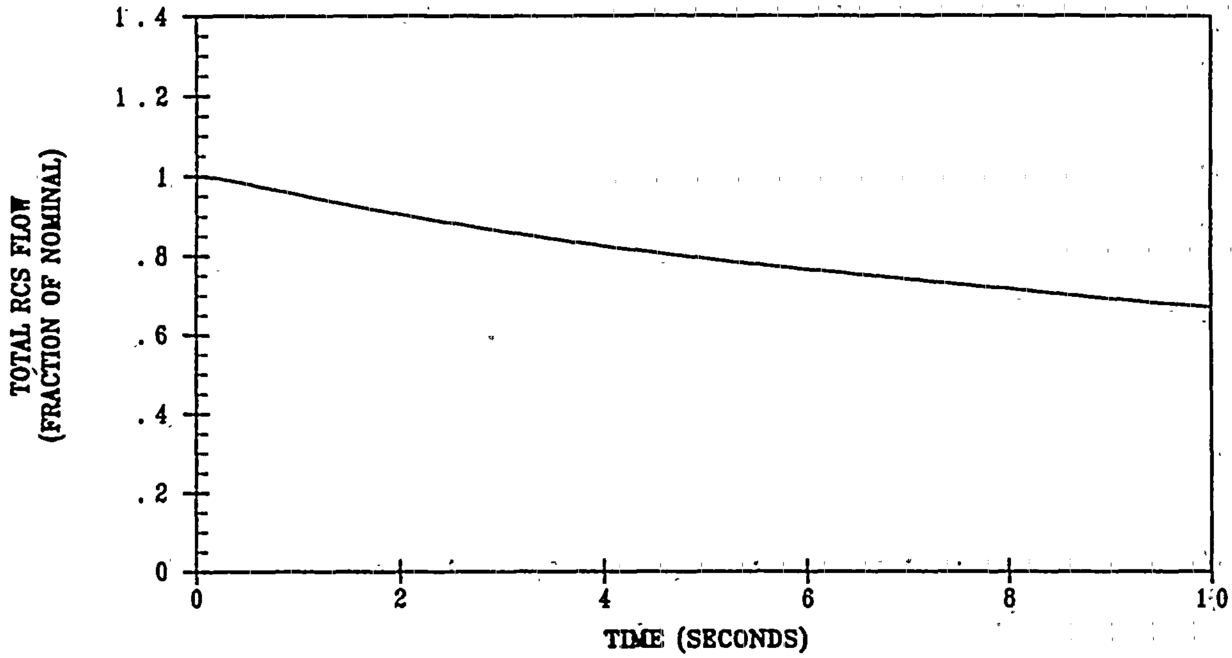


Figure 3.2.8-4 DNBR versus Time
Complete Loss of Forced Reactor Coolant Flow
(All loops operating, all loops coasting down)



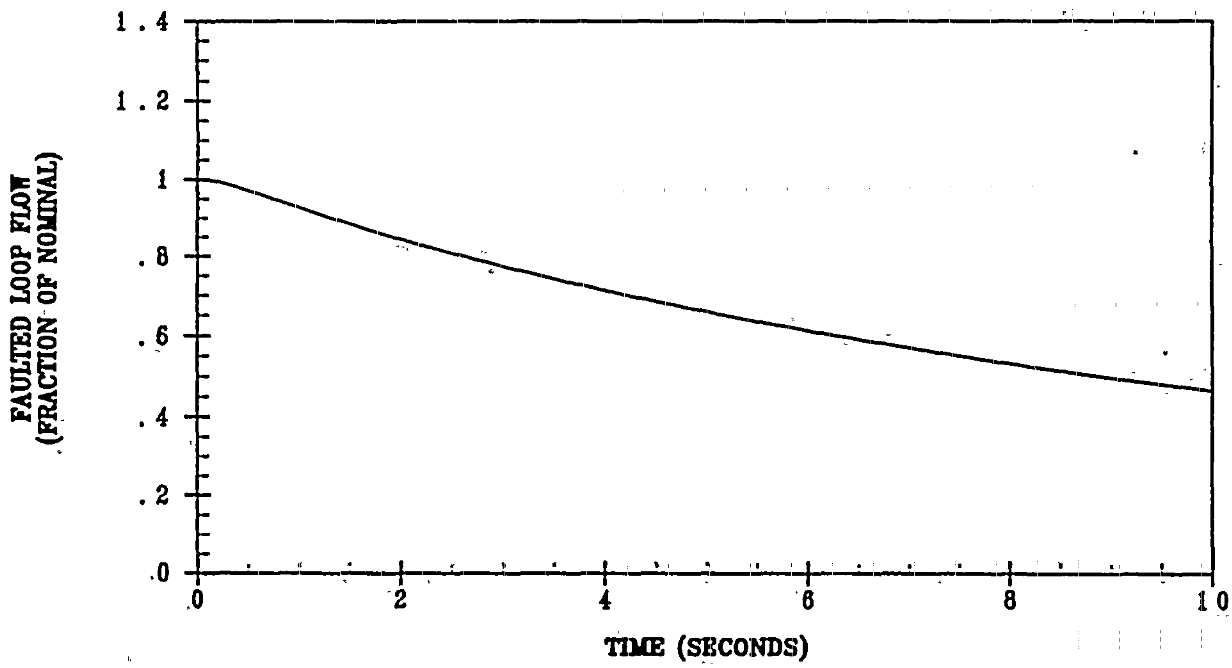
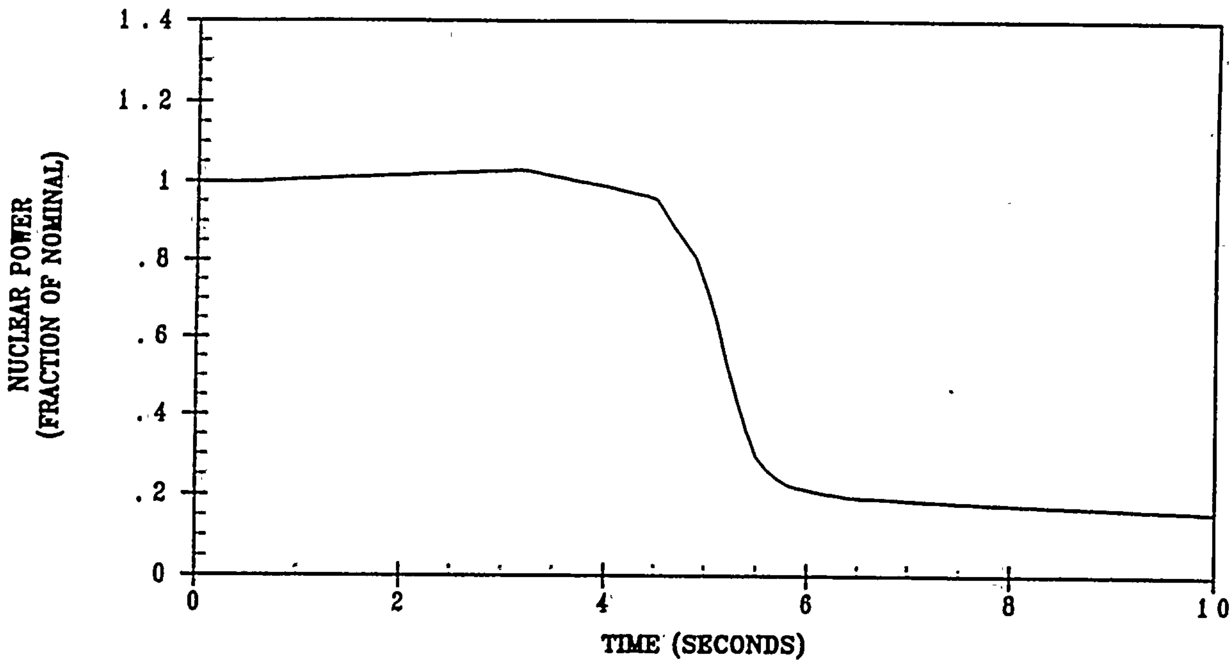


Figure 3.2.8-5 Flow Coastdown versus Time

Partial Loss of Forced Reactor Coolant Flow

(All loops initially operating, two loops coasting down)



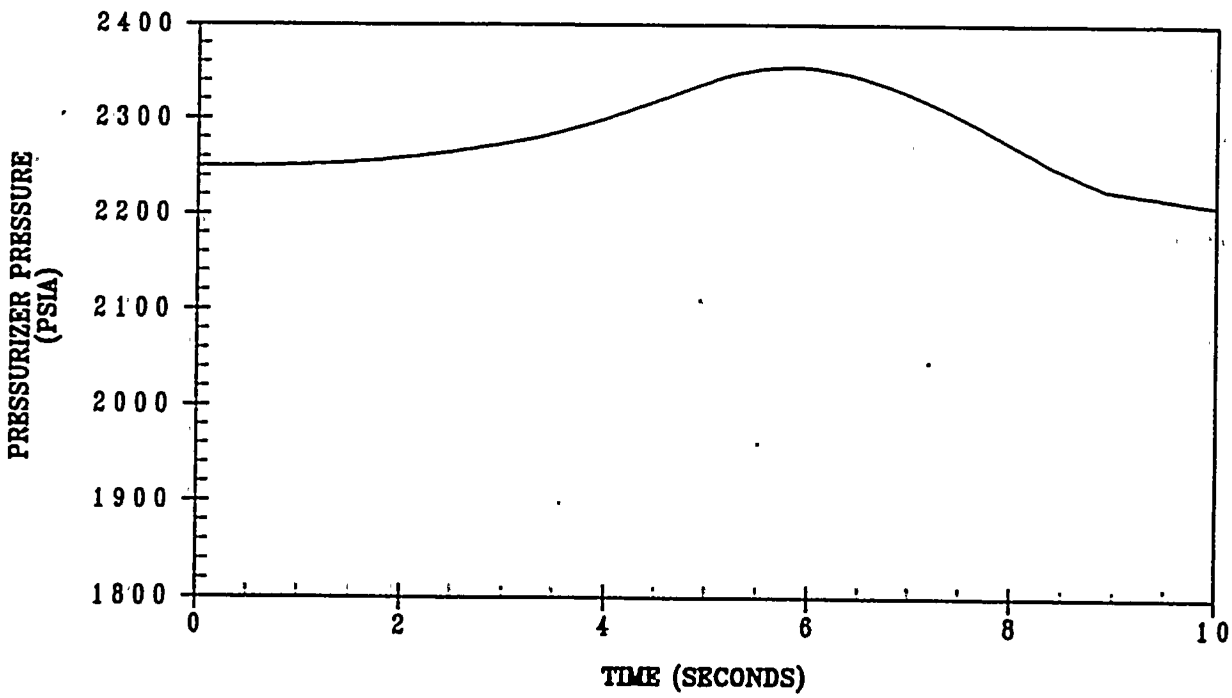
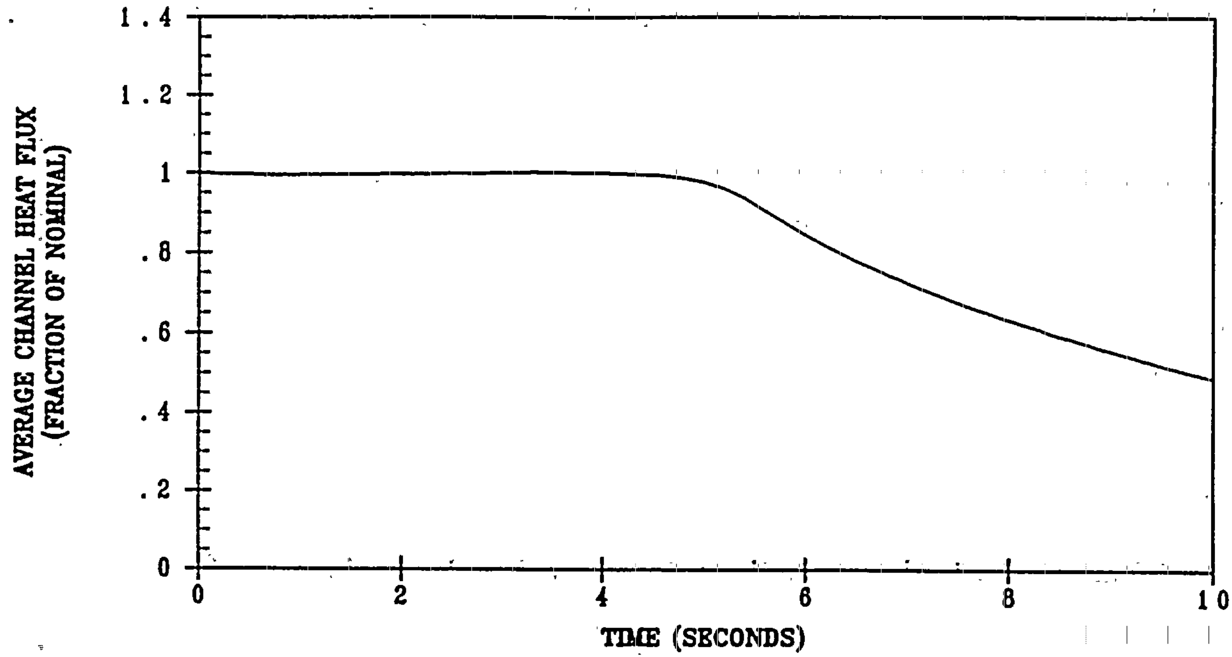


Figure 3.2.8-6 Nuclear Power and Pressurizer Pressure Transients

Partial Loss of Forced Reactor Coolant Flow

(All loops initially operating, two loops coasting down)



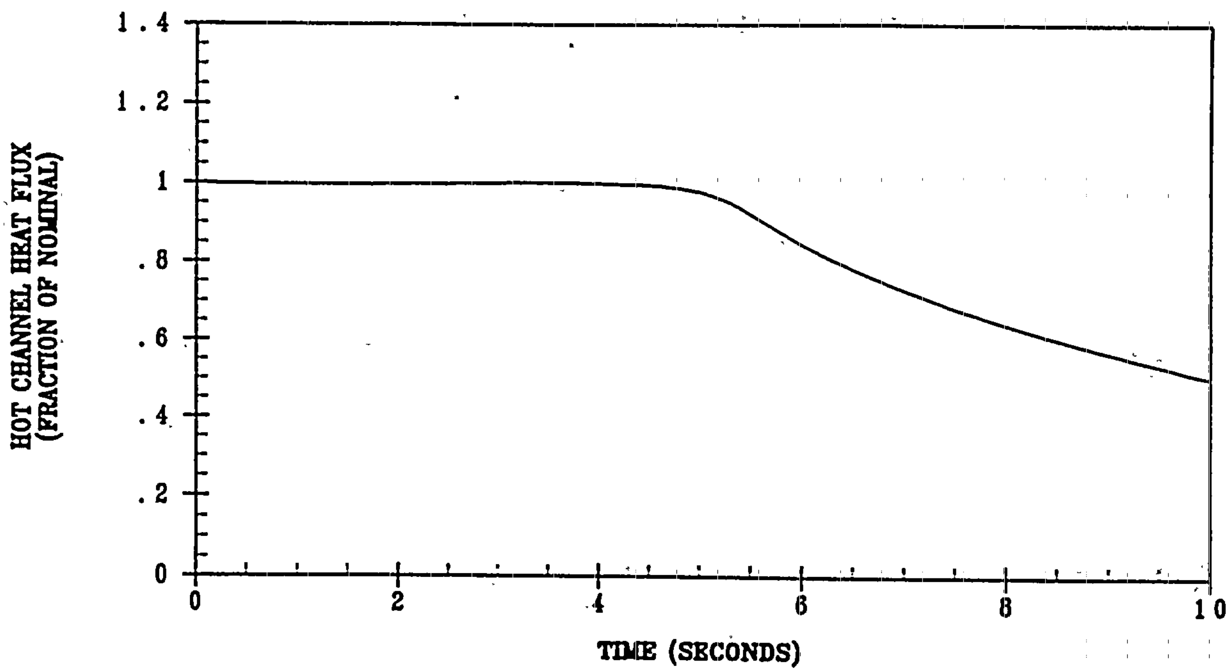


Figure 3.2.8-7 Average and Hot Channel Heat Flux Transients
Partial Loss of Forced Reactor Coolant Flow
(All loops initially operating, two loops coasting down)

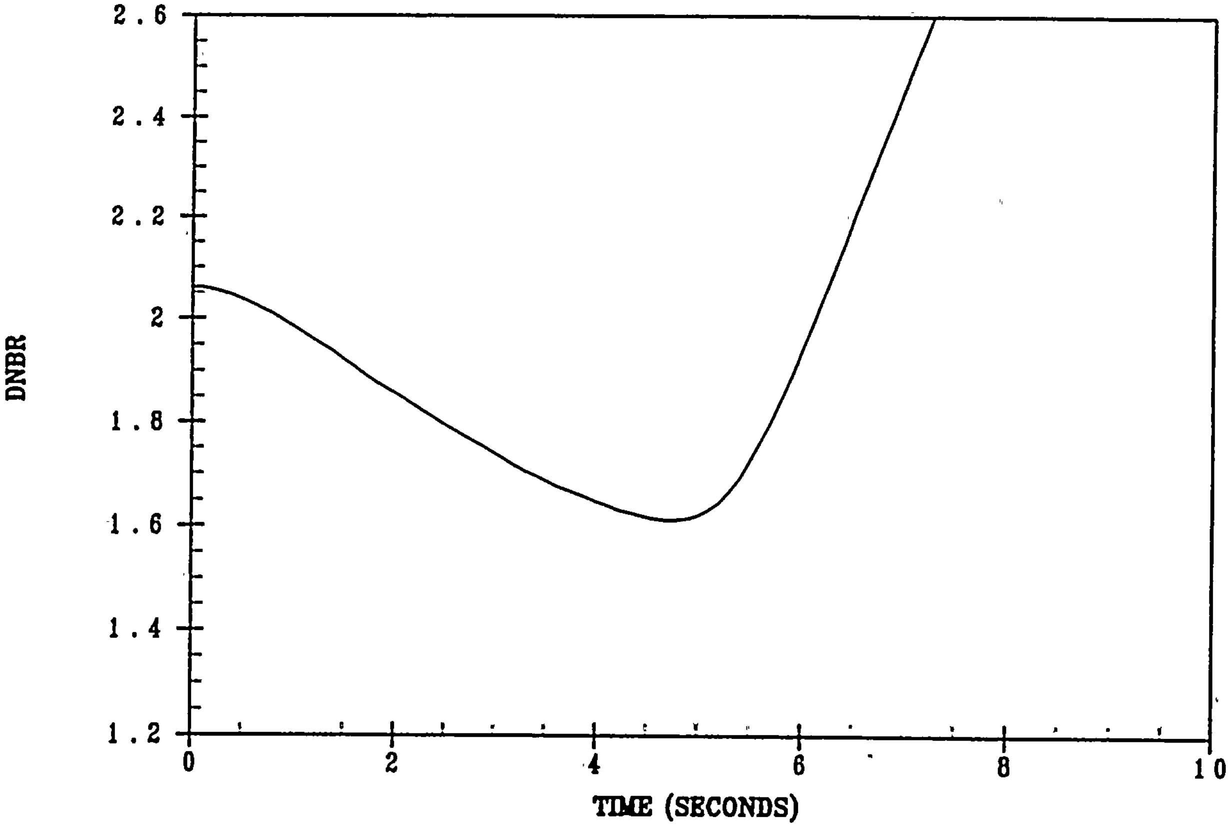


Figure 3.2.8-8 DNBR versus Time
Partial Loss of Forced Reactor Coolant Flow
(All loops initially operating, two loops coasting down)

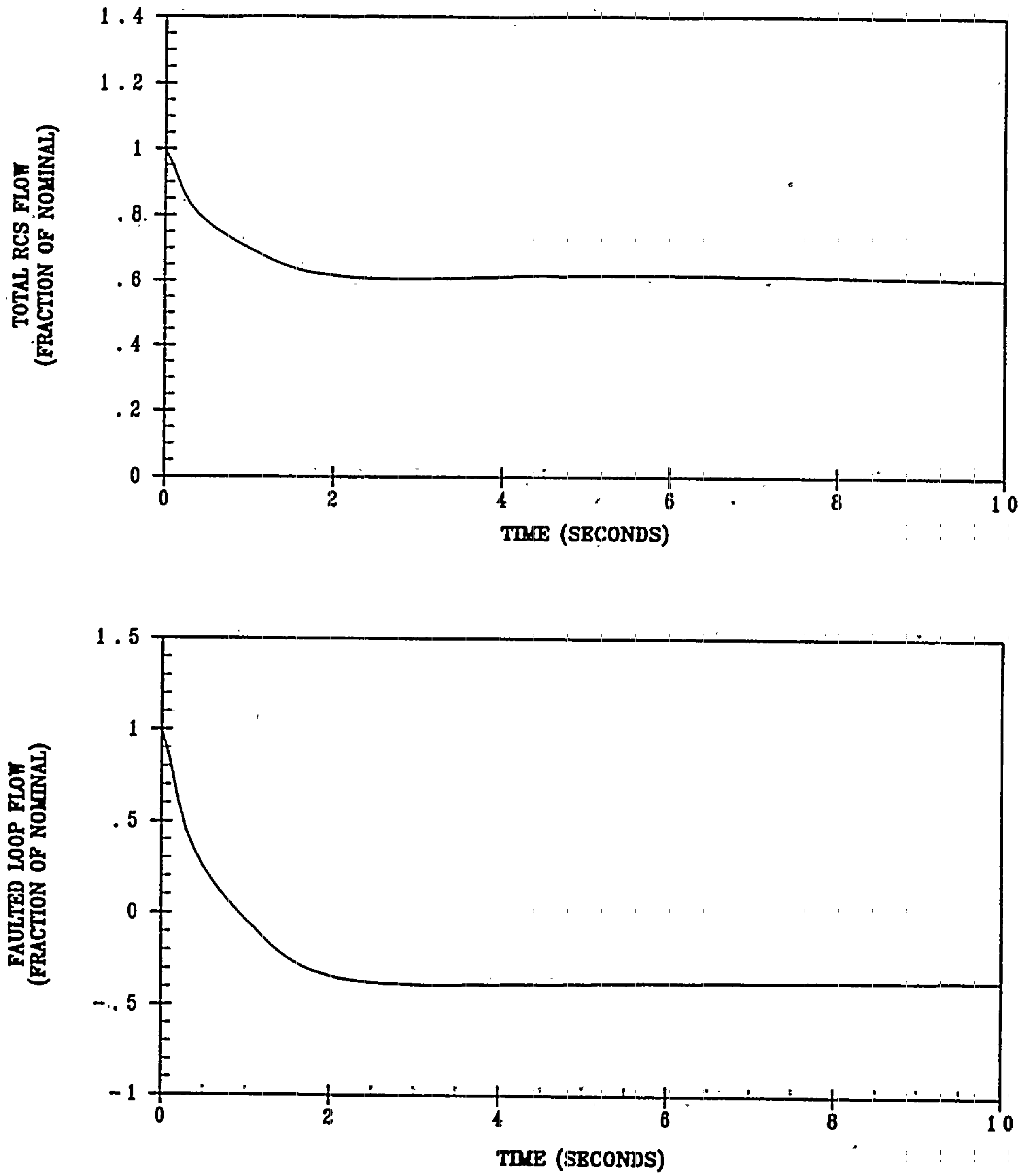


Figure 3.2.8-9 Flow Coastdown versus Time

Reactor Coolant Pump Shaft Seizure

(All loops initially operating, one locked rotor)

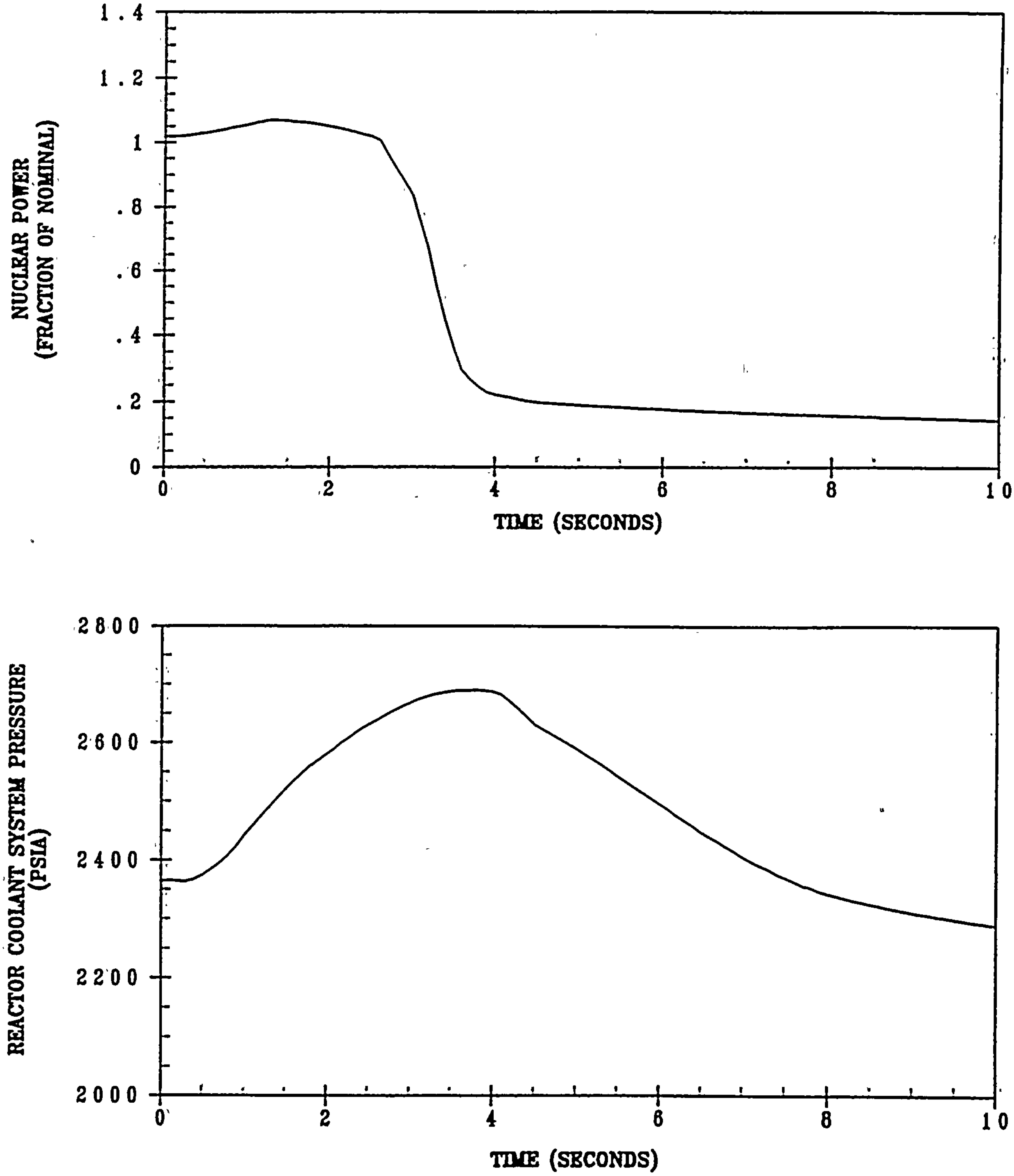


Figure 3.2.8-10 Nuclear Power and RCS Pressure Transients

Reactor Coolant Pump Shaft Seizure

(All loops initially operating, one locked rotor)

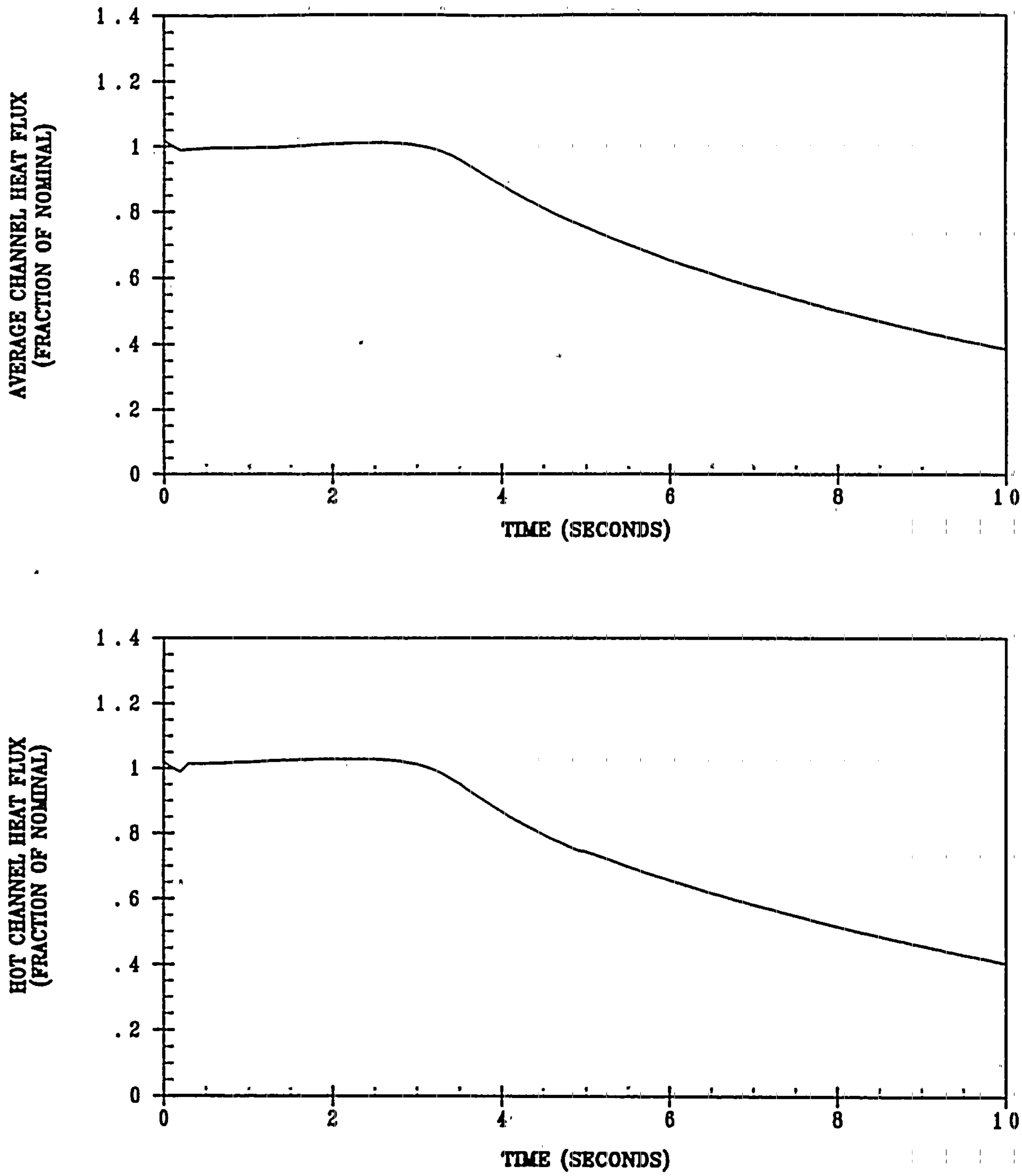


Figure 3.2.8-11 Average and Hot Channel Heat Flux Transients
Reactor Coolant Pump Shaft Seizure
(All loops initially operating, one locked rotor)

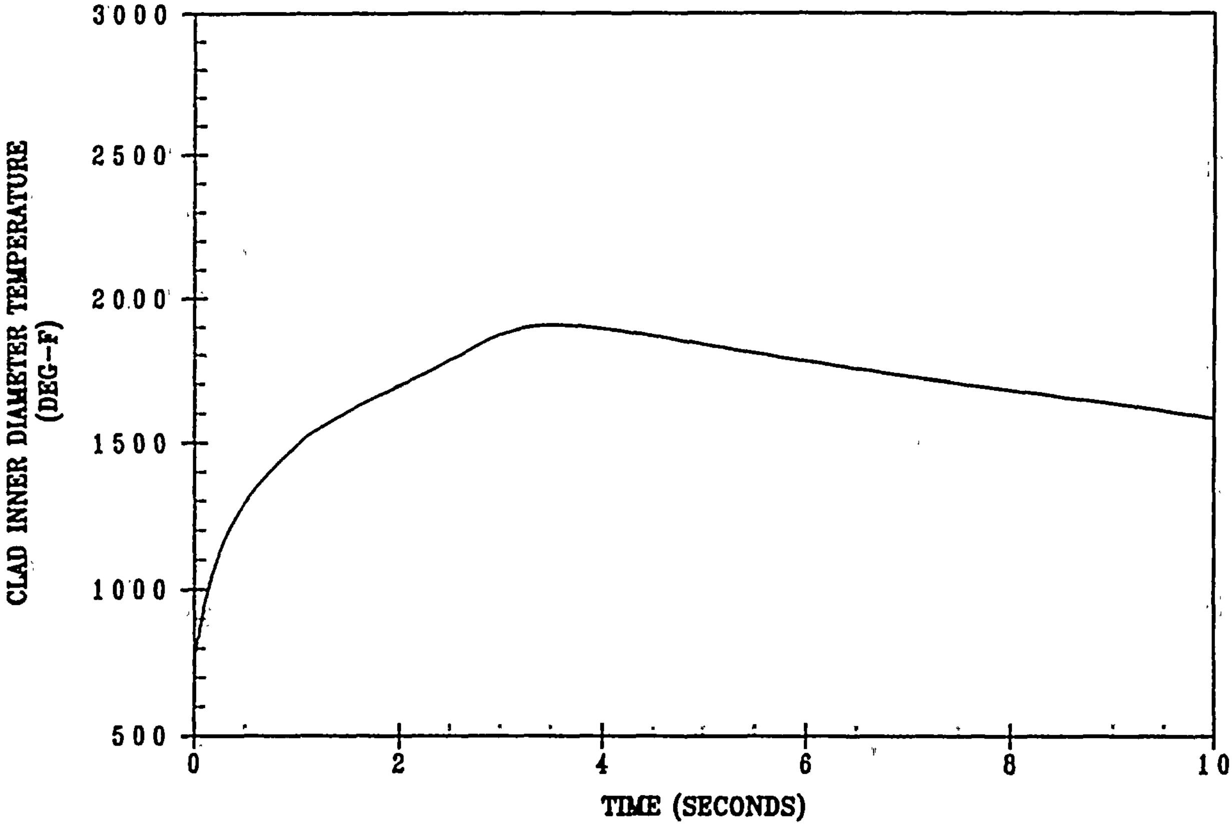


Figure 3.2.8-12 Clad Inner Diameter Temperature versus Time
Reactor Coolant Pump Shaft Seizure
(All loops initially operating, one locked rotor)

3.2.9 Loss of External Electrical Load and/or Turbine Trip

3.2.9.1 Identification of Causes and Accident Description

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. A loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating condition. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all non-emergency AC power is presented in Section 3.2.11.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The station is designed to accept a 50% step loss of load without actuating a reactor trip with all NSSS control systems in automatic (reactor control system, pressurizer pressure and level, steam generator water level control, and steam dumps). The automatic steam dump system, with 27% dump capacity to the condenser, together with the rod control system, is able to accommodate the 50% load rejection. Reactor power is reduced to a new equilibrium value consistent with the capability of the rod control system.

For a turbine or generator trip, the reactor would be tripped directly from a signal derived from the turbine autostop oil pressure (a two out of three signal). Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal or the overtemperature ΔT signal. In the event of feedwater flow also being lost, the reactor may also be tripped by a steam generator low-low water level signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod control, or the direct reactor trip on turbine trip.

The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer and steam generator safety valves are then able to maintain the RCS and Main Steam System pressures within 110% of the corresponding design pressure without a direct reactor trip on turbine trip action.

The Turkey Point Units 3 and 4 Reactor Protection System in conjunction with the primary and secondary system designs preclude overpressurization without requiring the automatic rod control, pressurizer pressure control and/or turbine bypass control system.

3.2.9.2 Input Parameters and Assumptions

Four cases are analyzed for a total loss of load from full power conditions: a) minimum reactivity feedback with pressure control, b) maximum reactivity feedback with pressure control, c) minimum reactivity feedback without pressure control and d) maximum reactivity feedback without pressure control. The primary concern for the cases analyzed with pressure control is minimum DNBR; the primary concern for the cases analyzed without pressure control is maintaining reactor coolant and main steam system pressure below 110% of the design pressure.

The major assumptions used in the analysis are summarized in the following.

Initial Operating Conditions

The cases with pressure control are analyzed using the Revised Thermal Design Procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full power operation. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit as described in Reference 1.

The cases without pressure control are analyzed using the Standard Thermal Design Procedure. Initial uncertainties on core power, reactor coolant temperature, and pressure are applied in the most conservative direction to obtain the initial plant conditions for the beginning of the transient.

Reactivity Coefficients

The total loss of load transient is analyzed with both minimum and maximum reactivity feedback. The minimum feedback (BOL) cases assume a positive moderator temperature coefficient and the least-negative Doppler coefficient. The maximum feedback (EOL) cases assume a large (absolute value) negative moderator temperature coefficient and the most-negative Doppler power coefficient.

Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual rod control. If the reactor were in automatic rod control, the control rod banks would move prior to trip and reduce the severity of the transient.

Pressurizer Spray and Power-Operated Relief Valves

The loss of load event is analyzed both with and without pressurizer pressure control (for both minimum and maximum reactivity feedback). The pressurizer PORVs and sprays are assumed operable for the cases with pressure control. The cases with pressure control minimize the increase in primary pressure which is conservative for the DNBR transient. The cases without pressure control maximize the pressure increase which is conservative for the RCS overpressurization criterion. In all cases the steam generator and pressurizer safety valves are operable.

The pressurizer safety valves are modelled including the effects of the pressurizer safety valve loop seals using WOG methodology (Reference 3). A total pressurizer safety valve setpoint tolerance of -3%, +2% is supported in the analysis. For those cases which are analyzed primarily for DNBR (pressurizer pressure control cases), the negative tolerance is applied to conservatively reduce the setpoint. For those cases which are analyzed primarily for peak RCS pressure, the positive tolerance is applied to conservatively increase the setpoint pressure. In the peak RCS pressure cases, the pressurizer safety valve includes a 3% uncertainty (1% set pressure shift and a 2% set pressure tolerance) over the nominal setpoint of 2500 psia. Additionally, no steam flow is assumed until the water in the valve loop seals is purged.

Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow; however, eventually auxiliary feedwater flow would be initiated and a stabilized plant condition would be reached.

Reactor Trip

Only the overtemperature ΔT, high pressurizer pressure, and low-low steam generator water level reactor trips are assumed operable for the purposes of this analysis. No credit is taken for a reactor trip on high pressurizer level or the direct reactor trip on turbine trip.

Steam Release

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. This assumption maximizes secondary pressure. The main steam safety valves are assumed to lift and be full open at 6% above their respective setpoints. This 6% includes 3% each for safety valve setpoint uncertainty and accumulation.

3.2.9.3 Description of Analyses

For the Loss of External Electrical Load/Turbine Trip analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This assumption is made to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins, by delaying reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst-case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

A detailed analysis using the LOFTRAN (Reference 2) computer code is performed to determine the plant transient conditions following a total loss of load. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system; and computes pertinent variables, including the pressurizer pressure, steam generator pressure, steam generator mass, and reactor coolant average temperature.

3.2.9.4 Acceptance Criteria

Based on its frequency of occurrence, the Loss of External Electrical Load/Turbine Trip accident is considered a Condition II event as defined by the American Nuclear Society. The criteria are as follows:

- The critical heat flux shall not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures.
- Fuel temperature and fuel clad strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value which would cause fuel centerline melt.

3.2.9.5 Results

The calculated sequence of events for the four Loss of External Electrical Load/Turbine Trip cases is presented in Table 3.2.9-1.

Case 1:

Figures 3.2.9-1 through 3.2.9-3 show the transient response for the total loss of steam load event under BOL conditions, including a positive moderator temperature coefficient, with pressure control. The reactor is tripped on overtemperature ΔT. The neutron flux increases until the reactor is tripped, and although the DNBR value decreases below the initial value, it remains well above the safety analysis limit throughout the entire transient. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case 2:

Figures 3.2.9-4 through 3.2.9-6 show the transient response for the total loss of steam load event under EOL conditions, assuming a conservatively large positive moderator density coefficient (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler only power coefficient, with pressure control. The reactor trip does not occur under these conditions. The plant stabilizes at a power level established by the relief capacity of the main steam safety valves. Without operator intervention, the system would eventually reach a low-low steam generator water level reactor trip condition as the secondary system inventory decreases. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The pressurizer pressure remains below the safety valve setpoint during the transient. The actuation of the main steam safety valves also maintain secondary pressure below 110% of the design value.

Case 3:

Figures 3.2.9-7 through 3.2.9-9 show the transient response for the total loss of steam load event under BOL conditions, including a positive moderator temperature coefficient, without pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux remains essentially constant at full power until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case 4:

Figures 3.2.9-10 through 3.2.9-12 show the transient response for the total loss of steam load event under EOL conditions, assuming a conservatively large positive moderator density coefficient (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler only power coefficient, without pressure control. The reactor is tripped on high pressurizer pressure. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

3.2.9.6 Conclusions

The results of this analysis show that the plant design is such that a total loss of external electrical load without a direct reactor trip presents no hazard to the integrity of the RCS or the main steam system. All of the applicable acceptance criteria are met. The minimum DNBR for each case is greater than the safety analysis limit value. The peak primary and secondary pressures remain below 110% of design at all times. The protection features presented in Section 3.2.9.2 provide mitigation of the Loss of External Electrical Load/Turbine Trip transient such that the above criteria are satisfied.

3.2.9.7 References

1. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397 (Proprietary), April 1989.

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

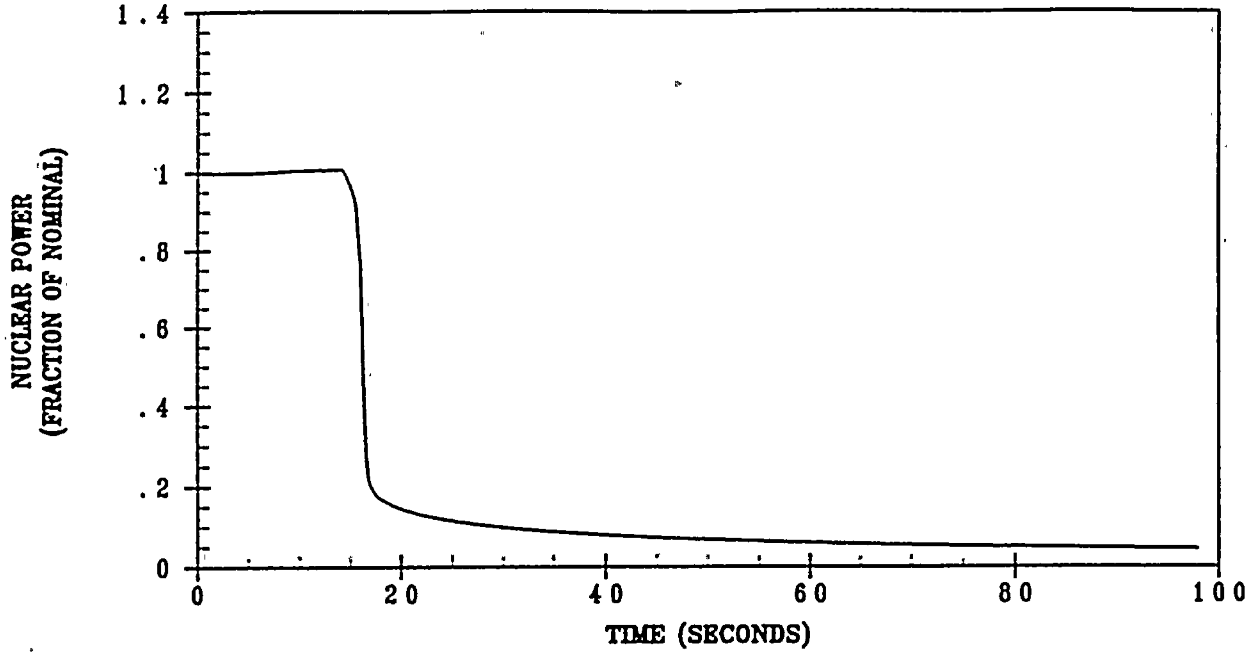
3. Barrett, G. O., et al., "Pressurizer Safety Valve Set Pressure Shift," WCAP-12910, March 1991.

Table 3.2.9-1
Sequence of Events - Loss of Load/Turbine Trip Event

Case	Event	Time (SEC)
1. With pressurizer pressure control (minimum reactivity feedback)	Turbine Trip	0.0
	Overtemperature ΔT Setpoint reached	12.0
	Peak pressurizer pressure occurs	13.8
	Rods begin to drop	14.0
	Minimum DNBR occurs	15.2
2. With pressurizer pressure control (maximum reactivity feedback)	Turbine Trip	0.0
	Peak pressurizer pressure occurs	7.6
(See Note 1)	Minimum DNBR occurs	*
3. Without pressurizer pressure control (minimum reactivity feedback)	Turbine Trip	0.0
	High Pressurizer Pressure Setpoint reached	7.2
	Rods begin to drop	9.2
	Peak pressurizer pressure occurs	10.2
	Minimum DNBR occurs	*
4. Without pressurizer pressure control (maximum reactivity feedback)	Turbine Trip	0.0
	High Pressurizer Pressure Setpoint reached	7.4
	Rods begin to drop	9.4
	Peak pressurizer pressure occurs	10.6
	Minimum DNBR occurs	*

* Never falls below initial value

Note 1. A reactor trip condition is never reached in the analysis. The reactor stabilizes at a power level established by the relief capacity of the MSSVs. Eventually, a low-low steam generator water level reactor trip would occur.



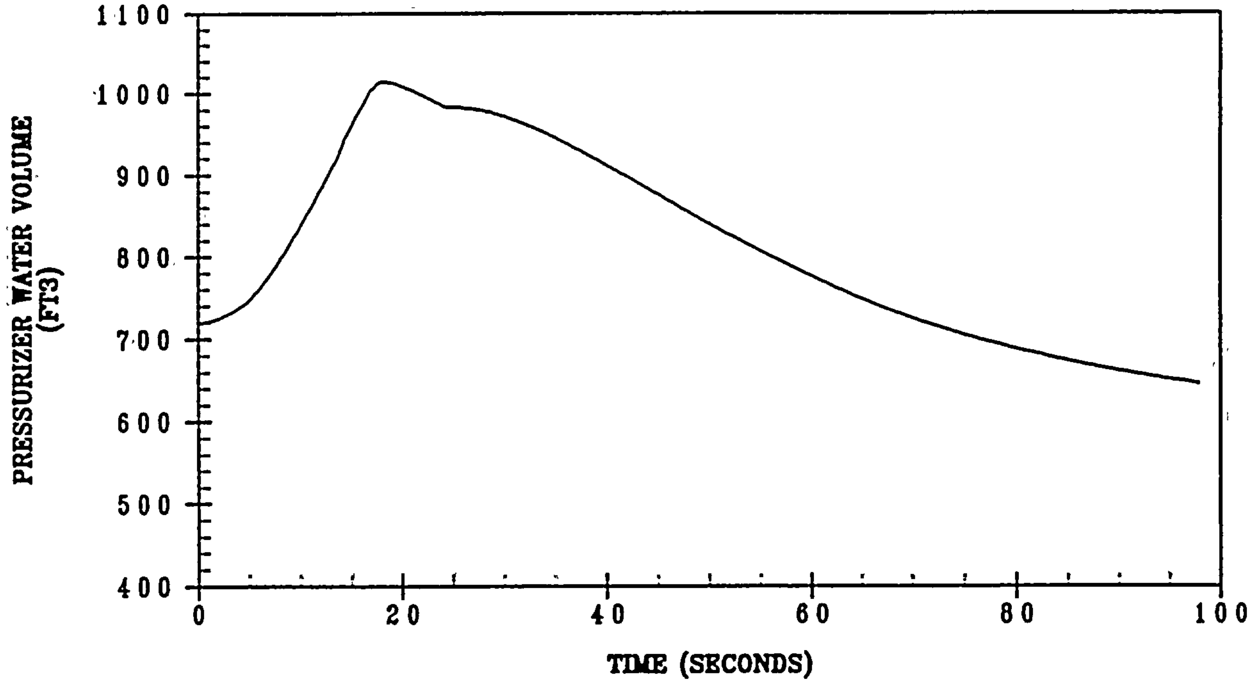


Figure 3.2.9-1 Total Loss of External Electrical Load with Pressure Control, Minimum Reactivity Feedback

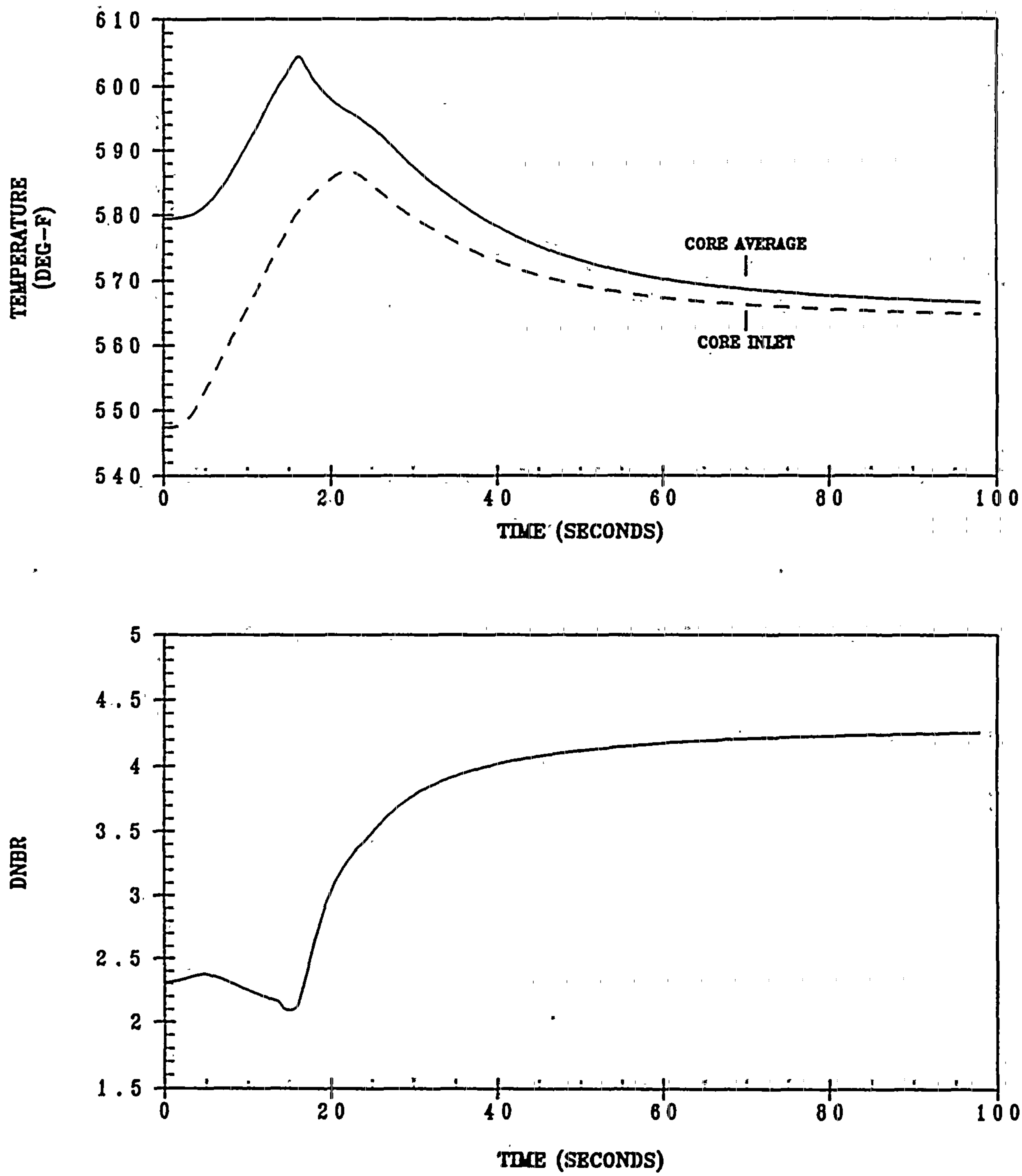


Figure 3.2.9-2 Total Loss of External Electrical Load with Pressure Control, Minimum Reactivity Feedback

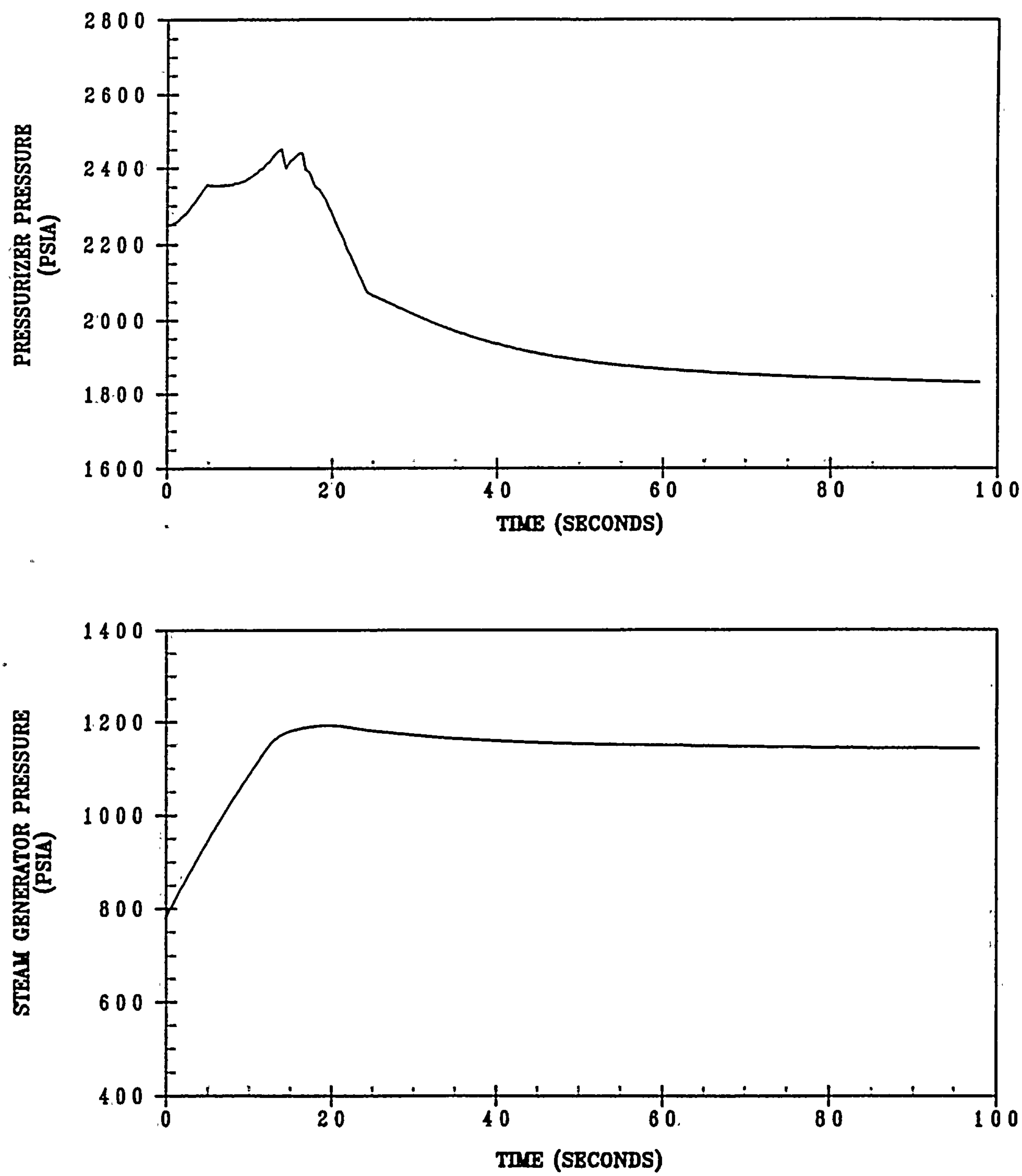


Figure 3.2.9-3 Total Loss of External Electrical Load with Pressure Control, Minimum Reactivity Feedback

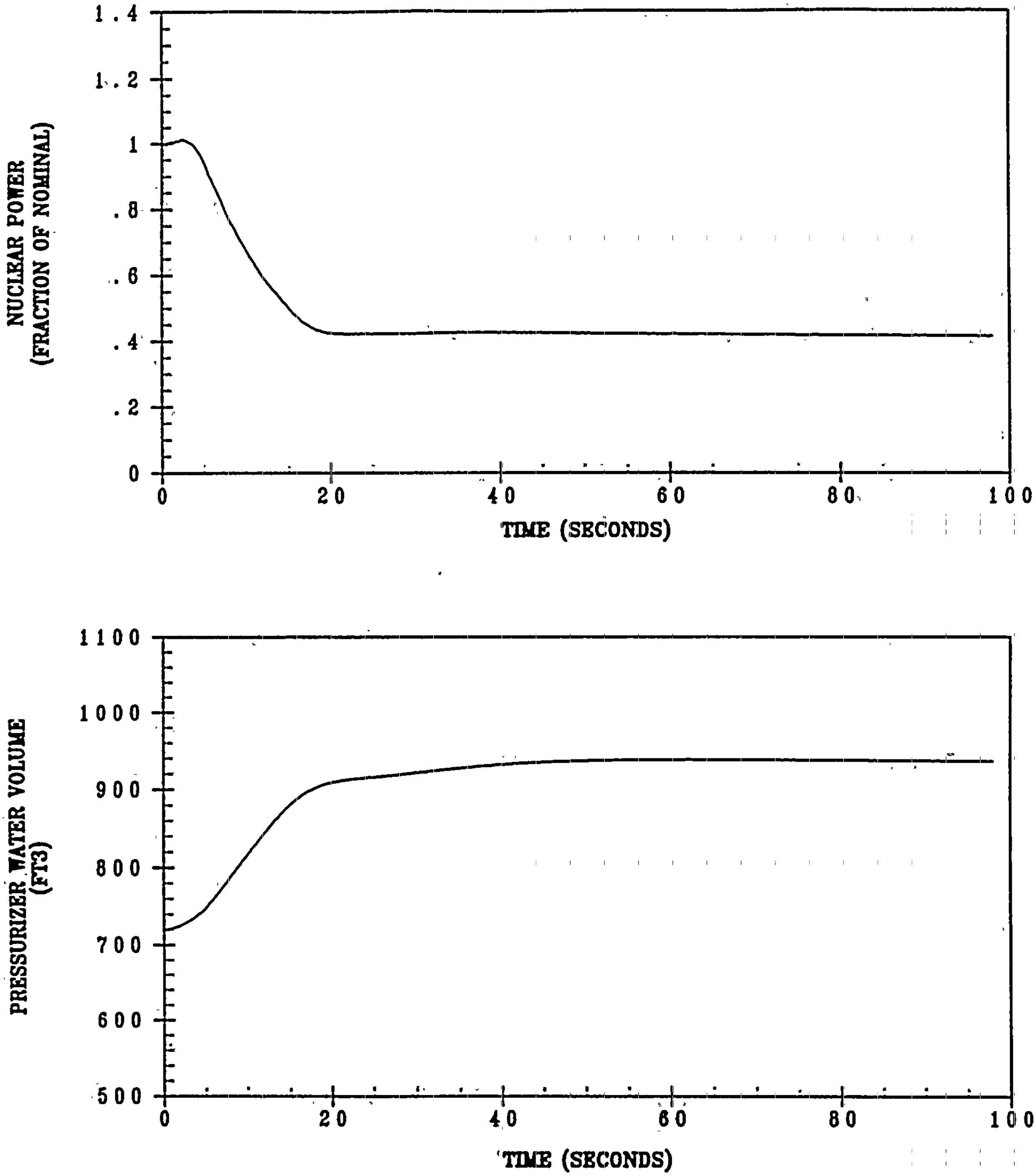


Figure 3.2.9-4 Total Loss of External Electrical Load with Pressure Control, Maximum Reactivity Feedback

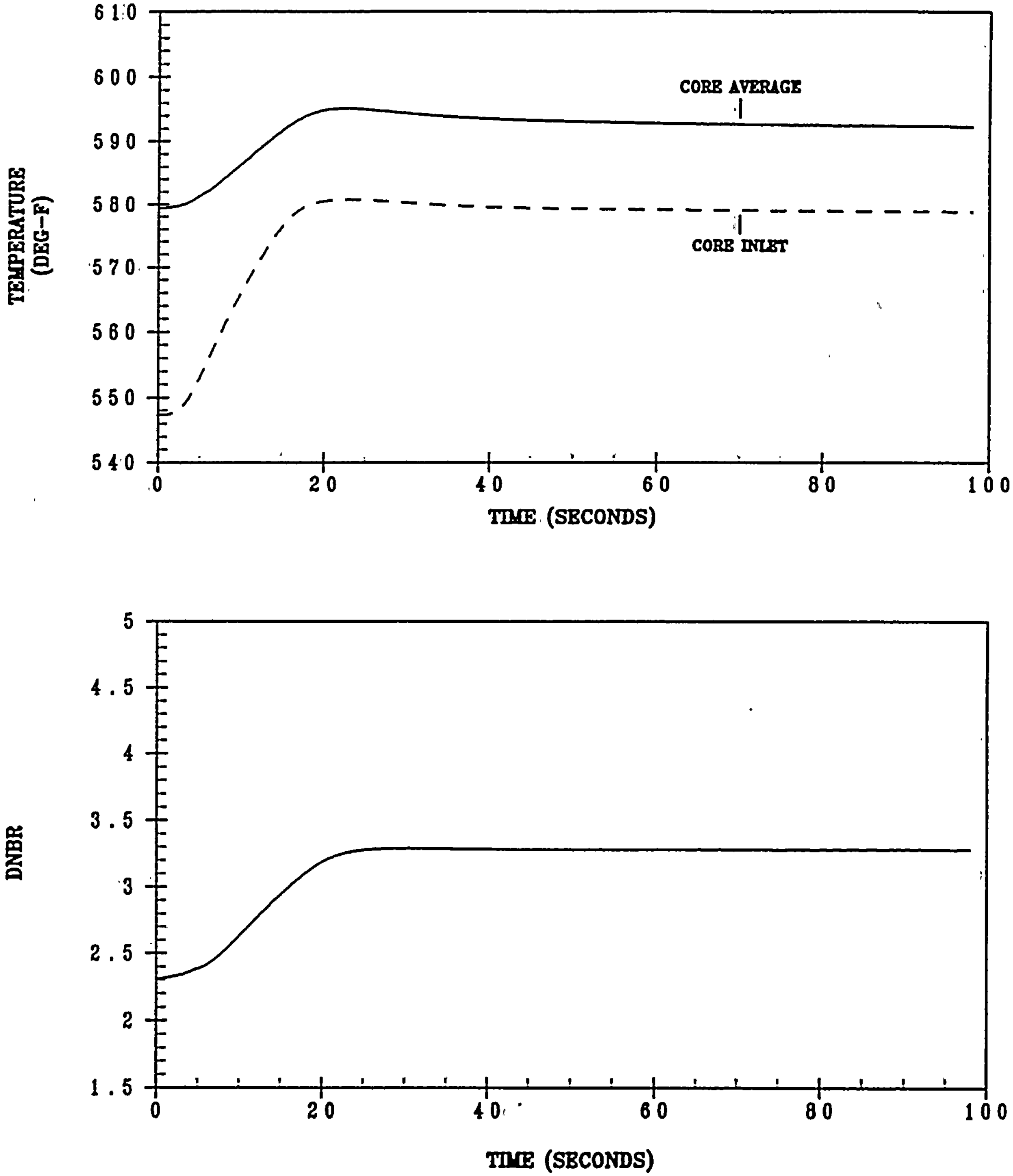


Figure 3.2.9-5 Total Loss of External Electrical Load with Pressure Control, Maximum Reactivity Feedback

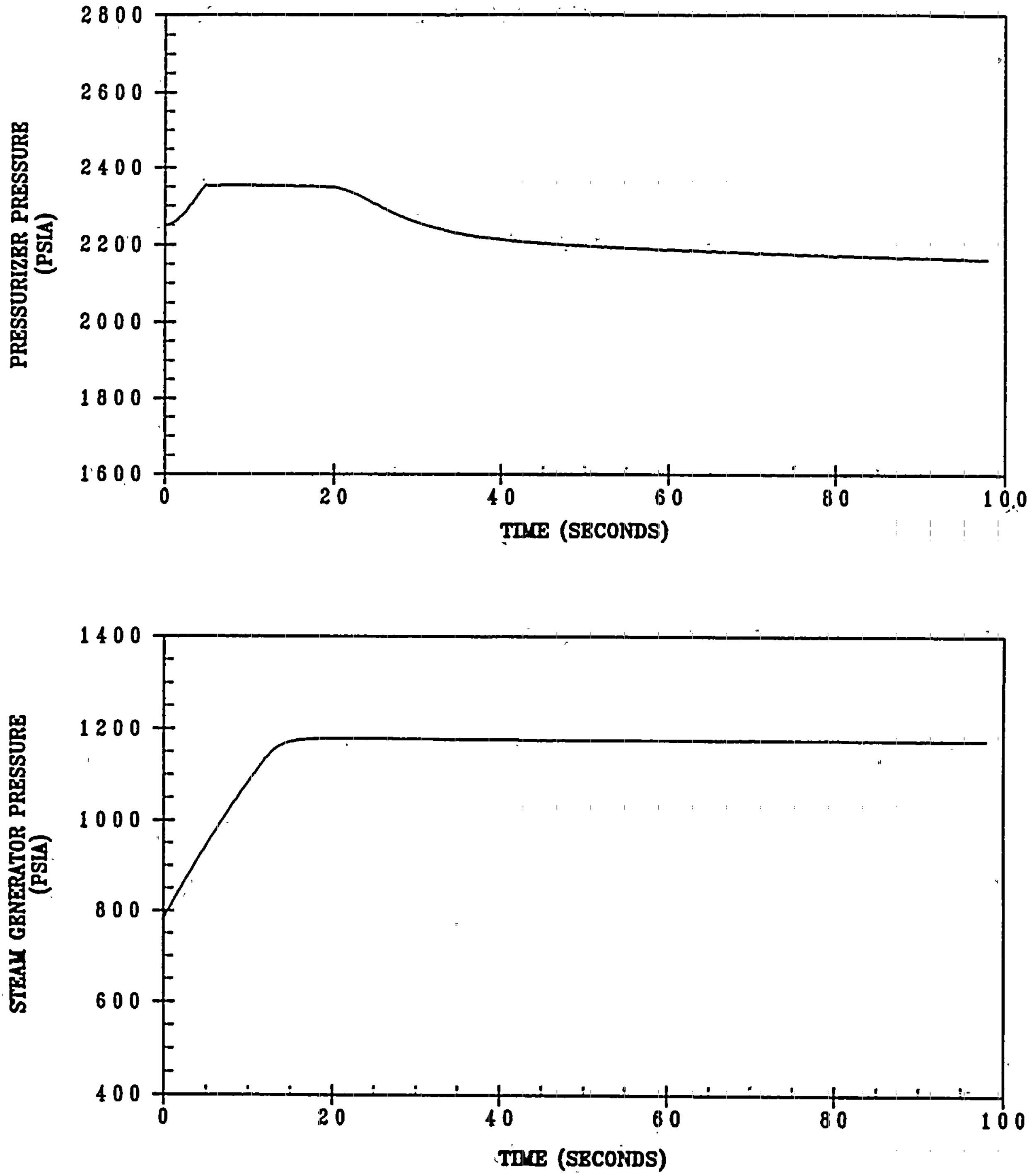
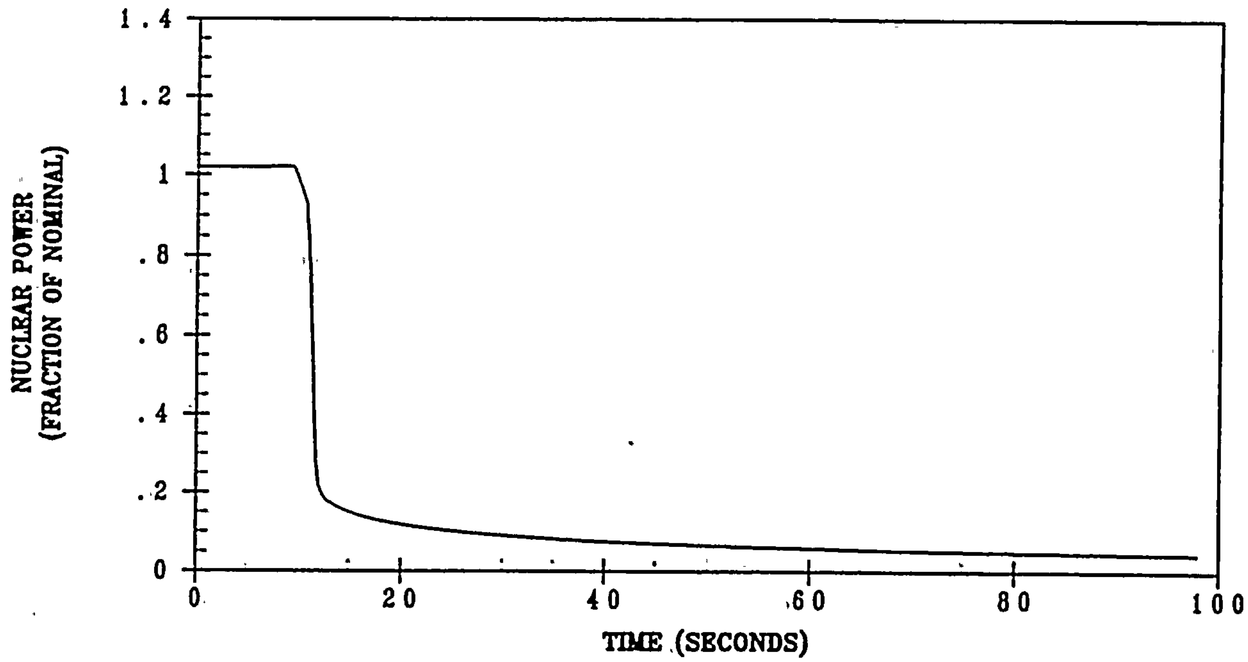


Figure 3.2.9-6 Total Loss of External Electrical Load with Pressure Control, Maximum Reactivity Feedback



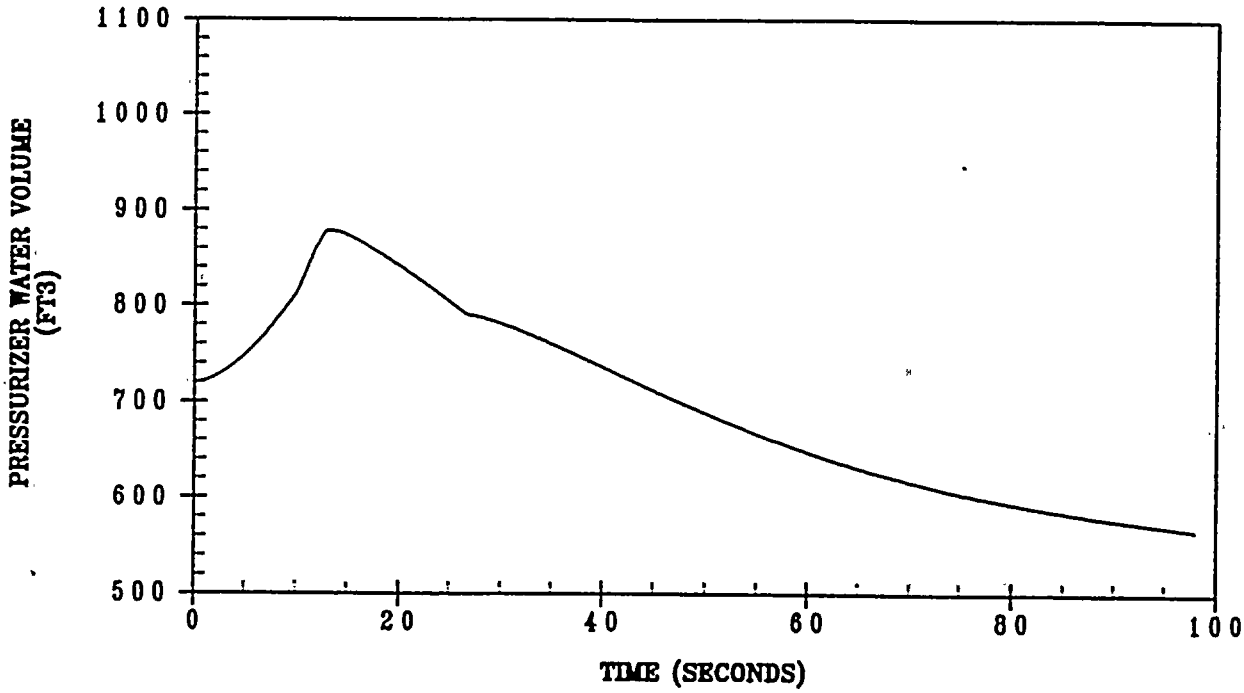


Figure 3.2.9-7 Total Loss of External Electrical Load without Pressure Control, Minimum Reactivity Feedback

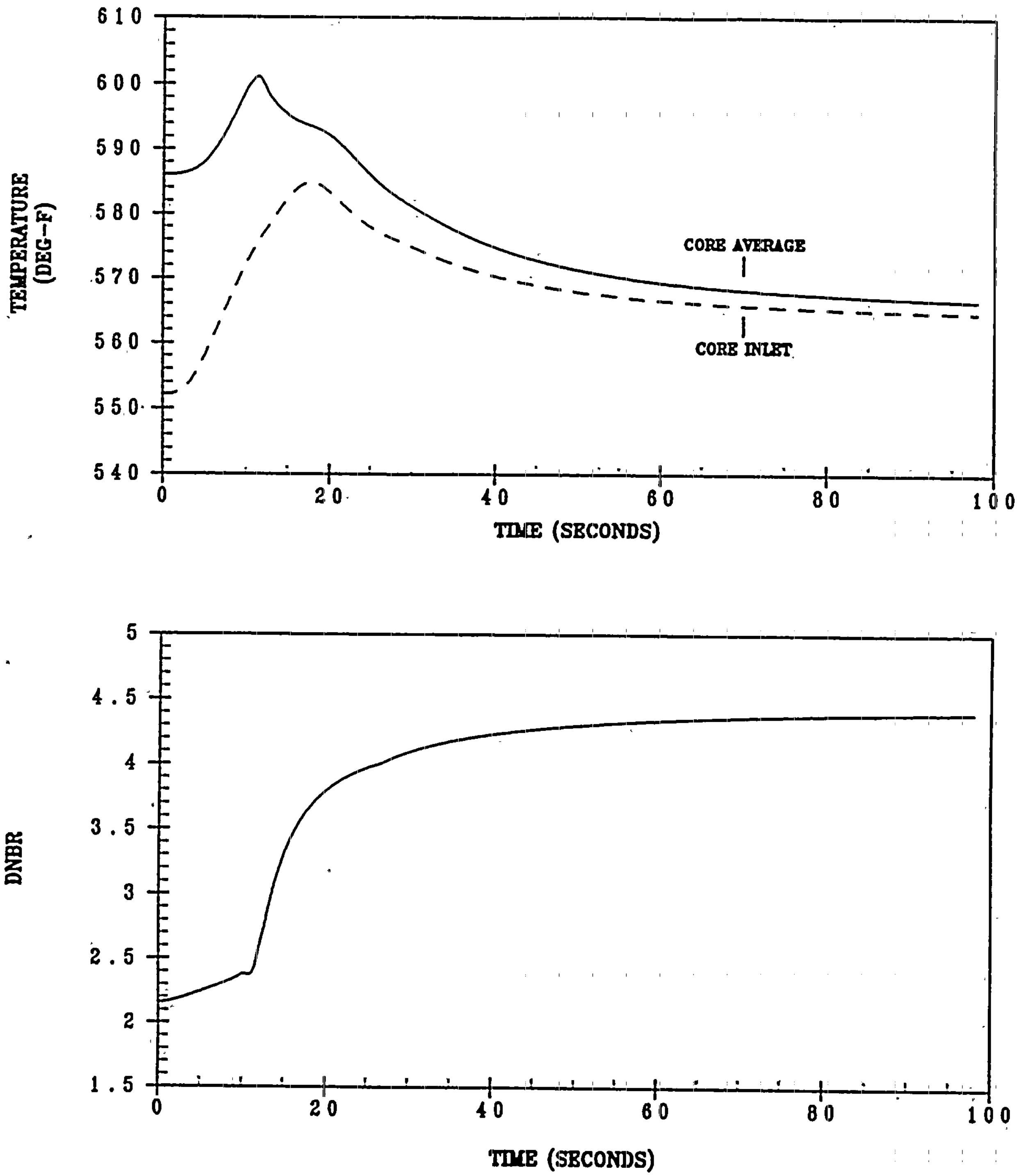


Figure 3.2.9-8 Total Loss of External Electrical Load without Pressure Control, Minimum Reactivity Feedback

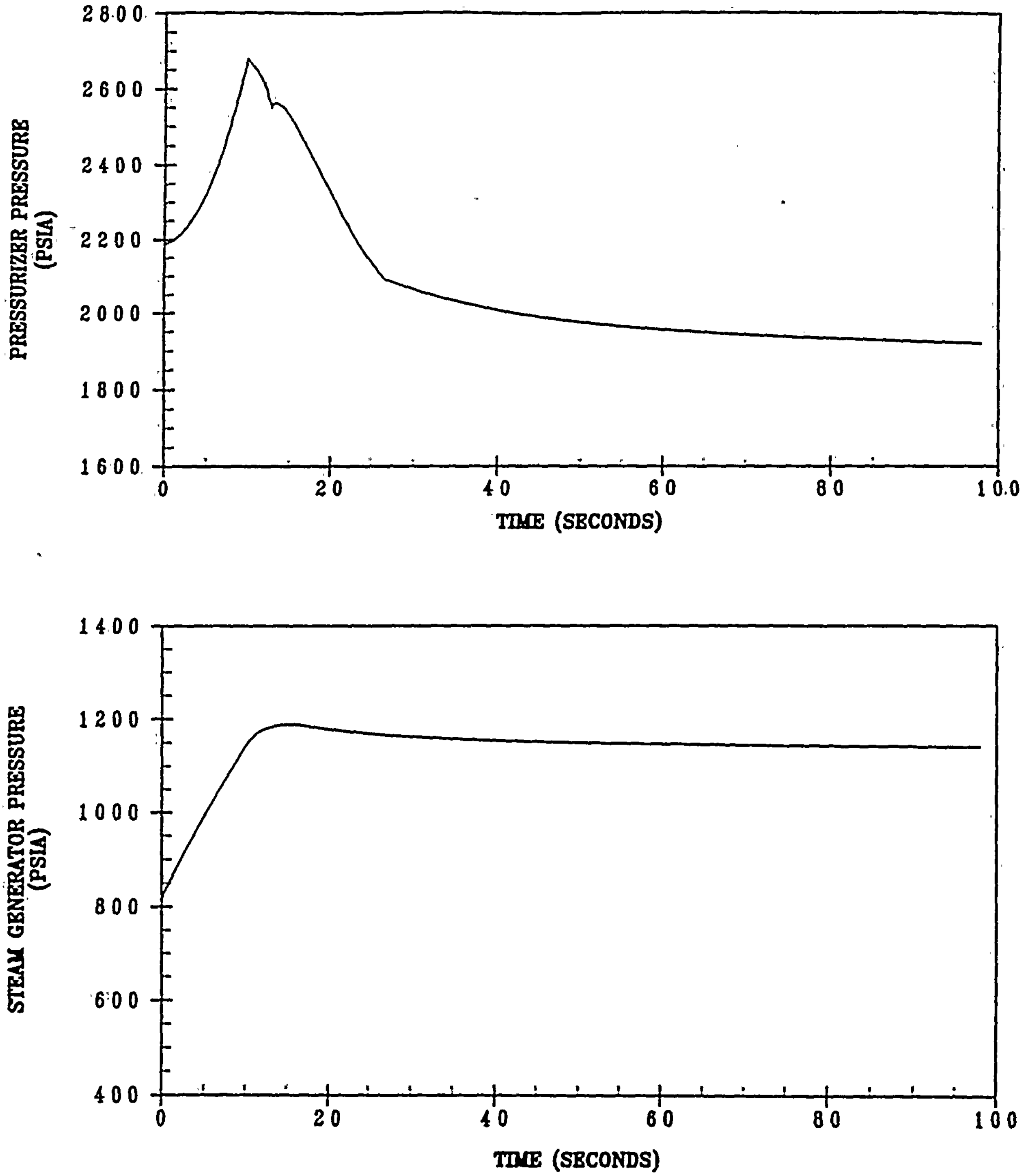


Figure 3.2.9-9 Total Loss of External Electrical Load without Pressure Control, Minimum Reactivity Feedback

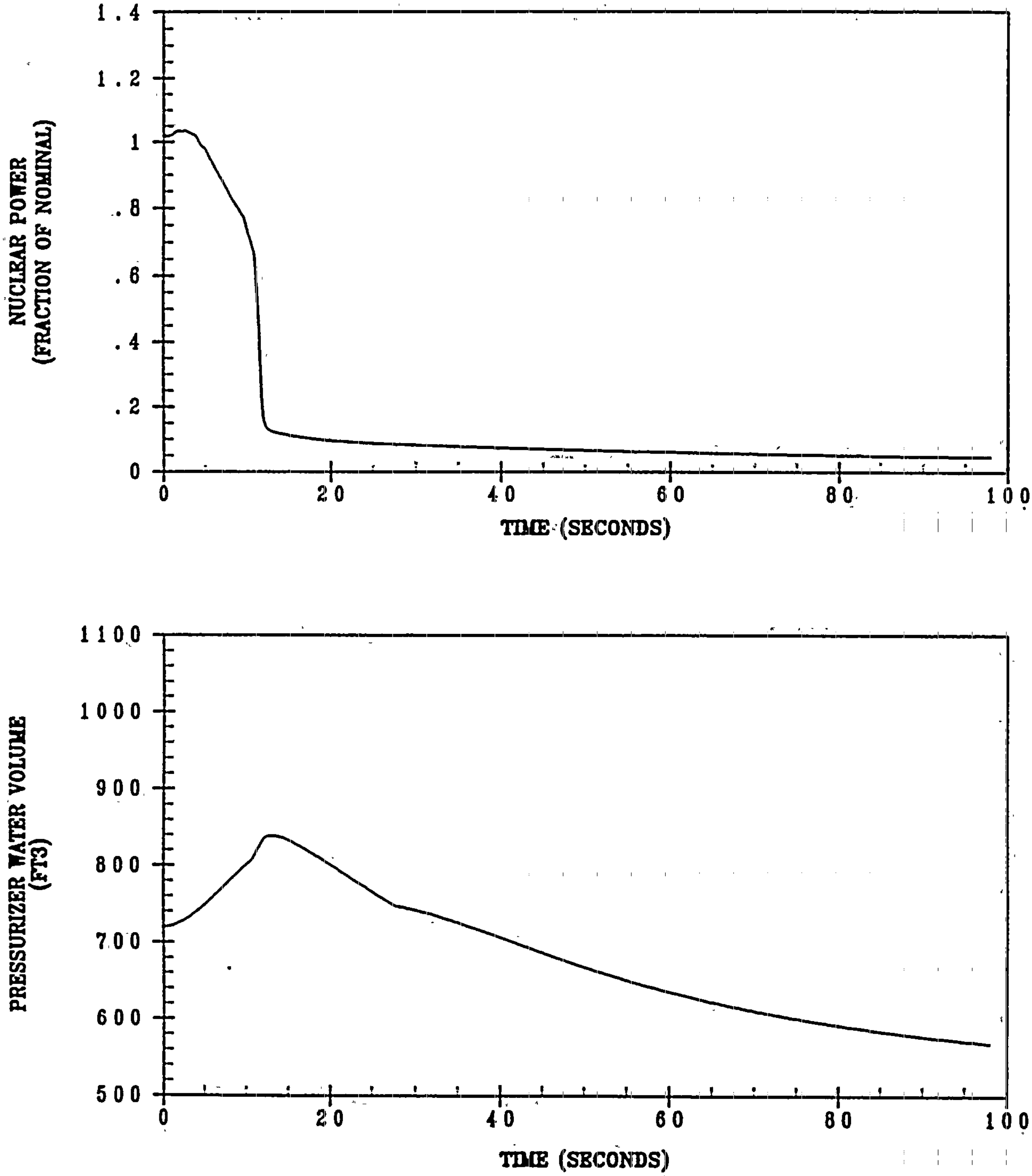
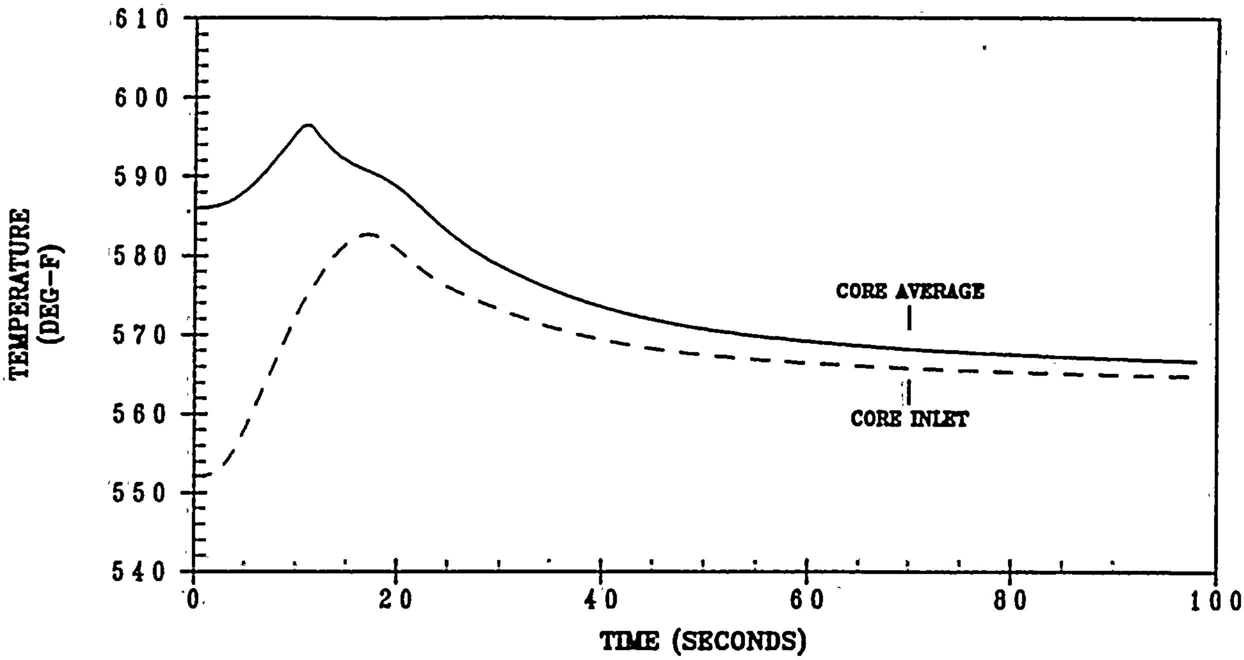


Figure 3.2.9-10 Total Loss of External Electrical Load without Pressure Control, Maximum Reactivity Feedback



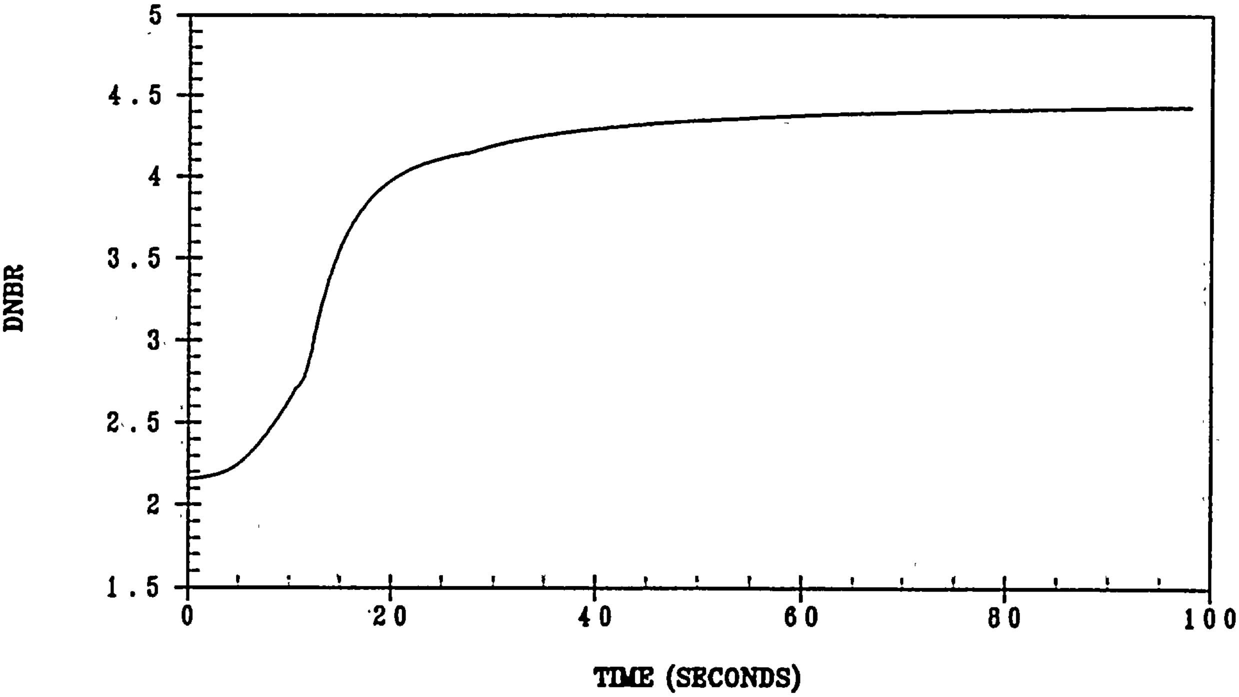


Figure 3.2.9-11 Total Loss of External Electrical Load without Pressure Control, Maximum Reactivity Feedback

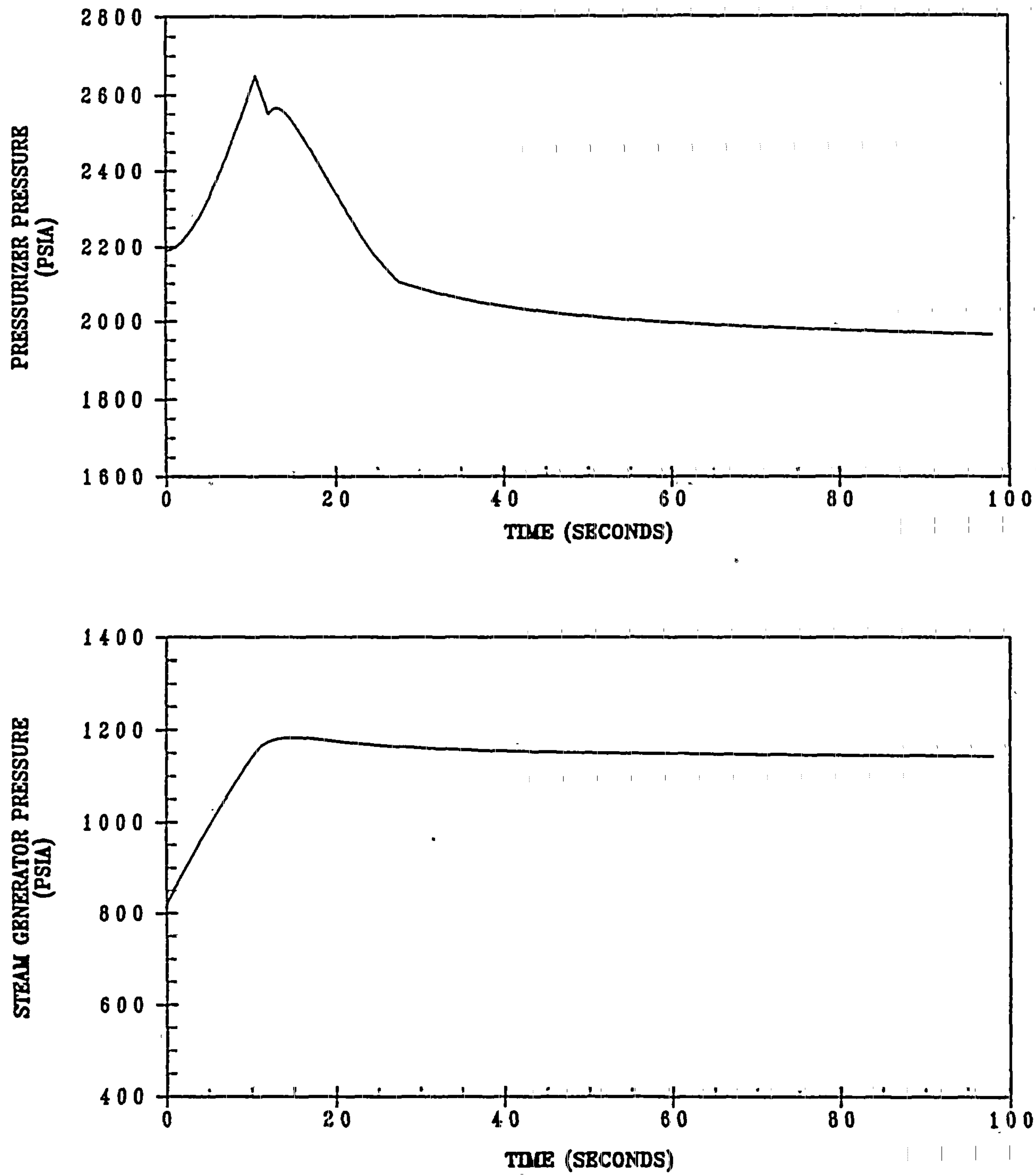


Figure 3.2.9-12 Total Loss of External Electrical Load without Pressure Control, Maximum Reactivity Feedback

3.2.10 Loss of Normal Feedwater

3.2.10.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this accident, core damage would possibly occur as a result of the loss of heat sink while at power. If an alternative supply of feedwater is not supplied to the plant, residual heat following a reactor trip may heat the primary system water to the point where water relief from the pressurizer could occur. A significant loss of water from the RCS could lead to core uncovery and subsequent core damage. However, since a reactor trip occurs well before the steam generator heat transfer capability is reduced, the primary system conditions never approach those that would result in a DNB condition.

The loss of normal feedwater that occurs as a result of the loss of AC power is discussed in Section 3.2.11.

The following events occur following the reactor trip for the loss of normal feedwater as a result of main feedwater pump failures or valve malfunctions:

A. As the steam system pressure rises following the trip, the steam system atmospheric dump valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the atmospheric dump valves are not available, the self-actuated main steam safety valves will lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

B. As the no-load temperature is approached, the steam system atmospheric dump valves (or the self-actuated safety valves, if the atmospheric dump valves are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

The following provide the necessary protection against core damage in the event of a loss of normal feedwater.

A. Reactor trip on low-low water level in any steam generator.

B. Reactor trip on steam flow-feedwater flow mismatch coincident with low steam generator water level in any loop.

C. Three turbine-driven auxiliary feedwater (AFW) pumps, shared by Turkey Point Units 3 and 4, start automatically on any of the following:

   1. Low-low water level in any steam generator

2. Any safety injection signal
3. Loss of offsite power (automatic transfer to diesel generators)
4. Loss of voltage to A and B 4.16 kV bus
5. Trip of both unit main feedwater pumps
6. Manual actuation
7. AMSAC (for ATWS)

The analysis shows that following a loss of normal feedwater, the AFW System is capable of removing the stored and residual heat thus preventing overpressurization of the RCS, overpressurization of the secondary side, water relief from the pressurizer, and uncovery of the reactor core.

3.2.10.2 Input Parameters and Assumptions

The following assumptions are made in the analysis.

A. The plant is initially operating at 102% of the NSSS power of 2311.4 MWt, which includes a maximum reactor coolant pump heat of 11.4 MWt. The RCPs are assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the Thermal Design value. Although not assumed in the analysis, the reactor coolant pumps could be manually tripped at some later time in the transient to reduce the heat addition to the RCS caused by the operation of the pumps.

B. The initial reactor vessel average coolant temperature is conservatively assumed to be 6.0°F higher than the nominal value (high) to account for the temperature uncertainty on nominal temperature. The initial pressurizer pressure uncertainty is 60 psi and is conservatively subtracted from the nominal pressure value.

C. Reactor trip occurs on steam generator low-low water level at 4.0% of narrow range span.

D. It is assumed that only one AFW pump is available to supply a minimum of 310 gpm to three steam generators, 120 seconds following a low-low steam generator water level signal.

E. The pressurizer sprays and PORVs are assumed operable. This maximizes the pressurizer water volume. If these control systems did not operate, the pressurizer safety valves would prevent the RCS pressure from exceeding the RCS design pressure limit during this transient.

F. Secondary system steam relief is achieved through the self-actuated main steam safety valves. Note that steam relief will, in fact, be through the steam generator atmospheric dump valves or condenser dump valves for most cases of loss of normal feedwater. However, since these valves are not safety grade, they have been assumed unavailable.

G. The main steam safety valves are assumed to lift and be full open at 6% above their respective setpoint pressures. This 6% includes 3% each for safety valve setpoint uncertainty and accumulation.

H. The AFW line purge volume is conservatively assumed to be the maximum average value of the two Units.

I. Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 2). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

3.2.10.3 Description of Analysis

A detailed analysis using the LOFTRAN (Reference 1) computer code is performed in order to determine the plant transient conditions following a loss of normal feedwater. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

3.2.10.4 Acceptance Criteria

Based on its frequency of occurrence, the loss of normal feedwater accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event:

- The critical heat flux shall not be exceeded. This is typically demonstrated by precluding Departure from Nucleate Boiling (DNB).

- Pressure in the reactor coolant and main steam systems shall be maintained below 110% of the design pressures.

- The pressurizer should not reach a water-solid condition.

3.2.10.5 Results

The calculated sequence of events for this accident is listed in Table 3.2.10-1. Figures 3.2.10-1 and 3.2.10-2 show the significant plant parameters following a loss of normal feedwater with the assumptions listed in Section 3.2.10.2.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to reduction of the steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Two minutes following the initiation of the low-low level trip, the turbine-driven AFW pumps automatically start; consequently, reducing the rate at which the steam generator water level is decreasing.

The capacity of one AFW pump is such that the water level in the steam generator will not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without the pressurizer reaching a water solid condition and precluding any water relief through the RCS pressurizer relief or safety valves. From Figure 3.2.10-1 it can be seen that at no time does the pressurizer go water solid. If the auxiliary feedwater delivered is greater than that of one AFW pump, or the initial reactor power is less than 102% of the NSSS power, or the steam generator water level in one or more steam generators is above the conservatively low 4% narrow range span level assumed for the low-low steam generator setpoint, the results for this transient will be bounded by the analysis presented.

3.2.10.6 Conclusions

With respect to DNB, the loss of normal feedwater event is bounded by the loss of load/turbine trip analysis (Section 3.2.9). The only difference between these two events is the turbine trip which is not assumed in a loss of normal feedwater until after the reactor trip. This allows for continued heat removal (steam flow), which is a benefit, until rod motion occurs following reactor trip. The loss of load/turbine trip analysis is described in Section 3.2.9. The results of the analysis show:

- Pressure in the reactor coolant and main steam system is maintained below 110% of the design pressure.

- The pressurizer does not reach a water solid condition.

Therefore the loss of normal feedwater event does not adversely affect the core, the RCS, or the main steam system since the AFW capacity is such that all applicable acceptance criteria are met.

3.2.10.7 References

1. Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

2. ANSI/ANS-5.1 - 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

Table 3.2.10-1

Time Sequence Of Events For Loss Of Normal Feedwater Flow

Event	Time (seconds)
Main feedwater flow stops	10
Low-low steam generator water level reactor trip setpoint reached	62.4
Rods begin to drop	64.4
Flow from one turbine driven AFW pump is initiated	182.4
Feedwater lines are purged and cold AFW is delivered to three Steam Generators.	746.0
Peak water level in pressurizer occurs	2956.0
Core stored and RCP heat decreases to AFW heat removal capacity	~3000

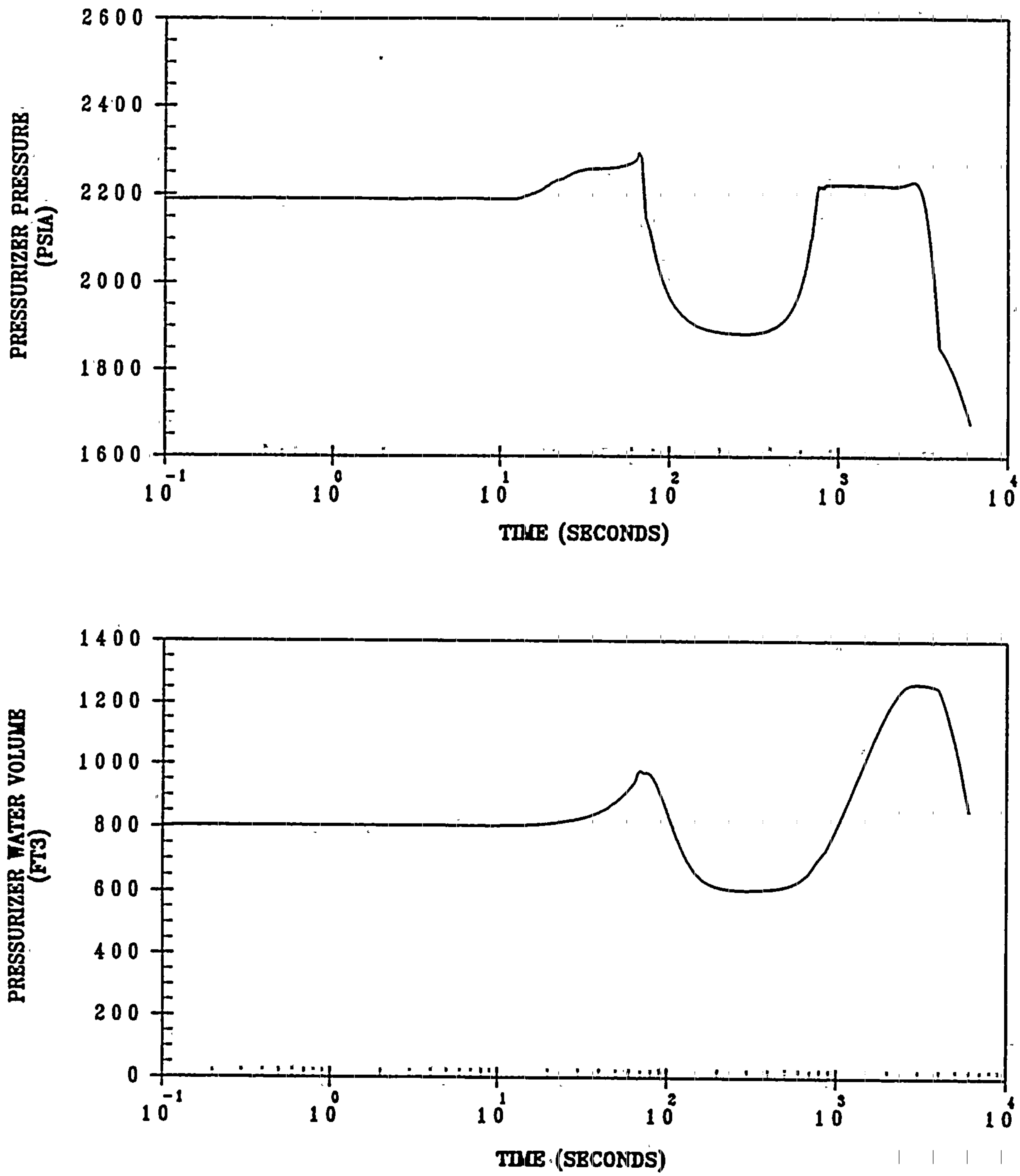


Figure 3.2.10-1 Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater

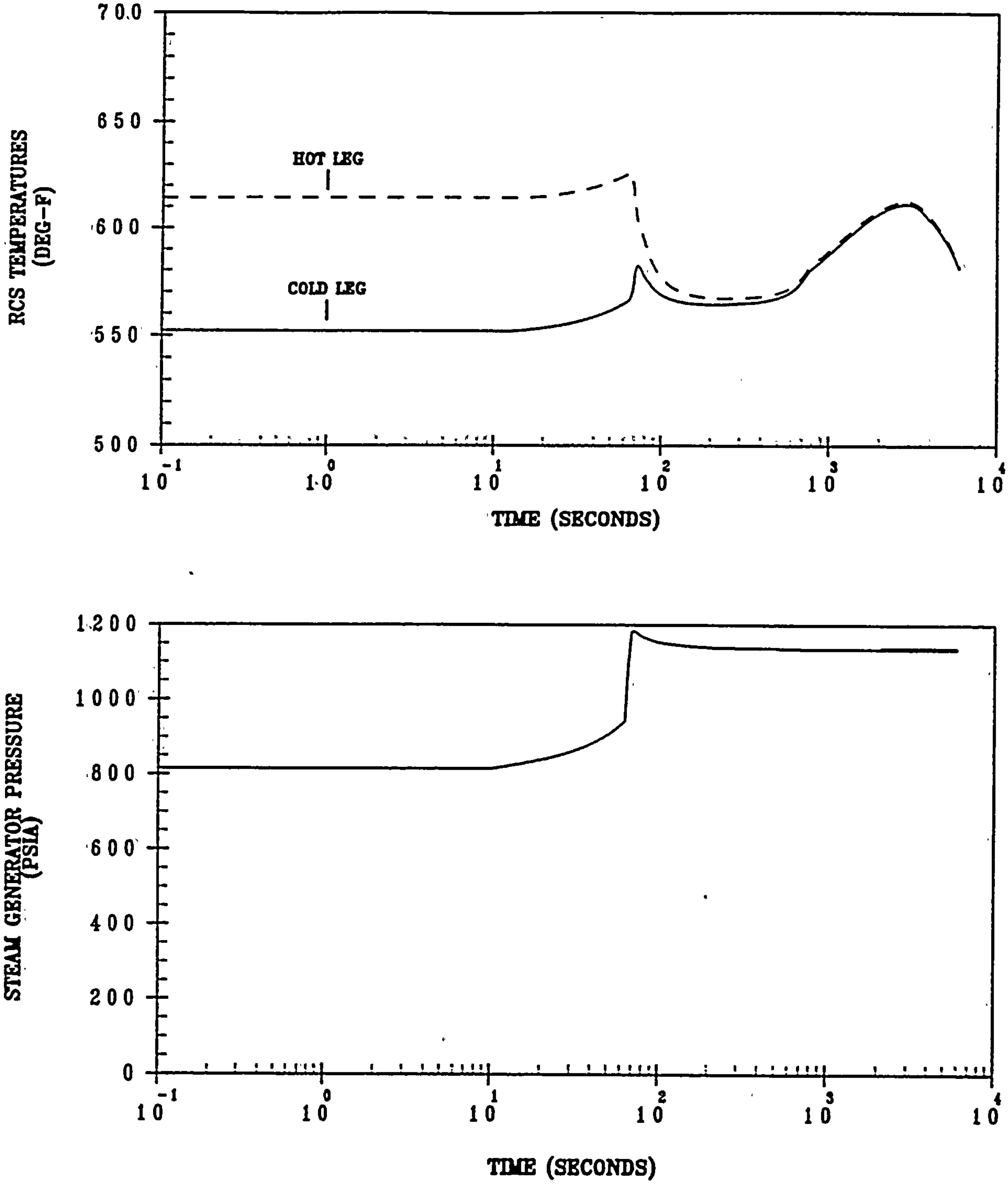


Figure 3.2.10-2 Loop Temperatures and Steam Generator Pressure for Loss of Normal Feedwater

3.2.11 Loss of Non-emergency AC Power to the Plant Auxiliaries

3.2.11.1 Identification of Causes and Accident Description

A loss of non-emergency AC power will result in a loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip or by a loss of the onsite AC distribution system. The events following a loss of AC power with turbine and reactor trips are described in the sequence listed below.

A. The plant vital instruments are supplied by emergency power sources.

B. As the steam system pressure rises following the trip, the steam system atmospheric dump valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the atmospheric dump valves are not available, the self-actuated main steam safety valves will lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

C. As the no-load temperature is approached, the steam system atmospheric dump valves (or the self-actuated safety valves, if the atmospheric dump valves are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

D. Both emergency diesel generators associated with the unit will automatically start following the loss of voltage to the A and B 4.16 kv buses of that unit. At the same time, these buses will be isolated from their normal supply and their motor supply and feed breakers will be opened. The breaker from the emergency diesel generator to its associated 4.16 kv bus will close energizing the buses. Equipment will be sequentially loaded onto the 4.16 kv buses. Load centers and motor control centers will be energized as controlled by the load sequencers. All required additional manual loads will be powered by the emergency diesel generators as required by procedures.

The following provide the necessary protection against core damage in the event of a loss of non-emergency AC power.

A. Reactor trip on low-low water level in any steam generator

B. Reactor trip on steam flow-feedwater flow mismatch coincident with low steam generator water level in any loop

C. Three turbine-driven auxiliary feedwater (AFW) pumps, shared by Turkey Point Units 3 and 4, start automatically on any of the following:

1. Low-low water level in any steam generator
2. Any safety injection signal
3. Loss of offsite power (automatic transfer to diesel generators)
4. Loss of voltage to A and B 4.16 kv bus
5. Trip of both unit main feedwater pumps
6. Manual actuation.
7. AMSAC (for ATWS)

Following the loss of power to the reactor coolant pumps (RCPs), coolant flow is necessary for core cooling and the removal of residual and decay heat. Following the RCP coastdown due to the loss of AC power, the natural circulation capability of the RCS will remove decay heat from the core, aided by the AFW flow in the secondary system. Therefore, the analysis for this event is performed to demonstrate that the resultant natural circulation flow in the RCS in conjunction with the AFW flow is sufficient to remove decay heat from the core.

Turkey Point Units 3 and 4 share common electrical and AFW systems. Thus, a loss of non-emergency AC Power to the plant auxiliaries could simultaneously affect both units. The AFW system would then be required to provide flow to both units.

The worst single failure that may occur in the AFW system would result in the availability of only one of the three turbine driven AFW pumps. For this condition, the flow from the one AFW pump could be as low as 233.4 gpm to one of the units until the operator takes actions from the control board to realign the flow split to the units.

The analysis is performed for one unit, conservatively bounding both units.

3.2.11.2 Input Parameters and Assumptions

The major assumptions used in this analysis are identical to those used in the loss of normal feedwater analysis (Section 3.2.10) with the following exceptions.

A. Loss of AC power is assumed to occur at the time of reactor trip on low-low SG water level. No credit is taken for the immediate insertion of the control rods as a result of the loss of AC power to the station auxiliaries.

B. Power is assumed to be lost to the RCPs. To maximize the amount of stored energy in the RCS, the power to the RCPs is not assumed to be lost until after the start of rod motion.

C. A heat transfer coefficient in the steam generators associated with RCS natural circulation is assumed following the RCP coastdown.

D. The RCS flow coastdown is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, the as-built pump characteristics and conservative estimates of system pressure losses.

E. The worst single failure assumed to occur is in the AFW system. This results in the availability of only one AFW pump supplying minimum flow to three steam generators, 95 seconds following a low-low steam generator water level signal. The AFW flow is less than that assumed for a loss of normal feedwater because Turkey Point Units 3 and 4 have a shared AFW system and a loss of AC power may occur simultaneously at both units.

3.2.11.3 Description of Analysis

A detailed analysis using the LOFTRAN (Reference 1) computer code is performed in order to determine the plant transient conditions following a loss of non-emergency AC power. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

3.2.11.4 Acceptance Criteria

Based on its frequency of occurrence, the loss of non-emergency AC power incident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event.

The critical heat flux should not be exceeded. This is typically demonstrated by precluding Departure from Nucleate Boiling (DNB). With respect to DNB, the loss of non-emergency AC power event is bounded by the complete loss of flow analysis since the coastdown in the loss of non-emergency AC power event does not occur until after reactor trip which is less limiting. Hence, the loss of non-emergency AC power event is bounded by the complete loss of flow analysis described in Section 3.2.8.1.

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures.

The pressurizer should not reach a water-solid condition.

3.2.11.5 Results

Figures 3.2.11-1 and 3.2.11-2 show plant parameters following a loss of non-emergency power with the assumptions listed in Section 3.2.11.2. The calculated sequence of events for this accident is listed in Table 3.2.11-1.

The first few seconds after the loss of non-emergency AC power to the RCPs, the flow transient for a loss of non-emergency AC power event will closely resemble a simulation of the complete loss of flow incident, where core damage due to rapidly increasing core temperatures is prevented by the reactor trip, which, for a loss of non-emergency AC power event, is on a low-low steam generator water level signal. After the reactor trip, stored and residual heat must be removed to prevent damage to the core and the reactor coolant and main steam systems. The LOFTRAN code results show that the natural circulation and AFW flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The capacity of the turbine-driven AFW pump is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to establish enough natural circulation flow in order to dissipate core residual heat without water release through the RCS relief or safety valves. Figure 3.2.11-1 illustrates that the pressurizer never reaches a water solid condition. Hence, no water relief from the pressurizer occurs.

3.2.11.6 Conclusions

Results of the analysis show that, for the loss of non-emergency power to the station auxiliaries event, all applicable safety criteria are met. The DNBR transient is bounded by the complete loss of flow event (Section 3.2.8.1) and remains above the safety analysis limit value. Assuming the worst single failure occurs in the AFW system, the available AFW capacity and the natural circulation capability of the RCS following reactor coolant pump coastdown is sufficient to prevent the pressurizer from reaching a water solid condition such that sufficient long-term heat removal capability exists to prevent fuel or clad damage. Pressure in the reactor coolant and main steam systems is maintained below 110% of the design pressures.

3.2.11.7 Reference

1. Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

Table 3.2.11-1

Time Sequence Of Events For Loss Of Non-Emergency AC Power

Event	Time (seconds)
Main feedwater flow stops	10
Low-low steam generator water level reactor trip setpoint reached	62.4
Rods begin to drop	64.4
Reactor coolant pumps begin to coastdown	66.4
Flow from one turbine driven AFW pump is initiated	157.4
Feedwater lines are purged and cold AFW is delivered to three Steam Generators.	906.0
Core stored and residual heat decreases to AFW heat removal capacity	~3500
Peak water level in pressurizer occurs	3596.0

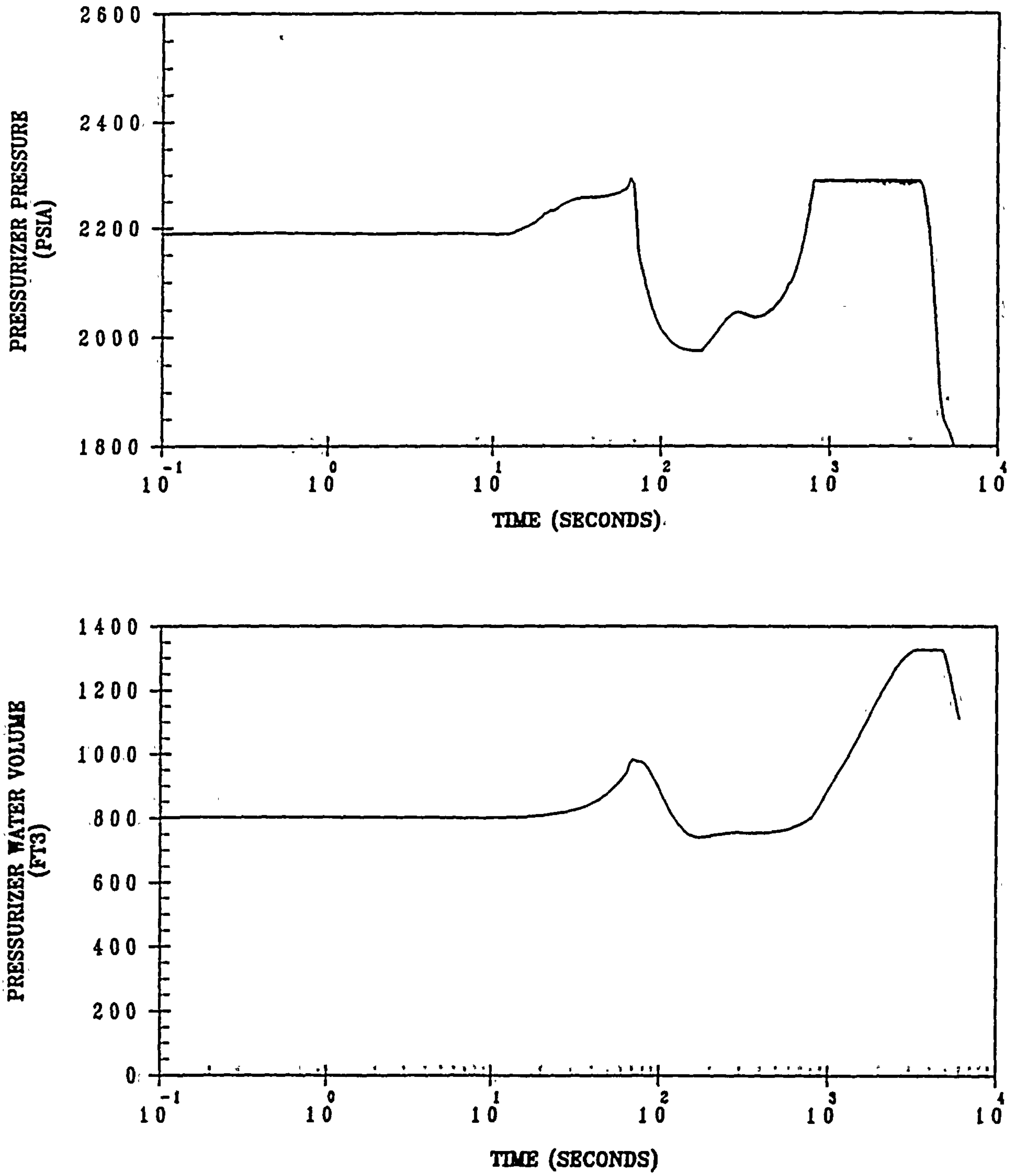


Figure 3.2.11-1 Pressurizer Pressure and Water Volume Transients for Loss of Offsite Power

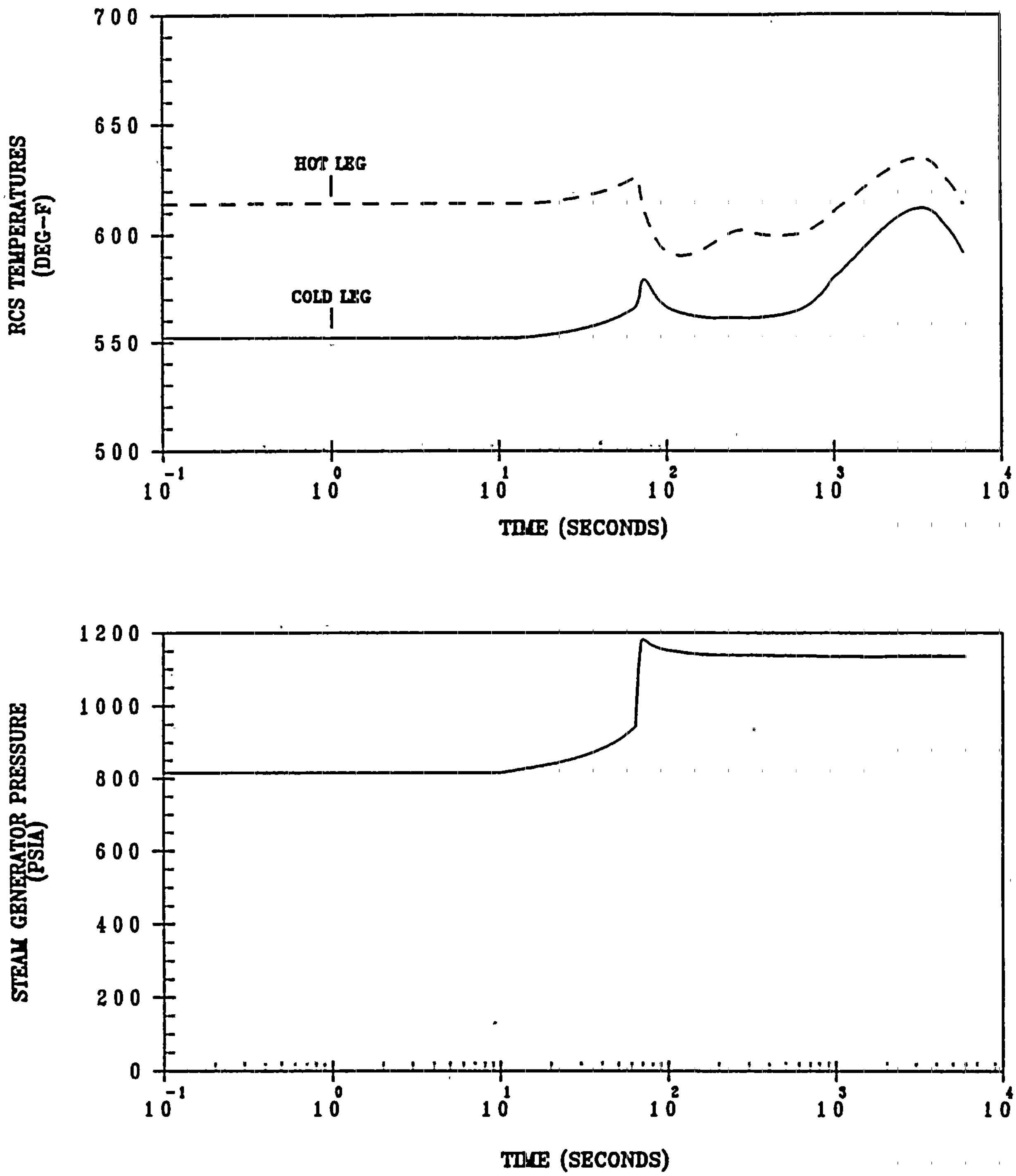


Figure 3.2.11-2 Loop Temperatures and Steam Generator Pressure for Loss of Offsite Power

3.2.12 Fuel Handling Accident Radiological Consequences

3.2.12.1 Introduction

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed for the accident occurring either inside containment or in the spent fuel pool. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the spent fuel pool area ventilation systems. This section describes the assumptions and analyses performed to determine the amount of activity released and the offsite doses resulting from this release.

3.2.12.2 Input Parameters and Assumptions

The analytical methods and assumptions used to determine the offsite doses due to a fuel handling accident (FHA) are primarily those outlined in References 3 and 4. Also addressed are the uprated power level of 2346 MWt, and a 12% I-131 gap fraction (20% increase over recommendation of Reference 3) for high burnup fuel (References 1 and 2).

Two cases are analyzed with respect to the amount of damage suffered by the dropped assembly. For the first case, it is assumed that all of the fuel rods in the equivalent of one assembly are damaged to the extent that all their gap activity is released. In the second case, only the fuel rods in one row of the assembly (i.e., 15 fuel rods) are damaged sufficiently to cause their gap activity to be released.

Since, per Technical Specifications, the reactor has to be subcritical for 100 hours before fuel is moved, 100 hours of radioactive decay is assumed in the analysis. The Technical Specifications require at least 23 feet of water to be above the reactor vessel flange while in refueling (mode 6). This is consistent with the guidance contained in Regulatory Guide 1.25 (Reference 3). With this water depth, decontamination factors (DF) of 133 for elemental iodine and 1 for methyl iodine are used for pool scrubbing (Reference 3). The iodine activity in the fuel rod gap is assumed to be 99.75% elemental and 0.25% methyl (Reference 3). The resulting overall pool scrubbing DF for iodine is 100.

All of the noble gas released from the damaged assembly is assumed to be released from the pool water (i.e., the pool scrubbing DF is 1) (Reference 3).

A conservatively high radial peaking factor of 1.7 is assumed for the damaged assembly.

No credit is taken for filtration of iodine for either the FHA inside containment or the FHA in the spent fuel pool. Although the containment purge will be automatically isolated on a containment high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be immediately released to the outside atmosphere.

The major assumptions and parameters used in the analysis are itemized in Table 3.2.12-1. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 3.2.12-2. Since the assumptions and parameters for a FHA inside containment are identical to those for a FHA in the spent fuel pool, the offsite doses are the same regardless of the location of the accident.

3.2.12.3 Description of Analyses Performed

The activity releases and offsite doses are determined for both a FHA inside containment and a FHA in the spent fuel pool. Offsite doses are calculated for both one damaged assembly and one damaged row of rods.

3.2.12.4 Acceptance Criteria

The dose limits for a FHA are "well within" the guideline values of 10 CFR 100, or 75 rem thyroid and 6 rem γ-body.

3.2.12.5 Results

The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary are as follows:

Damaged Fuel	EB (0-2 Hr)	LPZ (0-2 Hr)
1. Thyroid		
One Assembly	3.3 E+1	3.2 EO
One Row	2.4 EO	2.4 E-1
2. γ-Body		
One Assembly	9.3 E-2	9.0 E-3
One Row	6.8 E-3	6.6 E-4

3.2.12.6 Conclusions

The offsite thyroid and γ-body doses due to the FHA are within the acceptance criteria in Section 3.2.12.4.

3.2.12.7 References

1. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", D. A. Baker, et. al., February 1988.

2. Federal Register/Vol. 53, No. 39/ February 29, 1988/pages 6040 through 6043.

3. USAEC Safety Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", 3/23/72.

Table 3.2.12-1

Assumptions Used For Fuel Handling Accident Dose Analysis

Power	2346 MWt
Radial Peaking Factor	1.7
Damaged Fuel	
Case 1	1 Fuel Assembly
Case 2	15 Rods
Fuel Rod Gap Fractions	0.10 for iodines and noble gases, except 0.12 for I-131 0.30 for Kr-85
Percent of Gap Activity Released	100%
Pool Decontamination Factors	
Elemental Iodine	133
Methyl Iodine	1
Noble Gas	1
Iodine Species in Fuel Rod Gap	
Elemental Iodine	99.75%
Methyl Iodine	0.25%
Minimum Water Depth Above the Reactor Vessel Flange	23 feet
Filter Efficiency	no filtration assumed
Containment Isolation	no containment isolation

Table 3.2.12-2

Dose Conversion Factors, Breathing Rates and Atmospheric Dispersion Factors

Isotope	Dose Conversion Factor [1] (rem/curie)
I-131	1.07E6
I-132	6.29E3
I-133	1.81E5
I-134	1.07E3
I-135	3.14E4

Time Period (hr)	Breathing Rate [2] (m^3/sec)
0-8	3.47E-4

	Atmospheric Dispersion Factors, (sec/m^3)
Exclusion Boundary (0-2 hr)	1.54E-4
Low Population Zone	
0-2 hr	1.5E-5

[1] ICRP Publication 30
[2] Regulatory Guide 1.4

3.2.13 Dropped Spent Fuel Transfer Cask Radiological Consequences

3.2.13.1 Introduction

It is assumed that a spent fuel transfer cask is dropped into the spent fuel pool and damages fuel assemblies stored there. Activity released from the damaged assemblies is released to the outside atmosphere through the spent fuel pool area ventilation systems. This section describes the assumptions and analyses performed to determine the amount of activity released and the offsite doses resulting from this release.

3.2.13.2 Input Parameters and Assumptions

The input parameters and assumptions for the cask drop dose analysis are the same as those for the fuel handling accident (Section 3.2.12) with the following exceptions:

The offsite doses are determined on a per core basis. Thus, the base case doses are for 157 fuel assemblies (i.e., the total number of fuel assemblies in one core) being damaged by the dropped cask. It is assumed that the gap activity in every fuel rod in each damaged fuel assembly is released.

Since, the Technical Specifications prevent cask movement into the spent fuel pool until all the spent fuel in the pool has decayed for a minimum of 1525 hours, 1525 hours of radioactive decay is assumed in the analysis.

A radial peaking factor of 1.0 is used for the fuel assemblies stored in the spent fuel pool.

The major assumptions and parameters used in the analysis are itemized in Table 3.2.13-1.

3.2.13.3 Description of Analyses Performed

The base case activity releases and offsite doses are determined for 157 fuel assemblies in the spent fuel pool being damaged by the dropped cask. This is equivalent to a full core.

3.2.13.4 Acceptance Criteria

The dose limit assumed for a dropped cask is "well within" the guideline values of 10 CFR 100, or 75 rem thyroid and 6 rem γ-body. This is the same acceptance criteria assumed for the fuel handling accident.

3.2.13.5 Results

The base case offsite thyroid and whole body doses due to the dropped cask assuming 157 fuel assemblies being damaged are within the acceptance criteria in Section 3.2.13.4. The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary are as follows:

	EB (0-2 Hr)	LPZ (0-2 Hr)
Thyroid	1.77 E1	1.73 EO
γ-Body	2.42 E-2	2.36 E-3

3.2.13.6 Conclusions

With the number of fuel assemblies equivalent to one core damaged, the doses are well within the acceptance criteria. The theoretical limit as to the number of fuel assemblies that would have to be damaged without exceeding the acceptance criteria is approximately 4.0 cores (or 628 fuel assemblies). This amount of damage due to a dropped cask is not physically possible.

Table 3.2.13-1

Assumptions Used For Dropped Cask Dose Analysis

Power	2346 MWt
Radial Peaking Factor	1.0
Damaged Fuel (Base Case)	157 Fuel Assemblies
Fuel Rod Gap Fractions	0.10 for iodines and noble gases, except 0.12 for I-131 and 0.30 for Kr-85
Percent of Gap Activity Released	100%
Pool Decontamination Factors	
Elemental Iodine	133
Methyl Iodine	1
Noble Gas	1
Iodine Species in Fuel Rod Gap	
Elemental Iodine	99.75%
Methyl Iodine	0.25%
Minimum Water Depth Above the Reactor Vessel Flange	23 feet
Filter Efficiency	no filtration assumed

3.2.14 Volume Control Tank Rupture Radiological Consequences

3.2.14.1 Introduction

The volume control tank (VCT) is assumed to rupture and release its noble gas contents directly to the outside atmosphere. This section describes the assumptions and analyses performed to determine the amount of activity released and the offsite doses.

3.2.14.2 Input Parameters and Assumptions

The noble gas activity in the VCT is based on a 1% fuel defect level and a liquid level of 40%.

The major assumptions and parameters used in the analysis are itemized in Table 3.2.14-1. The average gamma energies used in the determination of the equivalent curies of Xe-133 in the VCT are given in Table 3.2.14-2.

3.2.14.3 Description of Evaluation Performed

The equivalent curies of Xe-133 in the VCT are calculated.

3.2.14.4 Acceptance Criteria

The dose limit for a radioactive release due to a waste gas system failure is 0.5 rem γ-body (Reference 1).

3.2.14.5 Results

There are 32,330 equivalent curies of Xe-133 released from the VCT. The offsite γ-body doses (rem) due to the VCT rupture are: EB (0-2 Hr) = 3.8E-2 and LPZ (0-2 Hr) = 3.6E-3.

3.2.14.6 Conclusions

The offsite γ-body doses due to the VCT rupture are well below the acceptance criteria.

3.2.14.7 Reference

1. NUREG-0800, Standard Review Plan 11.3, Gaseous Waste Managment Systems, Branch Technical Position ETSB 11.5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure", Rev. 0, July, 1981.

TABLE 3.2.14-1

Assumptions Used For Volume Control Tank Rupture Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity	1% Fuel Defect Level
VCT Liquid Level	40%
VCT Liquid Volume	120 ft^3
VCT Vapor Volume	180 ft^3

TABLE 3.2.14-2

Noble Gas Average Gamma Energy

Nuclide	$\bar{E}\gamma$, (Mev/Dis)
Kr-85m	0.16
Kr-85	0.0023
Kr-87	0.79
Kr-88	2.2
Xe-131m	0.0029
Xe-133m	0.02
Xe-133	0.03
Xe-135m	0.43
Xe-135	0.25
Xe-138	1.2

3.2.15 Gas Decay Tank Rupture Radiological Consequences

3.2.15.1 Introduction

A gas decay tank is assumed to rupture and release its noble gas contents directly to the outside atmosphere. This section describes the assumptions and analyses performed to determine the amount of activity released and the corresponding offsite doses.

3.2.15.2 Input Parameters and Assumptions

The noble gas activity in a gas decay tank is based on a 1% fuel defect level and a letdown flow rate of 120 gpm. The inventory of noble gas activity is assumed to be stripped from the RCS during a cold shutdown and placed in a single gas decay tank. There is negligible iodine activity in the gas decay tanks.

The major assumptions and parameters used in the analysis are itemized in Table 3.2.15-1. The noble gas average gamma energies and atmospheric dispersion factors used in the γ-body dose calculations are given in Table 3.2.15-2.

3.2.15.3 Description of Analyses Performed

The offsite γ-body doses due to the instantaneous release to atmosphere of the entire inventory of noble gas in the ruptured gas decay tank are calculated.

3.2.15.4 Acceptance Criteria

The dose limit for a radioactive release due to a waste gas system failure is 0.5 rem γ-body (Reference 1).

3.2.15.5 Results

There are 55,000 curies of equivalent Xe-133 released to the environment due to a postulated gas decay tank rupture. The resulting γ-body doses (rem) are: EB (0-2 Hr) = 6.4 E-2 and LPZ (0-2 Hr) = 6.2 E-3.

3.2.15.6 Conclusions

The offsite γ-body doses due to the gas decay tank rupture are well below the acceptance criteria.

3.2.15.7 References

1. NUREG-0800, Standard Review Plan 11.3, Gaseous Waste Management Systems, Branch Technical Position ETSB 11.5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," Rev. 0, July 1981.

Table 3.2.15-1

Assumptions Used For Gas Decay Tank Rupture Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity	1% Fuel Defect Level
Letdown Flow Rate	120 gpm
Gas Decay Tank Volume	525 ft^3

Table 3.2.15-2

Noble Gas Average Gamma Energy and Atmospheric Dispersion Factors

Nuclide	$\bar{E}\gamma$, (Mev/Dis)
Kr-85m	0.16
Kr-85	0.0023
Kr-87	0.79
Kr-88	2.2
Xe-131m	0.0029
Xe-133m	0.02
Xe-133	0.03
Xe-135m	0.43
Xe-135	0.25
Xe-138	1.2

	Atmospheric Dispersion Factors, (sec/m^3)
Exclusion Boundary (0-2 hr)	1.54E-4
Low Population Zone	
0-2 hr	1.5E-5
2-12 hr	6.5E-6
12-720 hr	2.4E-7

3.2.16 Main Steam Line Break Core Response

3.2.16.1 Identification of Causes and Accident Description

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a valve malfunction (UFSAR Section 14.2.5.1) or due to a break in pipe line (UFSAR Section 14.2.5.2). The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod is stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the injection of boric acid at the boric acid concentration from the refueling water storage tank.

3.2.16.2 Description of Analysis

The main steam line break core response events have not been reanalyzed to support the NSSS power uprating for Turkey Point Units 3 and 4; an evaluation of the UFSAR licensing basis analyses (UFSAR Sections 14.2.5.1 and 14.2.5.2) was performed instead. The events are analyzed assuming hot zero power conditions. Since the hot zero power conditions for the NSSS power uprating as well as all other key analysis assumptions have remained unchanged, the current UFSAR steam line break core response analyses remains valid. A DNB evaluation of the statepoints obtained for the most limiting steam line break core response case was performed.

3.2.16.3 Acceptance Criteria

The valve malfunction incident discussed in UFSAR Section 14.2.5.1 is classified as an ANS Condition II event. A major break in a pipe line (UFSAR Section 14.2.5.2) is classified as an ANS Condition IV event. Minor secondary system pipe breaks are classified as ANS Condition III events. All of these events are analyzed to meet Condition II criteria. The only criterion that may be challenged during this event is the one that states that the critical heat flux should not be exceeded. The evaluation shows that this criterion is met by ensuring that the minimum DNBR does not go below the limit value at any time during the transient.

3.2.16.4 Results

The evaluation of the limiting main steam line break core response statepoints indicates that the minimum DNBR stays above the safety analysis limit value at all times during this event.

3.2.16.5 Conclusions

The evaluation shows that for all of the main steam line break core response events, the DNB design basis continues to be met at the uprated power level.

3.2.17 Rupture Of A Control Rod Drive Mechanism (CRDM) - RCCA Ejection

3.2.17.1 Identification of Causes and Accident Description

This accident is defined as a mechanical failure of a control rod drive mechanism pressure housing resulting in the ejection of the rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

A. Each full-length control rod drive mechanism housing is completely assembled and shop tested at 3450 psig.

B. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed Reactor Coolant System.

C. Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress ranges specified in the ASME Code, Section III, for Class 1 components.

D. The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that the gross failure of the housing will not occur. The joints between the latch mechanism housing and rod travel housing are threaded joints and reinforced by canopy-type rod welds.

The operation of a chemical shim plant is such that the severity of an ejection accident is limited. In general, the reactor is operated with the rod cluster control assemblies inserted only far enough to permit load follow. Reactivity changes caused by the core depletion are compensated by boron dilution. Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a rod cluster control assembly ejection accident. Therefore, should a rod cluster control assembly be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur. The position of all rod cluster control assemblies is continuously indicated in the control room. An alarm will occur if a bank of rod cluster control assemblies approaches its insertion limit or if one control rod assembly deviated from its bank. There are low and low-low level insertion alarm circuits for each bank. The control rod position monitoring and alarm systems are described in Reference 1.

3.2.17.2 Input Parameters and Assumptions

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 3.2.17-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three-dimensional static methods or a synthesis of one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. The analysis assumes adverse xenon distributions to provide worst-case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to fuel densification.

Power distributions before and after ejection for a "worst case" can be found in Reference 1. During plant startup physics testing, ejected rod worths and power distributions have been measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.65 percent at beginning of life and 0.48 percent at end of life. The ejected rod accident is sensitive to β if the ejected rod worth is equal to or greater than β_{eff}, as in the zero-power transients. In order to allow for future fuel cycle flexibility, conservative estimates of β of 0.50 percent at beginning of cycle and 0.42 percent at end of cycle are used in the analysis.

Reactivity Weighting Factor

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single-channel analysis. Physics calculations have been performed for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, a one-dimensional (axial) spatial kinetics method is employed, thus axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three-dimensional analysis.

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative when compared to the actual design conditions for the plant. As discussed above, no weighting factor is applied to these results. The resulting moderator temperature coefficient is at least +7 pcm/°F at the appropriate zero- or full-power nominal average temperature for the beginning-of-life cases.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

Heat Transfer Data

The FACTRAN (Reference 2) code used to determine the hot spot transient contains standard curves of thermal conductivity versus fuel temperature. During a transient, the peak centerline fuel temperature is independent of the gap conductances during the transient. The cladding temperature is, however, strongly dependent on the gap conductance and is highest for high gap conductances. For conservatism a high gap heat transfer coefficient value of 10,000 Btu/hr-ft^2-°F has been used during transients. This value corresponds to a negligible gap resistance and a further increase would have essentially no effect on the rate of heat transfer.

Coolant Mass Flow Rates

When the core is operating at full power, all three coolant pumps will always be operating. [However, for zero power conditions, the system is conservatively assumed to be operating with two pumps.] The principal effect of operating at reduced flow is to reduce the film boiling heat transfer coefficient. This results in higher peak cladding temperatures, but does not affect the peak centerline fuel temperature. Reduced flow also lowers the critical heat flux. However, since DNB is always assumed at the hot spot, and since the heat flux rises very rapidly during the transient, this produces only second order changes in the cladding and centerline fuel temperatures. All zero power analyses for both average core and the hot spot have been conducted assuming two pumps in operation.

Trip Reactivity Insertion

The control rods are assumed to be released 0.5 seconds after reaching the power range high neutron flux trip setpoint. The delay consists of 0.2 seconds for the instrumentation to produce a signal, 0.15 seconds for the reactor trip breaker to open and 0.15 seconds for coil release. In calculating the shape of the insertion versus time curve all the rods are assumed to be dropped as a single bank from the fully withdrawn position. This means that the initial movement is through the low worth region at the extreme top of the core, which results in a conservatively slow reactivity insertion versus time curve.

Fuel Densification Effects

Fuel densification effects on rod ejection are accounted for according to the methods described in Reference 3.

Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between individual fuel rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rod toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced.

Boiling in the hot spot region will produce a net fluid flow away from that region. However, the fuel releases heat to the water slowly, and it is considered inconceivable that cross flow is sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback which leads to the conclusion that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservative and therefore not included in the following analyses.

Radiological Consequences

The major assumptions and parameters used in the radiological analysis are consistent with Reference 9 and are itemized in Table 3.2.17-3.

3.2.17.3 Description of Analysis

This section describes the models used in the analysis of the rod ejection accident. Only the initial few seconds of the power transient are discussed, since the long term considerations are the same as for a loss of coolant accident.

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation uses spatial neutron-kinetics methods to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 1.

Average Core

The spatial-kinetics computer code, TWINKLE (Reference 4) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. This analysis uses the code as a one-dimensional axial kinetics code since it allows a more-realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors. The hot spot analysis uses the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is assumed within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 5) to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB heat flux is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted to force the full-power, steady-state temperature distribution to agree with fuel heat transfer design codes.

Radiological Consequences

The control rod ejection accident considers two fission product release paths to the environment. The first is containment leakage of fission products released from the primary system to the containment atmosphere. Second is leakage of fission products from the secondary system, outside containment, due to primary-to-secondary leakage in the steam generators.

3.2.17.4 Acceptance Criteria

Due to the extremely low probability of a rod cluster control assembly ejection accident, this event is classified as an ANS Condition IV event. As such, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 6). Extensive tests of UO_2 zirconium-clad fuel rods representative of those present in pressurized water reactor-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design exhibited failure as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 7) results which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreased 10 percent with fuel burnup. The clad failure mechanism appears to be melting for unirradiated (zero burnup) rods and brittle fracture for irradiated rods. The conversion ratio of thermal to mechanical energy is also important. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

The real physical limits of this accident are that the rod ejection event and any consequential damage to either the core or the Reactor Coolant System must not prevent long-term core cooling and any offsite dose consequences must be within the guidelines of 10 CFR 100. More-specific and restrictive criteria are applied to ensure fuel dispersal in the coolant, gross lattice distortion or severe shock waves will not occur. In view of the above experimental results, and the conclusions of WCAP-7588, Rev. I-A (Reference 1) and Reference 8, the limiting criteria are:

A. Average fuel pellet enthalpy at the hot spot must be maintained below 225 cal/gm for unirradiated and 200 cal/gm for irradiated fuel,

B. Peak reactor coolant pressure must be less than that which could cause RCS stresses to exceed the faulted-condition stress limits,

C. Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of Criterion A.

D. The dose acceptance criterion for a rod ejection accident is "well within" the 10 CFR 100 guideline value, or 75 rem thyroid and 6 rem γ-body.

3.2.17.5 Results

Results are presented for the four analyzed cases which cover beginning and end-of-life at zero and full power conditions.

A. <u>Beginning of Cycle, Full Power</u>

Control bank D is assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor are conservatively calculated to be 0.35 percent ΔK and 5.48, respectively. The peak hot spot average fuel pellet enthalpy is 190 cal/gm. The peak clad average temperature is 2660°F and the peak fuel centerline temperature is 5000°F. However, fuel melting remains well below the limiting criterion of 10 percent of the pellet volume at the hot spot.

B. <u>Beginning of Cycle, Zero Power</u>

For this condition, control bank D is assumed to be fully inserted with bank C at its insertion limit. The worst ejected rod is typically located in control bank D and has a worth of 0.71 percent ΔK and a hot channel factor (F_Q) of 8.0. The peak hot spot average fuel pellet enthalpy is 116 cal/gm. The peak clad average temperature reaches 2033°F; the fuel centerline temperature is 3267°F.

C. <u>End of Cycle, Full Power</u>

Control bank D is assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors are conservatively calculated to be 0.30 percent ΔK and 5.52 respectively. The peak hot spot average fuel pellet enthalpy is 147 cal/gm. This results in a peak clad average temperature of 2072°F and a peak fuel centerline temperature of 4508°F.

D. End of Cycle. Zero Power

The ejected rod worth and hot channel factor for this case are obtained assuming control bank D to be fully inserted with bank C at its insertion limit. The results are 0.84 percent ΔK and 14.3, respectively. The peak hot spot average fuel pellet enthalpy is 110 cal/gm. The peak clad average and fuel centerline temperatures are 1967°F and 3098°F, respectively.

A summary of the cases presented above is given in Table 3.2.17-1. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning-of-life full and zero power cases) are presented in Figures 3.2.17-1 and 3.2.17-2, and a time sequence of events is given in Table 3.2.17-2.

It is conservatively assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional THINC analysis. Although the analysis predicts limited fuel melting at the hot spot for the BOL Full-power case, in practice, melting is not likely since the analysis conservatively assumes that the hot spots before and after ejection were coincident.

A detailed calculation of the pressure surge for an ejected rod worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits (Reference 1). Since the severity of the present analysis does not exceed the "worst-case" analysis, the accident for this plant will not result in an excessive pressure rise or further adverse effects to the RCS.

E. Radiological Consequences

The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary are as follows:

	EB (0-2 Hr)	LPZ (0-30 Day)
Thyroid	5.9 E-1	6.9 E-2
γ-Body	1.6 E-2	2.3 E-3

3.2.17.6 Conclusions

Despite the conservative assumptions, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses demonstrate that the fission product release as a result of fuel rods entering DNB is limited to less than 10 percent of the fuel rods in the core. The resulting offsite doses are "well within" 10 CFR 100 guidelines.

3.2.17.7 References

1. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Special Kinetics Methods," WCAP-7588; Rev. 1A, January 1975.

2. Hargrove, H. G., "FACTRAN, a FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. "Fuel Densification-Turkey Point Unit No 3," WCAP-8074, February 1973.

4. Barry, R. F., Jr. and Risher, D. H., "TWINKLE, a Multi-dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975 (Proprietary) and WCAP-8028-A, January 1975 (Non-Proprietary).

5. Bishop, A. A., Sandberg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

6. Taxebius, T. G., ed., "Annual Report - SPERT Project, October 1968 - September 1969," IN-1370 Idaho Nuclear Corporation, June 1970.

7. Liimatainen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy 2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, P 177, November 1966.

8. Letter from W. J. Johnson of Westinghouse Electric Corporation to Mr. R. C. Jones of the Nuclear Regulatory Commission, Letter Number NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," October 23,1989.

9. USAEC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

Table 3.2.17-1

Results of the Rod
Cluster Control Assembly Ejection Accident Analysis

	Beginning of Cycle	Beginning of Cycle	End of Cycle	End of Cycle
Power level, percent	102	0	102	0
Ejected rod worth percent ΔK	0.35	0.71	0.30	0.84
Delayed neutron fraction, percent	0.50	0.50	0.42	0.42
Feedback reactivity weighting	1.3	1.42	1.3	2.32
Trip reactivity percent ΔK	4.0	2.0	4.0	2.0
F_q before rod ejection	2.694	--	2.694	--
F_q after rod ejection	5.48	8.0	5.52	14.3
Number of operational pumps	3	2	3	2
Max fuel pellet average temperature, °F	4286	2815	3457	2698
Max fuel centerline temperature, °F	5000	3267	4508	3098
Max clad average temperature, °F	2660	2033	2072	1967
Max fuel stored energy, cal/g	190	116	147	110
Fuel melt in hot pellet, percent	7.65	0	0	0

Table 3.2.17-2

Sequence of Events - RCCA Ejection Accident

Case	Event	Time (sec)
BOL, full power	Initiation of Rod Ejection	0.0
	Power Range High Neutron Flux Setpoint Reached	0.03
	Peak Nuclear Power Occurs	0.13
	Rods Begin to Fall	0.53
	Peak Clad Temperature Occurs	2.19
	Peak Heat Flux Occurs	2.20
	Peak Fuel Centerline Temperature Occurs	3.98
BOL, zero power	Initiation of Rod Ejection	0.0
	Power Range High Neutron Flux Setpoint Reached	0.25
	Peak Nuclear Power Occurs	0.30
	Rods Begin to Fall	0.75
	Peak Clad Temperature Occurs	2.31
	Peak Heat Flux Occurs	2.38
	Peak Fuel Centerline Temperature Occurs	3.40

Table 3.2.17-3

Assumptions Used for Rod Ejection Accident Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	60 µCi/gm of DE I-131
Activity Released to Reactor Coolant and Containment from Failed Fuel (Noble Gas & Iodine)	10.0% of Core Gap Activity
Fraction of Core Activity in Gap (Noble Gas & Iodine)	0.10
Activity Released to Reactor Coolant and Containment from Melted Fuel	
Iodine	0.125% of Core Activity
Noble Gas	0.25% of Core Activity
Secondary Coolant Activity Prior to Accident	0.10 µCi/gm of DE I-131
Total SG Tube Leak Rate During Accident	1.0 gpm
Iodine partition Factor in SGs	0.01
Steam Release from SGs	281,569 lb (0-95 sec)
Iodine Removal in Containment	
Instantaneous Iodine Plateout	50%
Elemental Iodine Deposition	5.95 hr^{-1} for DF $\leq$ 100 0 for DF > 100
Emergency Containment Filters	
Start Delay Time	300 sec
Number of Units	2
Flow Rate per Unit	33,750 cfm

Table 3.2.17-3 (cont.)

Assumptions Used for Rod Ejection Accident Dose Analysis

Filter Efficiency	
Elemental	90%
Methyl	30%
Particulate	95%
Operating Time	2 hr
Containment Free Volume	1.55×10^6 ft^3
Containment Leak Rate	
0-24 hr	0.25%/day
> 24 hr	0.125%/day

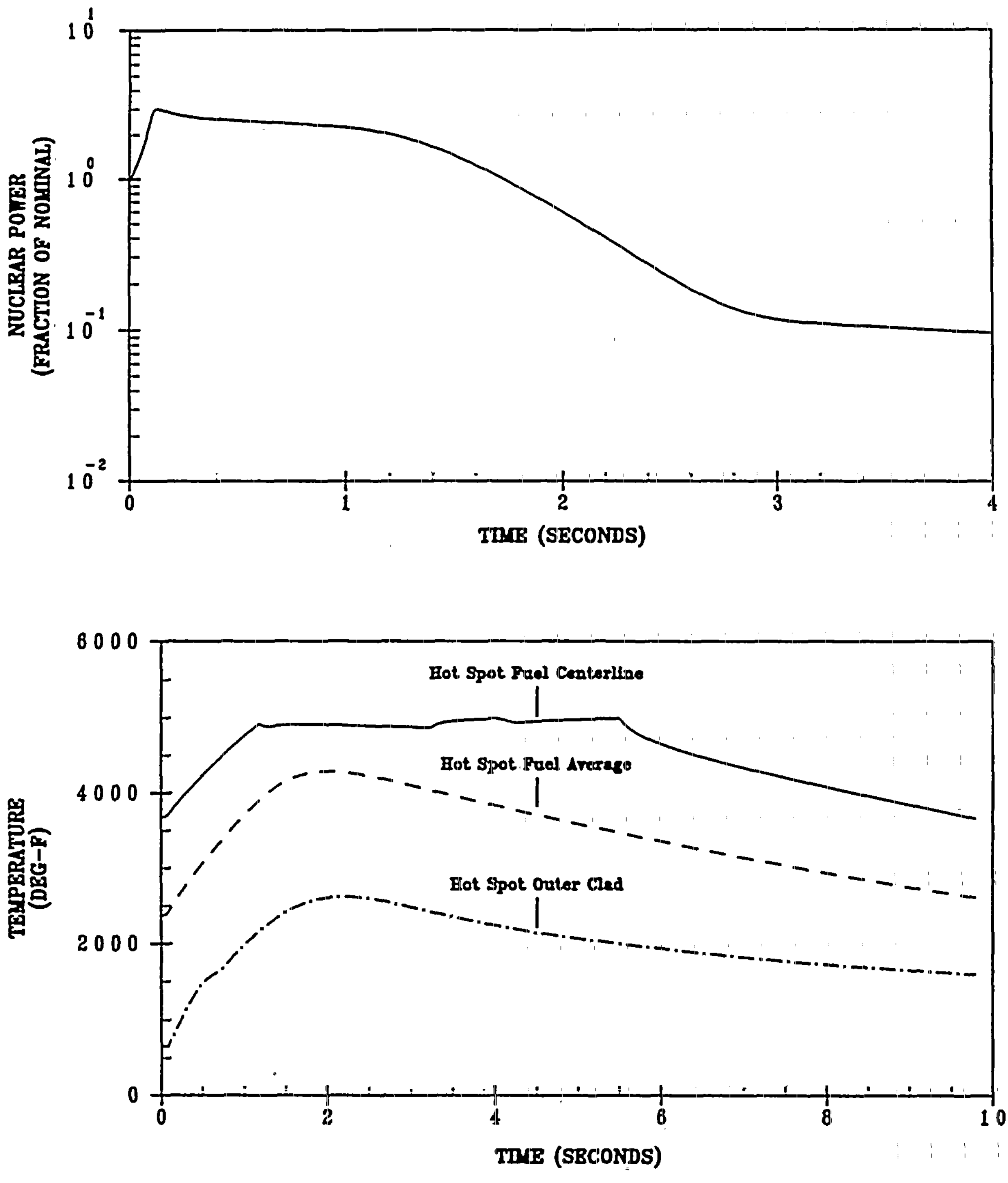


Figure 3.2.17-1 Rod Ejection Transient Beginning of Life, Full Power

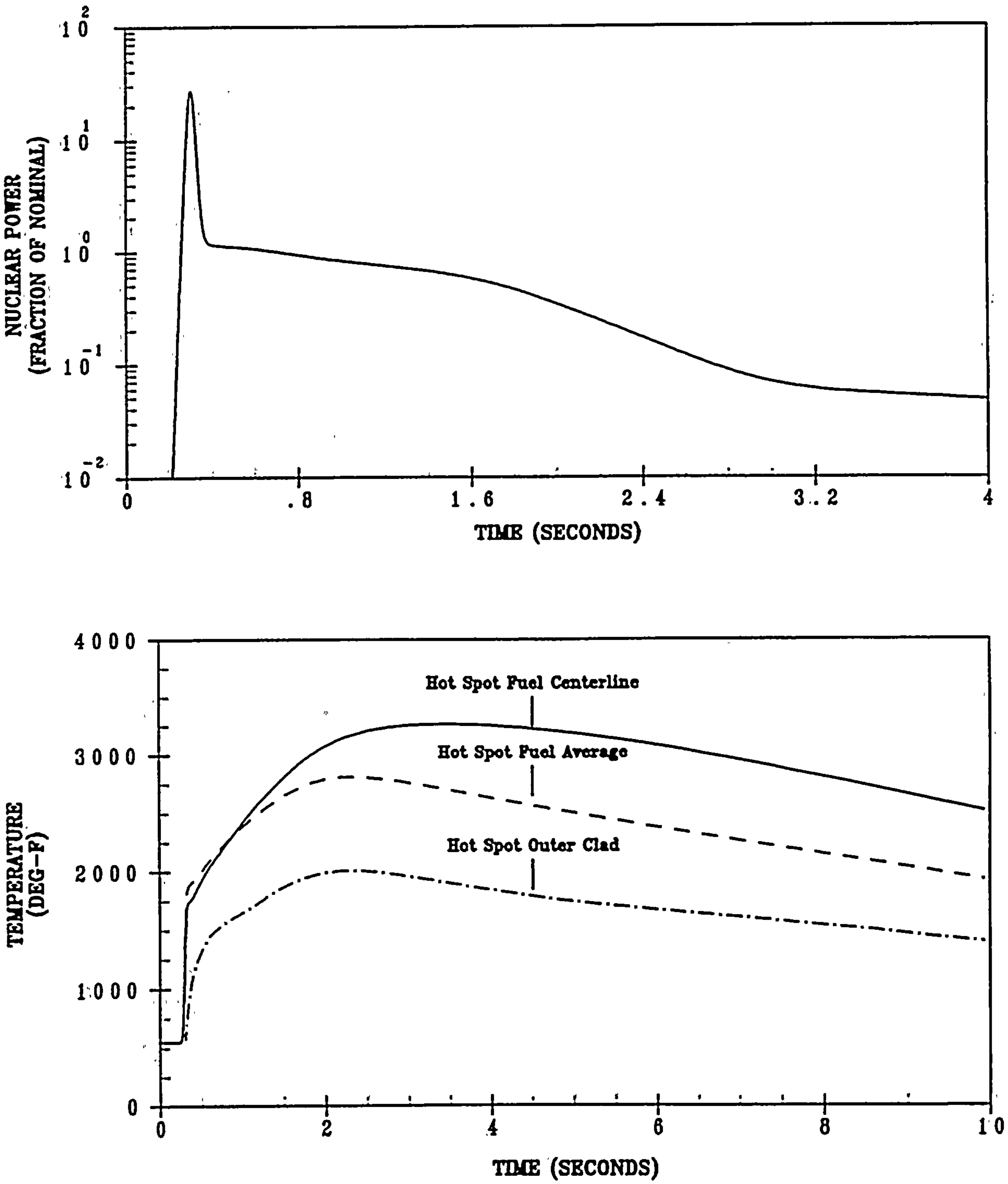


Figure 3.2.17-2 Rod Ejection Transient Beginning of Life, Zero Power

3.3 LOCA AND LOCA RELATED EVENTS

3.3.1 Large Break LOCA Accident Analysis

3.3.1.1 Introduction

This report contains information regarding the large break Loss-of-Coolant Accident (LOCA) analysis and evaluations performed in support of the uprating program for Turkey Point Units 3 and 4. A LOCA is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analyses reported here, a large break is defined as a rupture of the RCS piping with a cross-sectional area greater than 1.0 ft^2. This event is considered an American Nuclear Society (ANS) Condition IV event, which are design limiting faults that are not expected to occur during the life of a plant.

The purpose of analyzing the large break LOCA is to demonstrate conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the uprating. Important input assumptions, as well as analytical models and analysis methodology for the large break LOCA, are contained in subsequent sections. Analysis results are provided in the form of tables and figures, as well as a more detailed description of the limiting transient. It was determined that no design or regulatory limit related to the large break LOCA would be exceeded due to the uprated power and assumed plant parameters.

3.3.1.2 Input Parameters And Assumptions

The following important plant conditions and features are listed in Table 3.3.1-1. Several additional considerations that are not identified in Table 3.3.1-1 are discussed below:

The axial power shapes modeled in the large break LOCA analysis are the chopped cosine shape and a standard set of top-skewed shapes. A methodology has been implemented that explicitely considers top-skewed power shapes in the large break LOCA analysis. This methodology, known as ESHAPE, has scaled a set of top-skewed power shapes to the standard two-line segment K(Z) curve. This methodology has been utilized for the FPL large break LOCA analysis.

Figure 3.3.1-1 provides the degraded HHSI and the LHSI flow versus pressure curve modeled in the large break LOCA analysis.

Additional input assumptions and conditions upon which the large break analysis was based are listed in Tables 3.3.1-1 and 3.3.1-2. A complete list of plant specific Accident Analysis Parameters was confirmed by FPL for use in the large break LOCA analysis as part of the uprating program.

3.3.1.3 Description Of Analyses / Evaluations Performed

Analytical Model

The LOCA analysis presented here was performed with the BASH Westinghouse ECCS evaluation model (References 2 and 3). This version includes the BART (Reference 4) computer code which is a mechanistic core heat transfer model, and BASH which is a mechanistic reflood model.

The large break LOCA transient can be conveniently divided into three periods: blowdown, refill, and reflood. Also, three physical parts of the transient are analyzed for each period: the thermal-hydraulic transient in the reactor coolant system, the containment pressure and temperature, and the fuel and clad temperatures of the hottest rod. These considerations lead to the use of a system of computer codes designed to model the large break LOCA transient.

The SATAN-VI (Reference 5) code evaluates the thermal-hydraulic transient during blowdown. The REFILL (References 3 and 6) code computes, using output from the SATAN-VI code, the time to bottom of core recovery (BOCREC) and RCS conditions at BOCREC. Since the mass flow rate to the containment depends upon the local RCS and containment conditions, the REFILL and COCO codes are interactively linked. The COCO (Reference 7) code is used to model the containment pressure transient. The containment parameters used by COCO to determine the ECCS backpressure were reviewed by FPL prior to use in the LOCA reanalysis and are summarized in Table 3.3.1-2. The BOCREC conditions calculated by REFILL are used as input to the BASH code. Data from both the SATAN-VI code and the REFILL code out to BOCREC are input to the LOCBART (Reference 4) code which calculates core average conditions at BOCREC for use by the BASH code.

BASH provides a thermal-hydraulic response of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by REFILL. A more detailed description of the BASH code is available in Reference 2. The BASH code provides a sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. A dynamic interaction between core thermal-hydraulics and system behavior is expected, and experiments have shown this behavior. The BART code has been coupled with a loop model to form the BASH code and BART provides the entrainment rate for a given flooding rate. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated inlet flow is used by BART to calculate a new entrainment rate fed back to the loop code. This process of transferring data between BART, the loop code and back to BART forms the calculational process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a more dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior. The BASH code is also interactively linked with COCO to utilize the local conditions at each time step to calculate the containment response.

In the BASH ECCS model, the cladding heat-up transient is calculated by LOCBART which is a combination of the LOCTA (Reference 8) code with BART (Reference 4). A more detailed description of the LOCBART code can be found in (Reference 2). The LOCBART code is used throughout the transient to compute fuel and clad temperatures in the hottest rod. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

Figure 3.3.1.2 shows the interaction of the BASH large break model and the relationship of the computer codes to the LOCA sequence of events.

Analysis

Past licensing studies for break type and location were performed for a double-ended cold leg guillotine (DECLG) break with various values of discharge coefficient (C_D), double-ended hot leg guillotine (DEHLG), double-ended pump suction guillotine (DEPSG), and a range of split-type break sizes ranging from a 1.0 ft^2 area to a full double-ended area of the cold leg. This study determined that the DECLG type break was both the most limiting type and location. Furthermore, previous licensing basis analysis for Turkey Point has shown that the limiting discharge coefficient, C_D=0.4, is much more limiting than the non-limiting discharge coefficients, C_D=0.6 and C_D=0.8. Therefore, only the limiting Moody discharge coefficient, C_D=0.4, was reperformed utilizing the BASH evaluation model (EM). Sensitivities were performed of the RCS vessel average temperature as well as the top skewed power shapes.

The limiting single active failure used in the large break LOCA analysis is dependent upon the Maximum and Minimum ECCS scenarios. For the case of Minimum ECCS, the limiting single failure is the loss of the LHSI pump. Failure of the diesel generator is not limiting for large break LOCA due to the loss of a containment spray pump. Operation of all containment pressure reducing equipment is required by 10 CFR 50, Appendix K, as this results in a minimum containment pressure transient. In addition to the loss of a LHSI pump, the large break LOCA analysis conservatively assumed failure of one HHSI pump, but still modeled both containment spray pumps. The approval of the BASH EM (Reference 2) specifically requires consideration of the Maximum ECCS scenario. The Maximum ECCS analysis assumes no single failure within the ECCS. The limiting single failure assumed in the Maximum ECCS analysis is the loss of an auxiliary feedwater pump. The Maximum ECCS analysis requirement is dependent upon a full downcomer at the start of the reflood phase. Because Turkey Point does not have a full downcomer at the beginning of reflood, the Maximum ECCS analysis is unnecessary. Additional ECCS injection during the Maximum ECCS analysis will only contribute to filling the downcomer and increasing the reflood rate.

The limiting time for fuel burnup in the large break LOCA analysis is at the beginning of life where maximum pellet temperatures occur. The beginning of life analysis will bound burnup conditions up to 62,000 MWD/MTU.

Prior to break initiation, the plant is assumed to be in a full power (102%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Section 2.0 and Table 3.3.1-1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid.

Loss of Offsite Power (LOOP) is assumed to occur coincident with initiation of the large break LOCA. If a large break LOCA occur, depressurization of the RCS results in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1805 psia, is reached. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1615 psia, is reached. The safety injection signal may also result from the containment high signal. Both signals are modeled in the large break LOCA analysis and the fastest initiation of safety injection is used. Safety injection is delayed 35 seconds after the occurrence of the signal. This delay accounts for signal initiation, diesel generator start up and emergency power bus loading, as well as the time involved in aligning the valves and bringing the LHSI and HHSI pump up to full speed. Finally the RCS depressurizes to below 615 psia and the accumulators begin to inject borated water. These countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the large break LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. No credit is taken for control rod insertion. The core is shut down on only void formation during the depressurization result.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

The core heat removal mechanisms associated with the large break transient include the break itself and the injected ECCS water.

<u>Evaluations</u>

The effect of the open containment purge valves has been considered by evaluation. The Turkey Point Units 3 and 4 will have 48 and 54 inch diameter containment purge valves open for the initial seconds of the large break LOCA transient.

3.3.1.4 Acceptance Criteria For Analyses / Evaluations

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F,

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation,

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling, and

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Criteria 1 through 3 are explicitly covered by the large break LOCA analysis at uprated conditions.

For criterion 4), the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the PCT criterion of 10 CFR 50.46 and consequently, demonstrate the core remains amenable to cooling.

For criterion 5), Long-Term Core Cooling (LTCC) considerations are not directly applicable to the large break LOCA transient, but are assessed in Section 3.3.5 as part of the evaluation of ECCS performance.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA.

3.3.1.5 Results

In order to determine the conditions that produced the most limiting large break LOCA case (as determined by the highest calculated peak cladding temperature), two cases were examined. These cases included the limiting discharge coefficient, C_D=0.4, for high and low RCS T_{avg} operation. The limiting condition for the Turkey Point Units was found to be low RCS T_{avg} operation. The PCT attained during the low RCS T_{avg} transient was 2103°F, while the PCT for the high RCS T_{avg} transient was 2082°F (refer to Table 3.3.1-3). Table 3.3.1-4 provides the key transient event times.

A summary of the transient response for the limiting low T_{avg} C_D=0.4 break case is shown in Figures 3.3.1-3 through 3.3.1-18.

Limiting Temperature Conditions

Reduced operating temperature sometimes results in a PCT benefit for the large break LOCA. However, due to competing effects and the complex nature of large break LOCA transients, there have been some instances where more limiting results have been observed for the reduced operating temperature case. For this reason, a large break LOCA transient based on both a lower and upper bound RCS vessel average temperature was performed, and the lower bound was found to be more limiting. The lower bound RCS vessel temperature has a higher initial RCS mass which could prolong the blowdown period and decrease the water left in the accumulator at the end of blowdown.

The temperature window analyzed was based on a nominal vessel average temperature of 574.2°F, with ± 3°F for an operating window and ± 8.5°F to bound uncertainties. The upper bound vessel average temperature is 585.7°F, while the lower bound vessel average temperature is 562.7°F.

Plots of important parameters are given in Figures 3.3.1-19 through 3.3.1-28 at high T_{avg} conditions.

Skewed Power Shapes

Large Break LOCA analyses have traditionally been performed using a symmetric, chopped cosine, core axial power shape. Under certain conditions, calculations have shown that there is a potential for top-skewed power distributions to result in PCTs greater than those calculated with a chopped cosine axial power distribution. Explicit analyses were performed in which power distributions were skewed to peak power at the 8.5, 9.5, and 10.5 ft. elevations. The analyses results demonstrated that the 9.5 and 10.5 ft. skewed power shapes are bounded by the chopped cosine power shape, while a PCT increase of 14°F was calculated for the 8.5 ft skewed power shape. This resulted in a limiting case PCT of 2117°F.

Plots of important parameters are given in Figures 29 through 44 for the 8.5 ft. top-skewed power shape.

Evaluations

The Turkey Point Units will have 48 and 54 inch diameter containment purge valves open for the initial seconds of the large break LOCA transient. The open valves will reduce the containment pressure response during the large break LOCA, which is an adverse effect upon the calculated PCT. The calculated PCT effect is an increase of 27°F. Therefore, the limiting case PCT with evaluations is 2144°F.

The DRFA fuel stack height above the lower core plate was explicitly modeled for the various cases analyzed.

3.3.1.6 Conclusions

A limiting discharge coefficient, C_D=0.4, large break LOCA analysis supporting a range of vessel average temperature was performed. Peak cladding temperatures of 2103°F and 2082°F were calculated for the RCS low (562.7°F) and high (585.7°F) T_{avg} conditions respectively. After assessing the PCT effect for top skewed power shapes and containment purge on the most limiting case, the resulting PCT is 2144°F.

The analyses presented in this section show that the Emergency Core Cooling System provides sufficient core heat removal capability to maintain the calculated peak cladding temperatures below the required limit of 10 CFR 50.46. That is:

1. The calculated peak fuel element cladding temperature does not exceed 2200°F,

2. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching,

3. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen, does not exceed 1 percent of the total amount of fuel rod cladding,

4. The core remains amenable to cooling during and after the break, and

5. 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.

Hence, adequate protection is afforded by the emergency core cooling system in the event of a large break Loss-of-Coolant Accident.

Radiological Consequences

3.3.1.7 Introduction

A large pipe rupture in the RCS is assumed to occur. As a result of the accident, it is assumed that core damage occurs and iodine and noble gas activity is released to the containment atmosphere. A portion of this activity is released via containment leakage to the outside atmosphere. It is assumed that the containment purge system is open when the accident occurs and activity is released to the atmosphere through this path until the containment purge system is isolated. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from these releases.

3.3.1.8 Input Parameters and Assumptions

The offsite and control room doses due to containment leakage and due to an open containment purge system following a large break loss-of-coolant accident (LOCA) are determined using the analytical methods and assumptions of the Standard Review Plan (Reference 9). The assumptions are presented in Table 3.3.1-5.

3.3.1.9 Description of Analyses Performed

The offsite thyroid and γ-body doses, as well as the control room thyroid, γ-body and β skin doses, are determined for both the containment leak and containment purge activity release paths.

3.3.1.10 Acceptance Criteria

The offsite doses must be within the guidelines of 10CFR100, or 300 rem thyroid and 25 rem γ-body for the initial 2 hour period following the accident at the Exclusion Boundary (EB) and for the duration of the accident at the LPZ. The dose criteria for control room personnel following the accident are 5 rem γ-body, 30 rem thyroid, and 30 rem β skin (or 75 rem β skin with protective clothing).

3.3.1.11 Results

The offsite and control room doses due to containment leakage and containment purge, along with the total doses due to the activity release from these paths are within the acceptance criteria in Section 3.3.1.10.

The offsite and control room doses (rem) due to a LOCA are summarized below:

1. Thyroid

	EB (0-2 Hr)	LPZ (0-30 Day)	CR (0-30 Day)
Containment Leakage	2.33 E1	2.76 E0	1.49E+1
Containment Purge	2.91 E-1	2.83 E-2	7.28 E-2
Total	2.36 E1	2.79 E0	1.50 E1

2. γ-Body

	EB (0-2 Hr)	LPZ (0-30 Day)	CR (0-30 Day)
Containment Leakage	1.04 E0	1.61 E-1	4.39 E-1
Containment Purge	6.48 E-5	6.31 E-6	1.09 E-5
Total	1.04 E0	1.61 E-1	4.39 E-1

3. β-Skin

	CR (0-30 Day)
Containment Leakage	2.0 E1
Containment Purge	8.9 E-4
Total	2.0 E1

3.3.1.12 Conclusions

The total offsite doses and the total control room doses due to the large LOCA are within the acceptance criteria.

3.3.1.13 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Kabadi, J. N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A Rev. 2 (proprietary), WCAP-11524-NP-A Rev 2 (non-proprietary), March 1987; including Addendum 1-A 'Power Shape Sensitivity Studies' 12/87 and Addendum 2-A 'BASH Methodology Improvements and Reliability Enhancements' May 1988.

3. "Change in Methodology for Execution of BASH Evaluation Model," NTD-NRC-94-4143, May 23, 1994.

4. Young, M., et al., "BART-1A: A Computer Code for the Best Estimate Analyzed Reflood Transients", WCAP-9561-P-A (proprietary), WCAP-9695-NP-A (non-proprietary), 1984; including Addendum 3 Rev 1, July 1986.

5. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant", WCAP-8302-P (proprietary), WCAP-8306-NP (non-proprietary), June 1974.

6. Kelly, R. D., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WReflood Code)", WCAP-8170-P (proprietary), WCAP-8171-NP (non-proprietary), et al., June 1974.

7. Bordelon, F. M., and E. T. Murphy, "Containment Pressure Analysis Code (COCO)", WCAP-8327-P (proprietary), WCAP-8326-NP (non-proprietary), June 1974.

8. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (proprietary) and WCAP-8305 (non-proprietary), June 1974.

9. NUREG-0800, Standard Review Plan 15.6.5, Appendix A, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution", Rev. 1, July 1981.

Table 3.3.1-1

Input Parameters Used In The Large Break LOCA Analysis

Parameter	High T_{avg}	(Low T_{avg})
Reactor core rated thermal power[1], (MWt)	2300	
Peak linear power[1], (kw/ft)	14.0	
Total peaking factor (F_Q^T) at peak	2.35	
Power shape	Chopped Cosine and Top-Skewed	
$F_{\Delta H}$	1.64	
Fuel	15x15 DRFA	
Accumulator water volume, minimum (ft^3/acc.)[2]	865	
Accumulator tank volume (ft^3/acc.)[2]	1200	
Accumulator gas pressure, minimum (psig)	600	
Pumped safety injection flow	See Figure 1	
Steam generator tube plugging level (%)[3,4]	5	
Thermal Design Flow/loop, (gpm)[5]	85,000	
Vessel average temperature w/ uncertainties, (°F)	585.7	(562.7)
Reactor coolant pressure w/ uncertainties, (psia)	2320	

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1 Two percent is added to this power to account for calorimetric error.

2 The analysis value assumed the Tech Spec minimum and credited additional accumulator line volume.

3 Maximum plugging level in any one or all steam generators.

4 The analysis was performed at a SGTP level of 10% to bound the combined LOCA+Safe Shutdown Earthquake tube crush issue.

5 Flowrates conservatively based on 20% steam generator tube plugging.

Table 3.3.1-2

Large Break LOCA Containment Data for PCT Calculation

Net Free Volume	1,550,000 ft^3
Initial Conditions	
Pressure	12.7 psia
Temperature	90.0°F
RWST Temperature	35.0°F
Temperature Outside Containment	39.0°F
Initial Spray Temperature	39.0°F
Spray System	
Maximum Flow for one Spray Pump	1821.5 gpm
Number of Spray Pumps Operating	2
Post-Accident Spray System Initiation Delay	26 sec
Containment Fan Coolers	
Post-Accident Initiation Fan Coolers	26 sec
Number of Fan Coolers Operating	3

Table 3.3.1-3

Large Break LOCA Analysis Fuel Cladding Results

Moody Discharge Coefficient, C_D=0.4

	Low T_{avg}	High T_{avg}	8.5-foot Peak Power Shape[1]
Peak Cladding Temperature (°F)	2103	2082	2117
Peak Cladding Temperature Location (ft)[2]	8.00	8.00	8.50
Peak Cladding Temperature Time (sec)	137.5	146.3	128.9
Local Zr/H_2O Reaction, Max (%)	9.32	7.34	6.48
Local Zr/H_2O Reaction Location (ft)[2]	6.00	6.00	8.50
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	44.5	41.6	49.3
Hot Rod Burst Location (ft)[2]	6.00	6.00	7.00

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1 The 9.5-foot and 10.5-foot top-skewed shapes were shown to be non-limiting compared to the cosine.

2 Height from bottom of active fuel.

Table 3.3.1-4

Large Break LOCA Analysis Time Sequence of Events

Moody Discharge Coefficient, C_D=0.4

	Low T_{avg}	High T_{avg}	8.5-foot Peak Power Shape
Start of LOCA with LOOP (sec)	0.00	0.00	0.00
Reactor Trip Setpoint Exceeded (sec)	0.546	0.654	0.546
Safety Injection Setpoint Exceeded (sec)[1]	1.9	1.7	1.9
Accumulator Injection Begins (sec)	14.4	15.4	14.3
End-of-Bypass (sec)	33.205	29.593	33.156
End-of-Blowdown (sec)	33.205	31.389	33.156
Pump Injection Begins (sec)	36.9	36.7	36.9
Bottom of Core Recovery (sec)	53.9	50.7	53.8
Accumulator Empty (sec)	62.83	62.02	61.70
PCT Time (sec)	137.5	146.3	128.9

1 Safety Injection signal actuated off of containment high pressure as opposed to low pressurizer pressure.

Table 3.3.1-5

Assumptions Used for Large Break LOCA Dose Analysis
Containment Leakage

Power	2346 MWt
Iodine Chemical Species	
Elemental	91%
Methyl	4%
Particulate	5%
Iodine Removal in Containment	
Instantaneous Iodine Plateout	50%
Elemental Iodine Deposition	$5.94\ hr^{-1}$ for DF $\leq$ 100 0 for DF > 100
Emergency Containment Filters	
Start Delay Time	90 sec
Number of Units	2
Flow Rate per Unit	33,750 cfm
Filter Efficiency	
Elemental	90%
Methyl	30%
Particulate	95%
Operating Time	2 hr
Containment Free Volume	$1.55 \times 10^6\ ft^3$
Containment Leak Rate	
0-24 hr	0.25%/day
>24 hr	0.125%/day

Table 3.3.1-5 (continued)

Assumptions Used for Large Break LOCA Dose Analysis Containment Purge

Power	2346 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect level
Reactor Coolant Iodine Activity Prior to Accident	60 μCi/gm of DE I-131
Iodine Chemical Form	100% Elemental
Containment Purge System Flow Rate	7000 cfm
Containment Purge System Isolation Time	8 sec
Containment Purge System Filtration	None
ECFS Filtration	None
Iodine Plateout/Deposition in Containment	None
Containment Free Volume	1.55×10^6 ft^3

Table 3.3.1-5 (continued)

Assumptions Used for Large Break LOCA Dose Analysis
Control Room

Volume	50,301 ft^3
Unfiltered Inleakage	10 cfm
Filtered Makeup	525 cfm
Filtered Recirculation	375 cfm
Filter Efficiency	
Elemental	95%
Methyl	95%
Particulate	95%
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

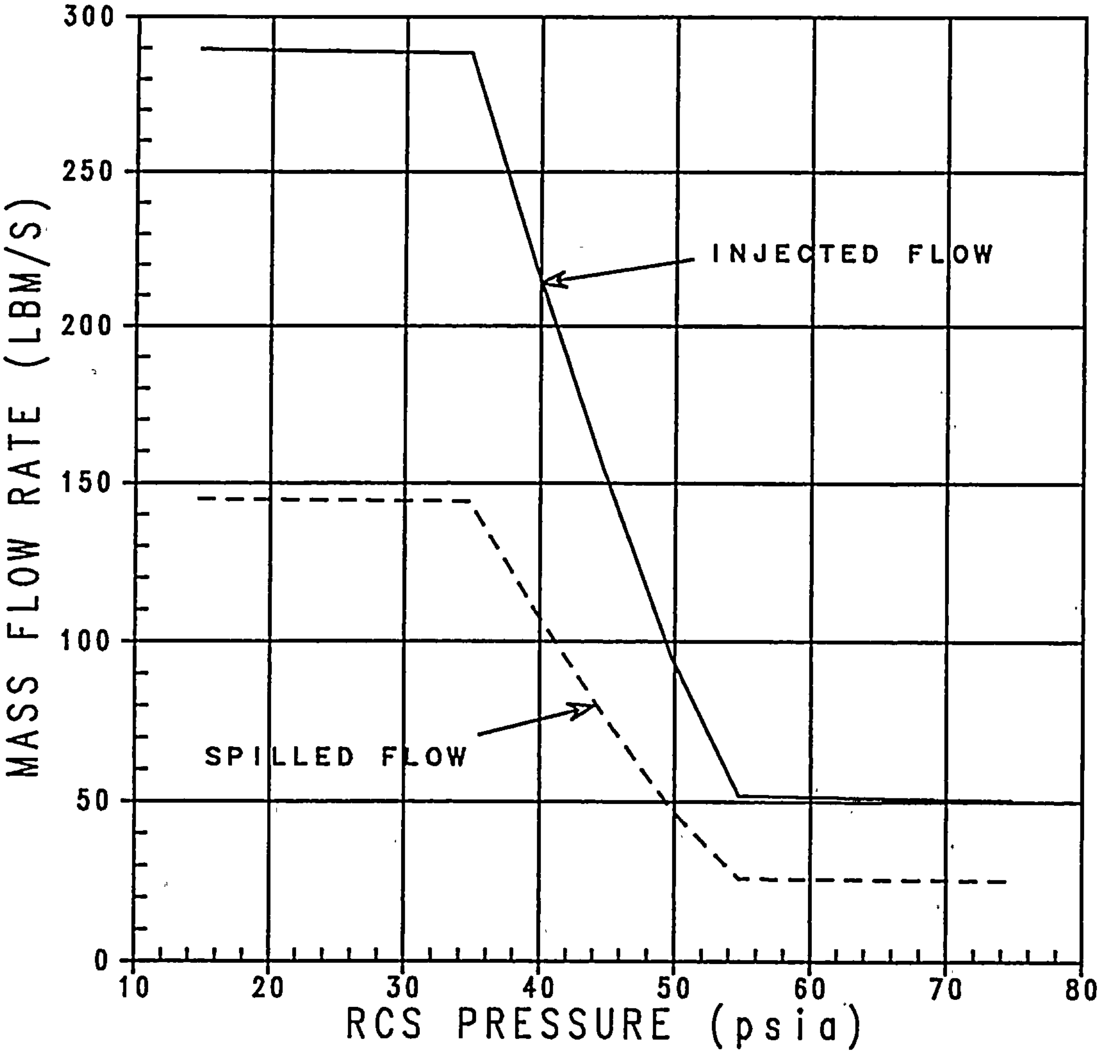

Figure 3.3.1-1: Large Break Pumped Safety Injection Flow Rate - 1 HHSI and 1 LHSI Pump

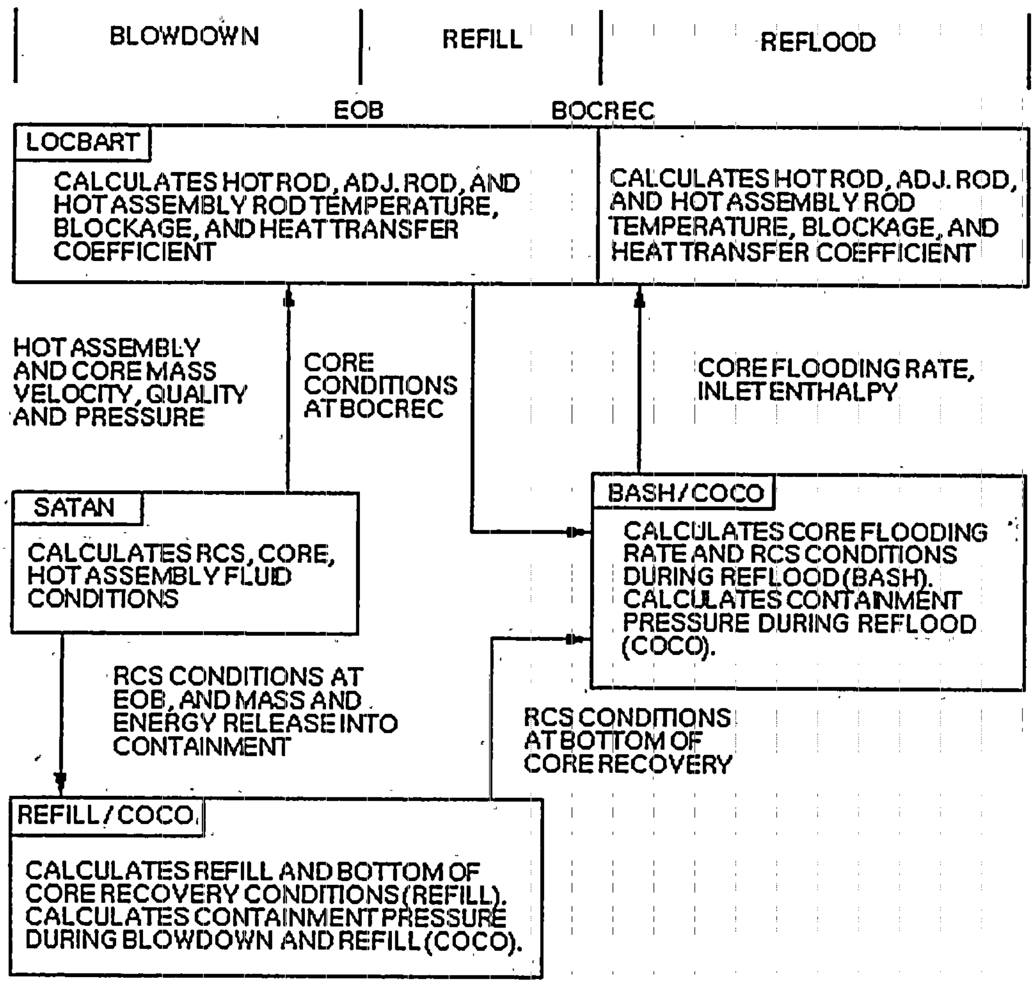

Figure 3.3.1-2: Code Interface Description for the Large Break LOCA Model

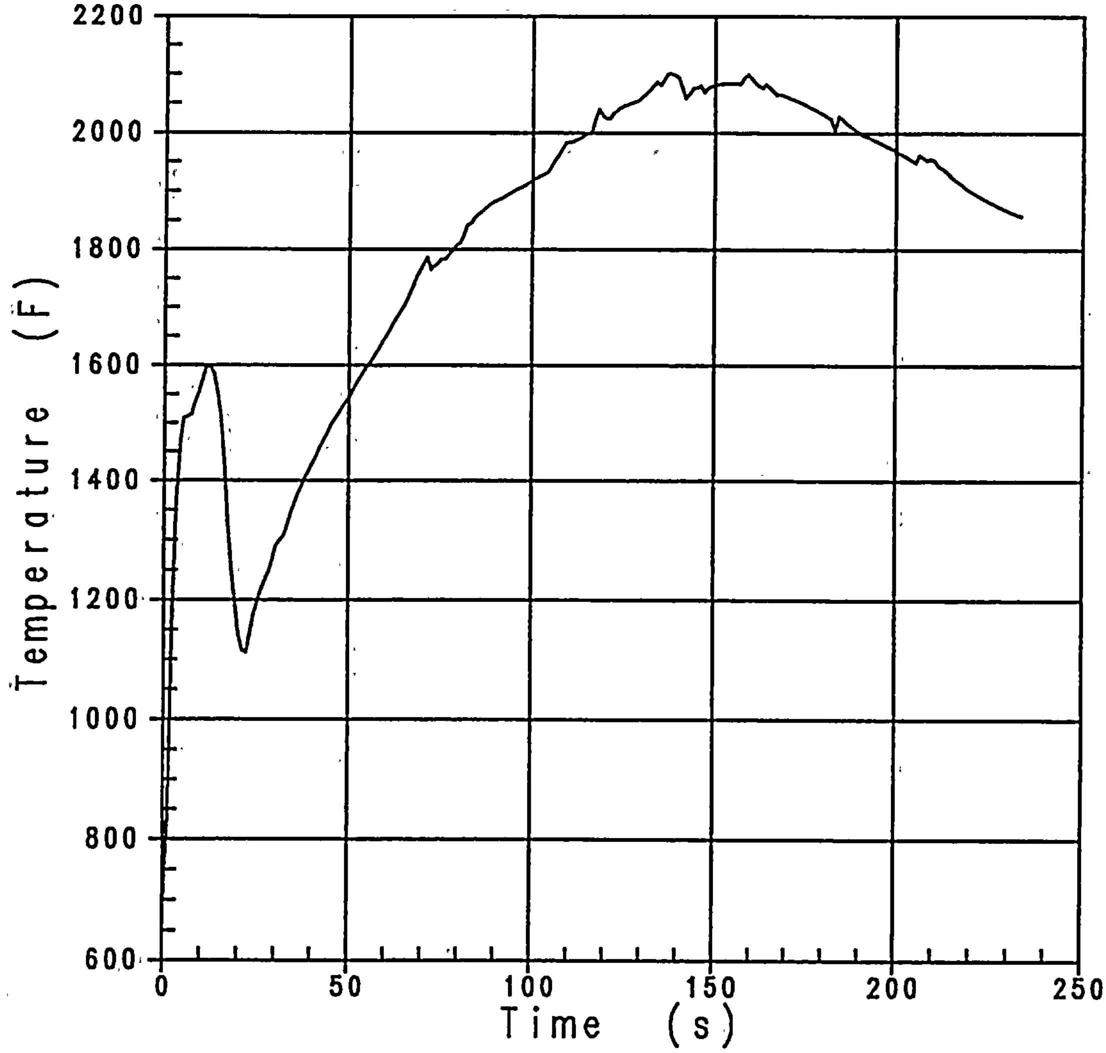

Figure 3.3.1-3: Peak Cladding Temperature for C_D=0.4, Low T_{avg}

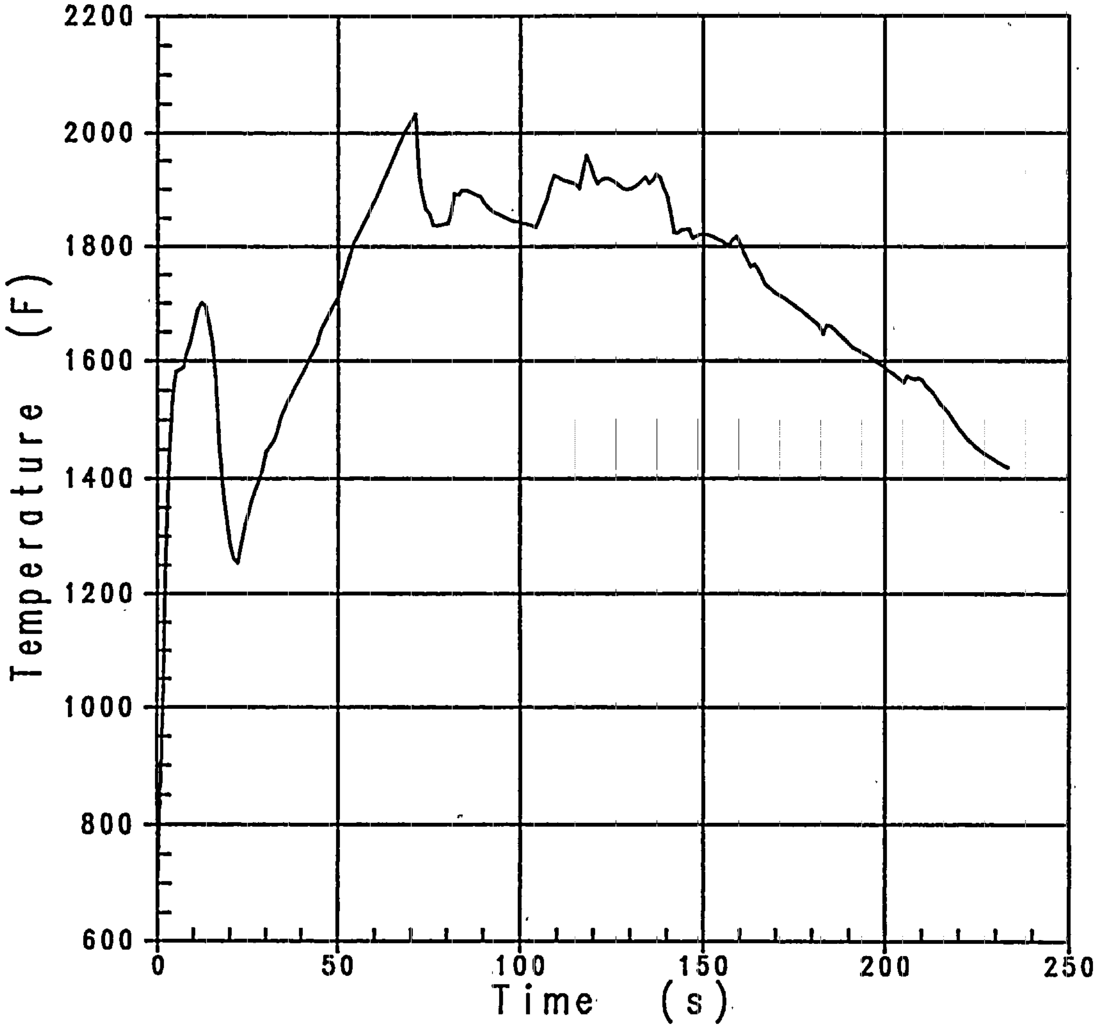

Figure 3.3.1-4: Cladding Temperature at Fuel Rod Burst Location for C_D=0.4, Low T_{avg}

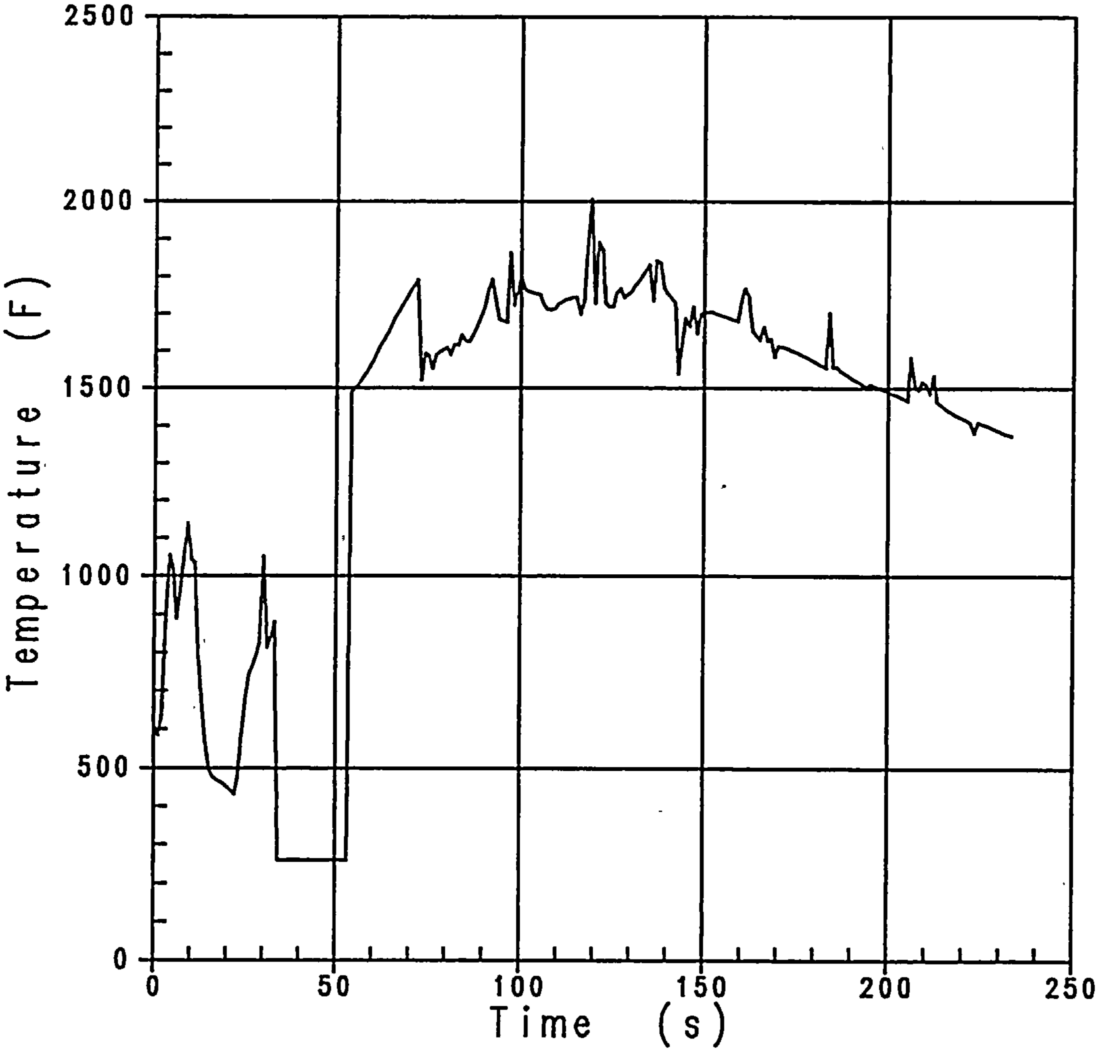

Figure 3.3.1-5: Local Fluid Temperature at PCT Elevation for C_D=0.4, Low T_{avg}

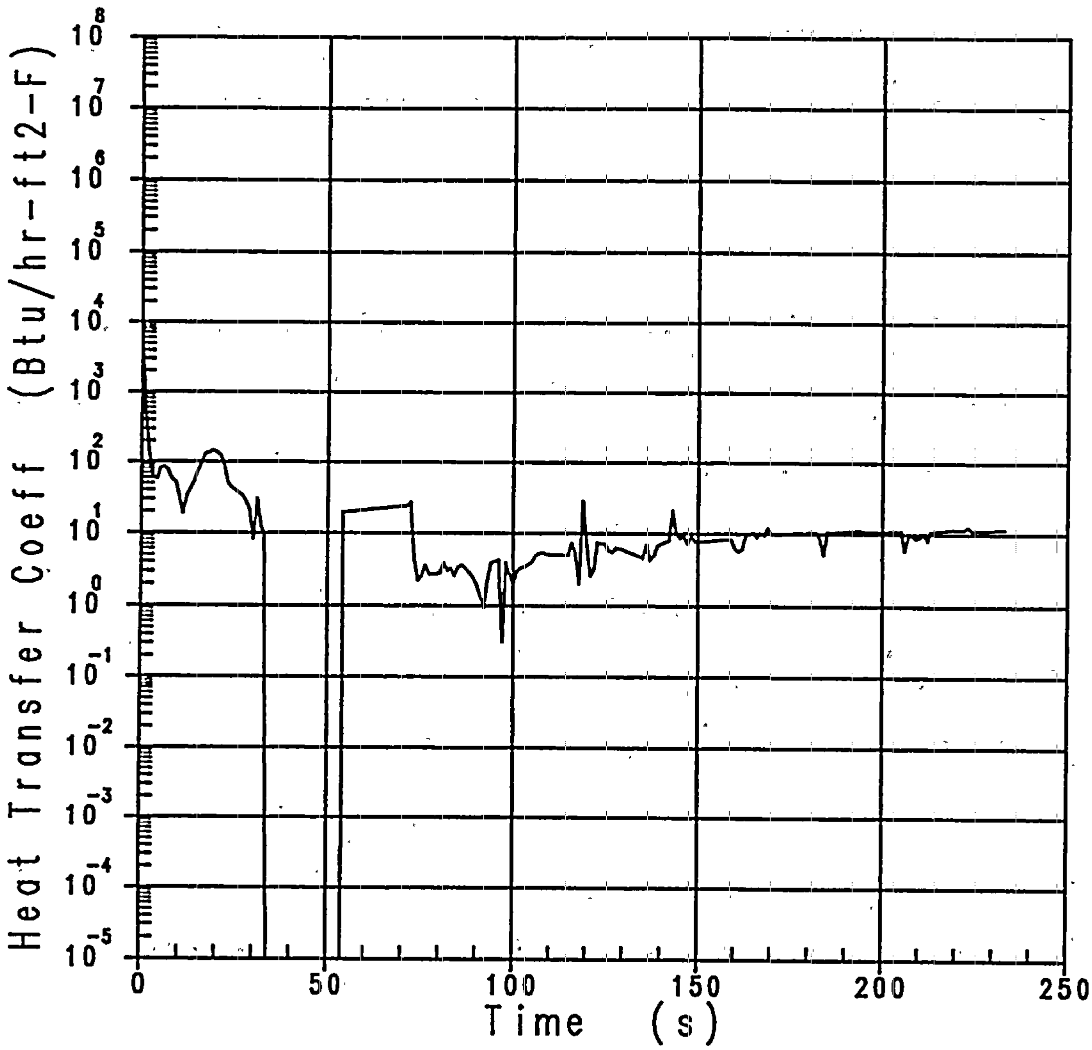

Figure 3.3.1-6: Local Heat Transfer Coefficient at PCT Elevation for C_D=0.4, Low T_{avg}

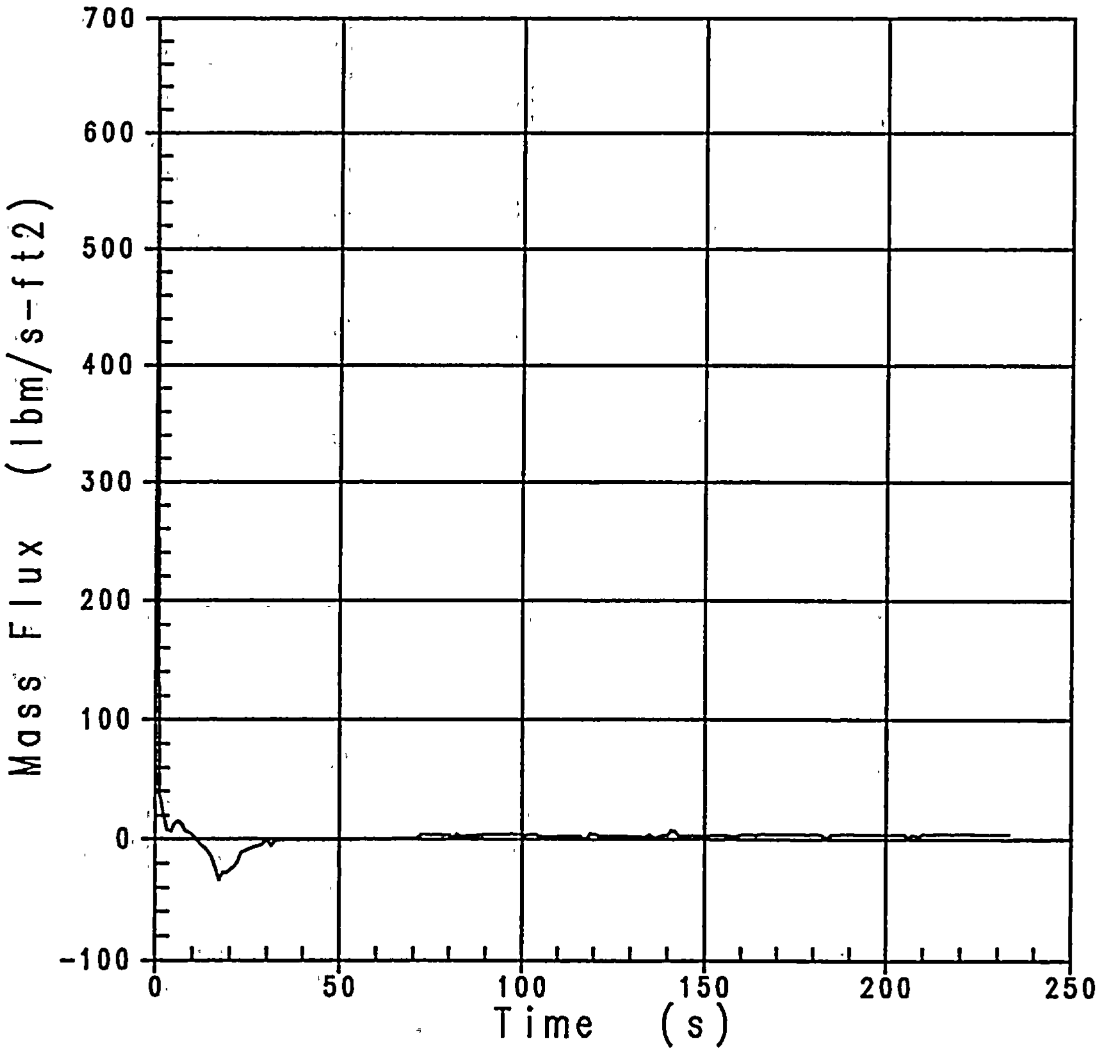

Figure 3.3.1-7: Local Mass Flux at PCT Elevation for C_D=0.4, Low T_{avg}

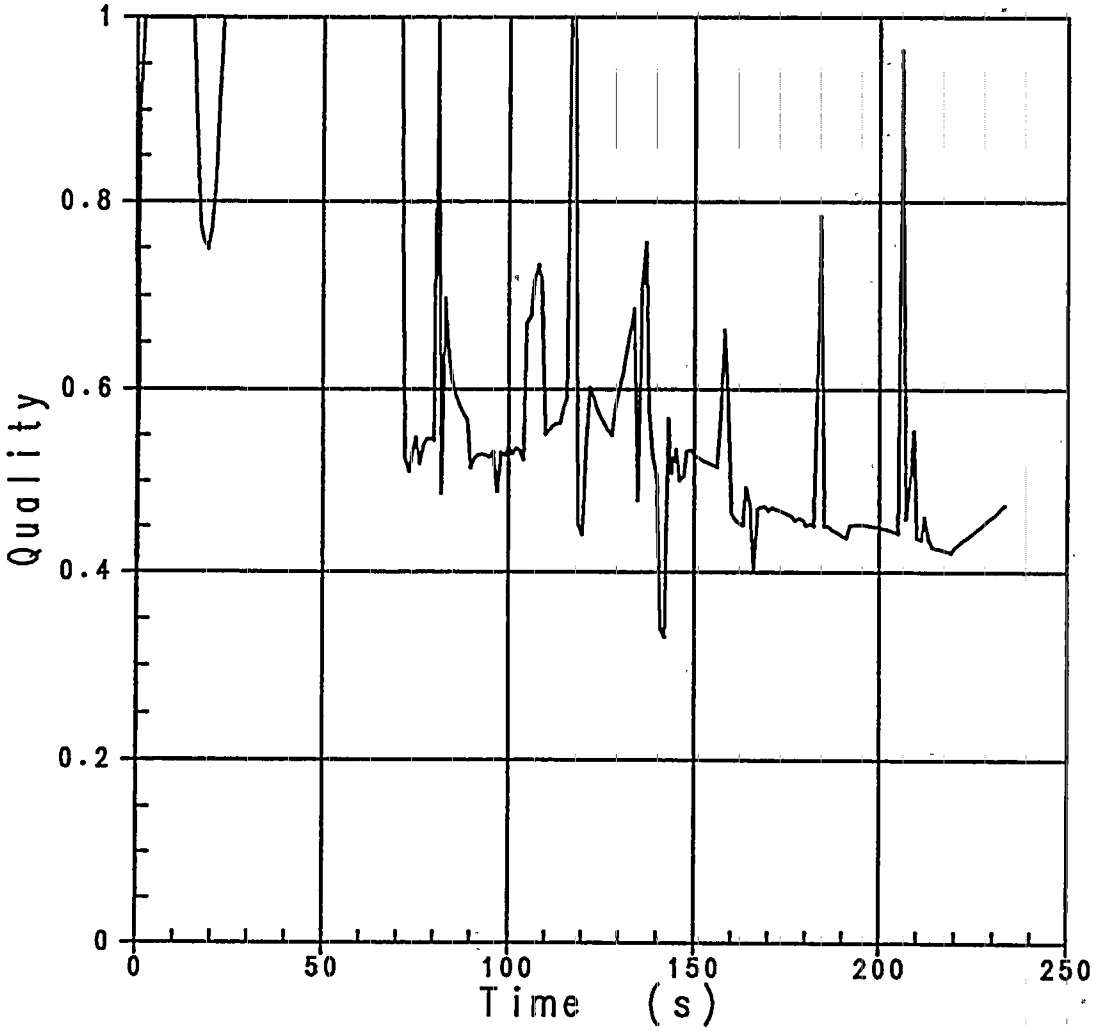

Figure 3.3.1-8: Local Quality at PCT Elevation for C_D=0.4, Low T_{avg}

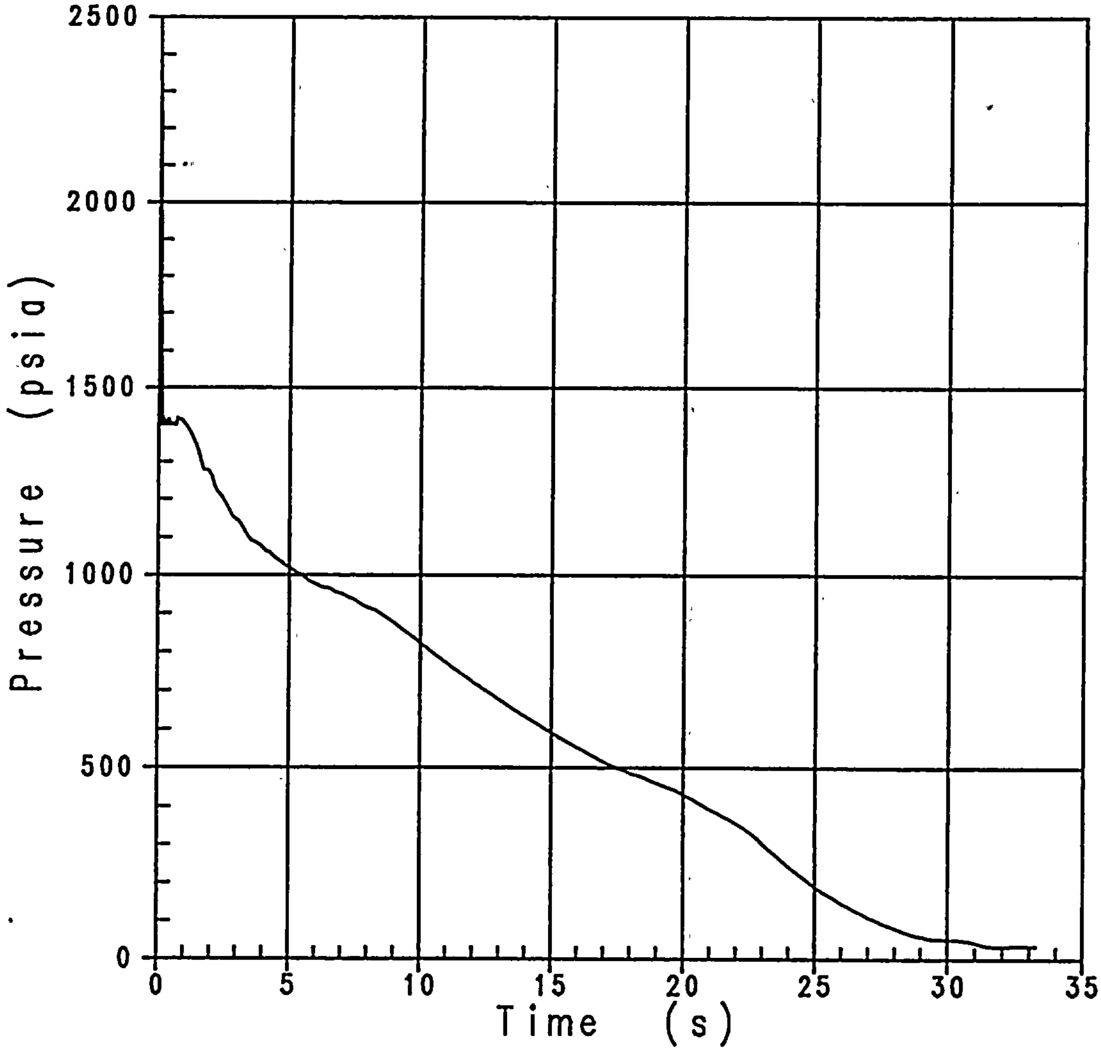

Figure 3.3.1-9: RCS Pressure During Blowdown for C_D=0.4, Low T_{avg}

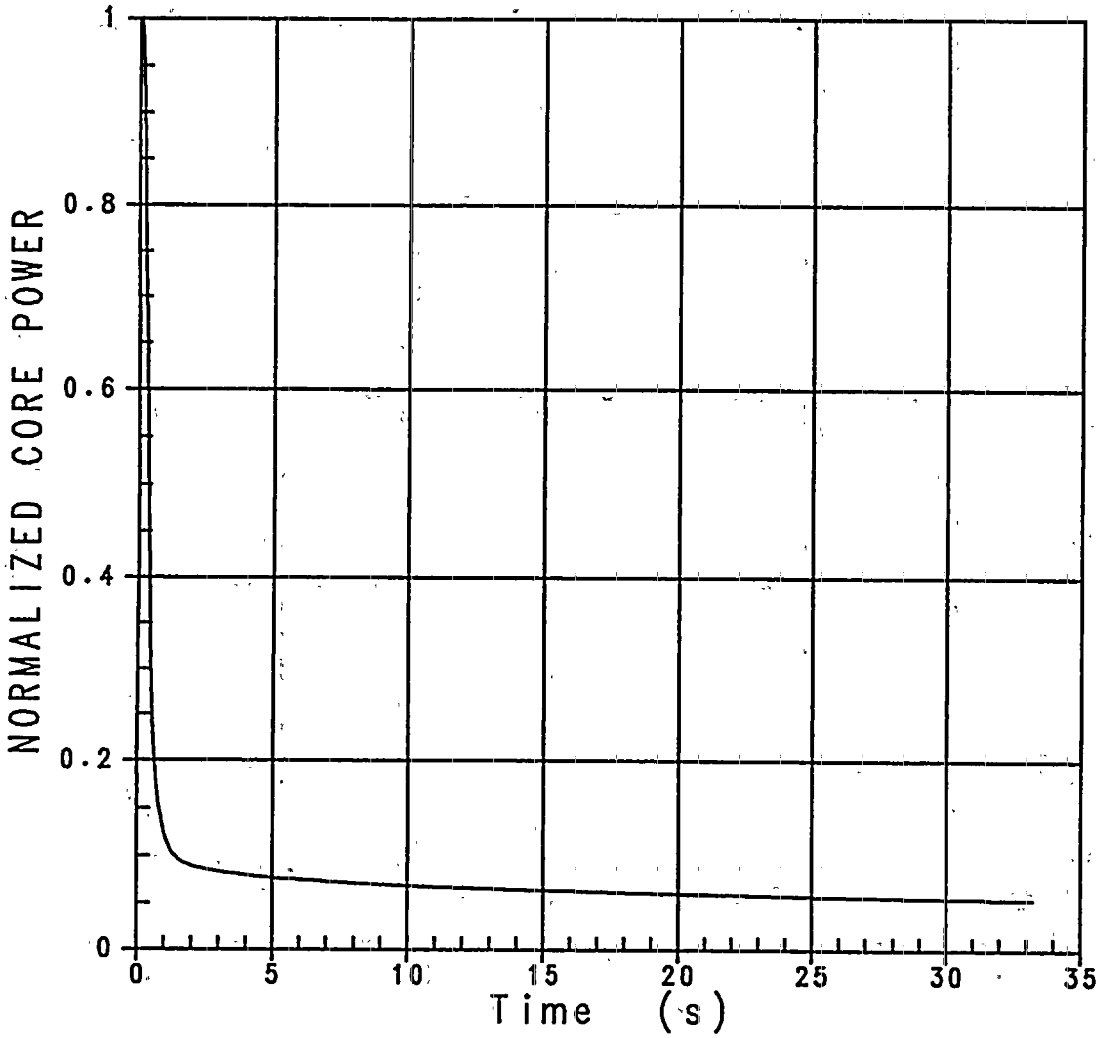

Figure 3.3.1-10: Decay Heat During Blowdown for C_D=0.4, Low T_{avg}

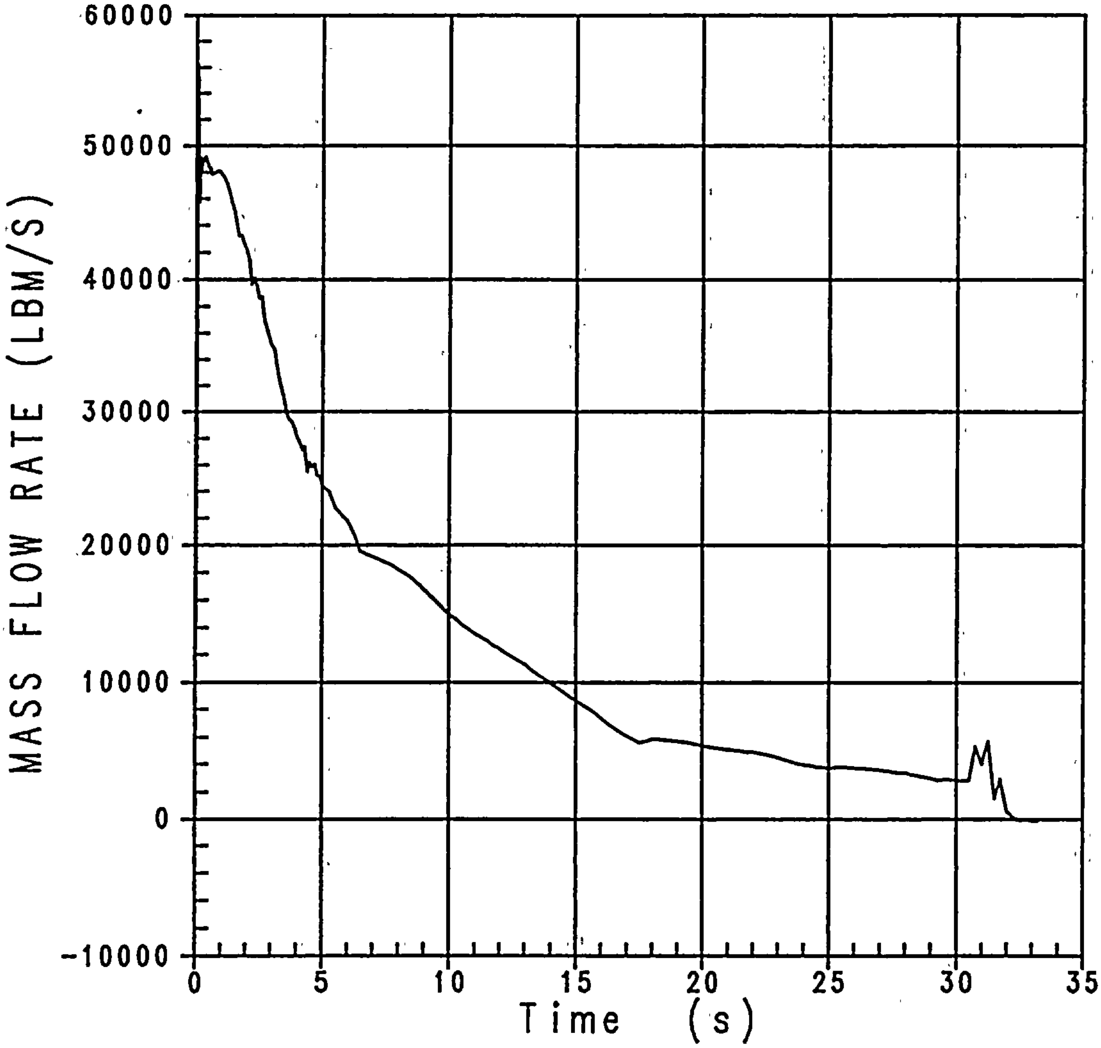

Figure 3.3.1-11: Break Flow During Blowdown for C_D=0.4, Low T_{avg}

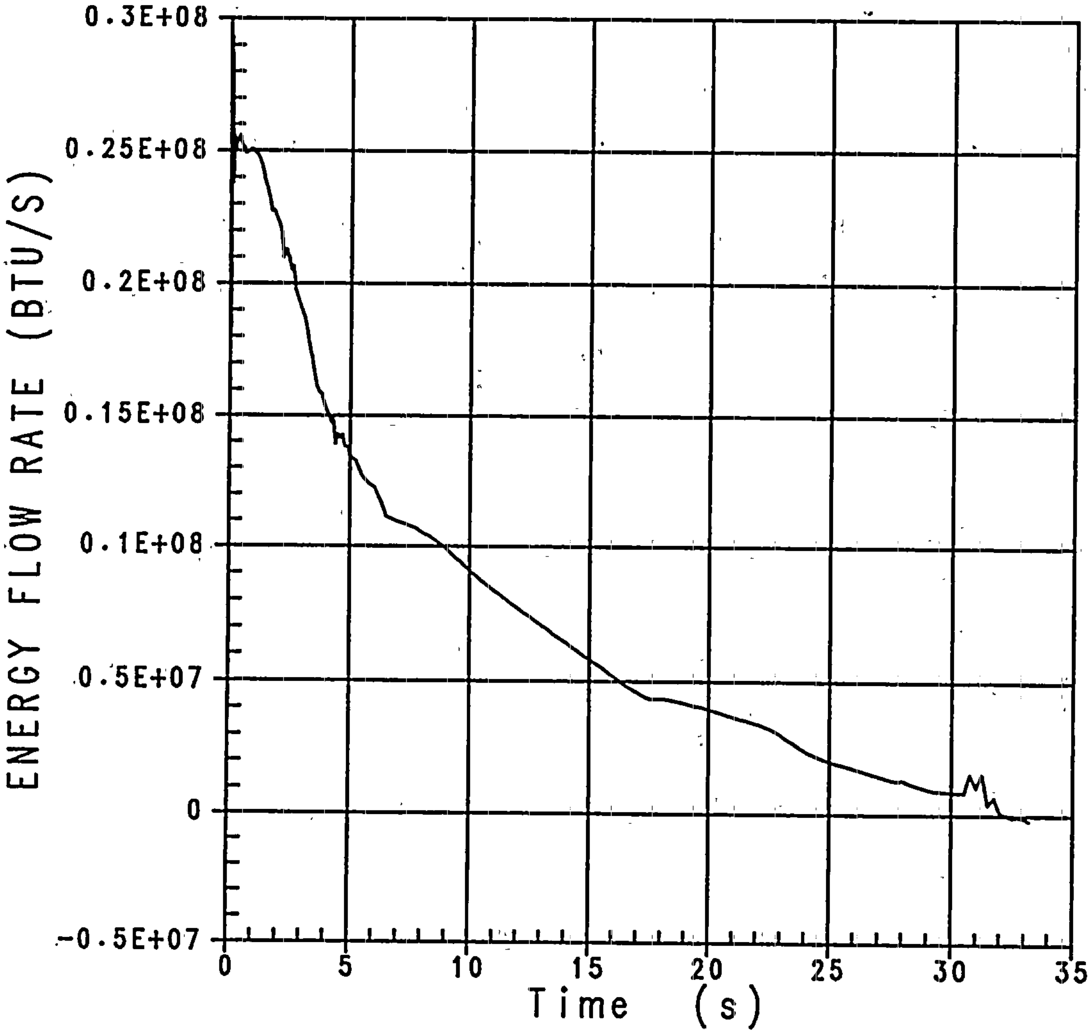

Figure 3.3.1-12: Break Energy During Blowdown for C_D=0.4, Low T_{avg}

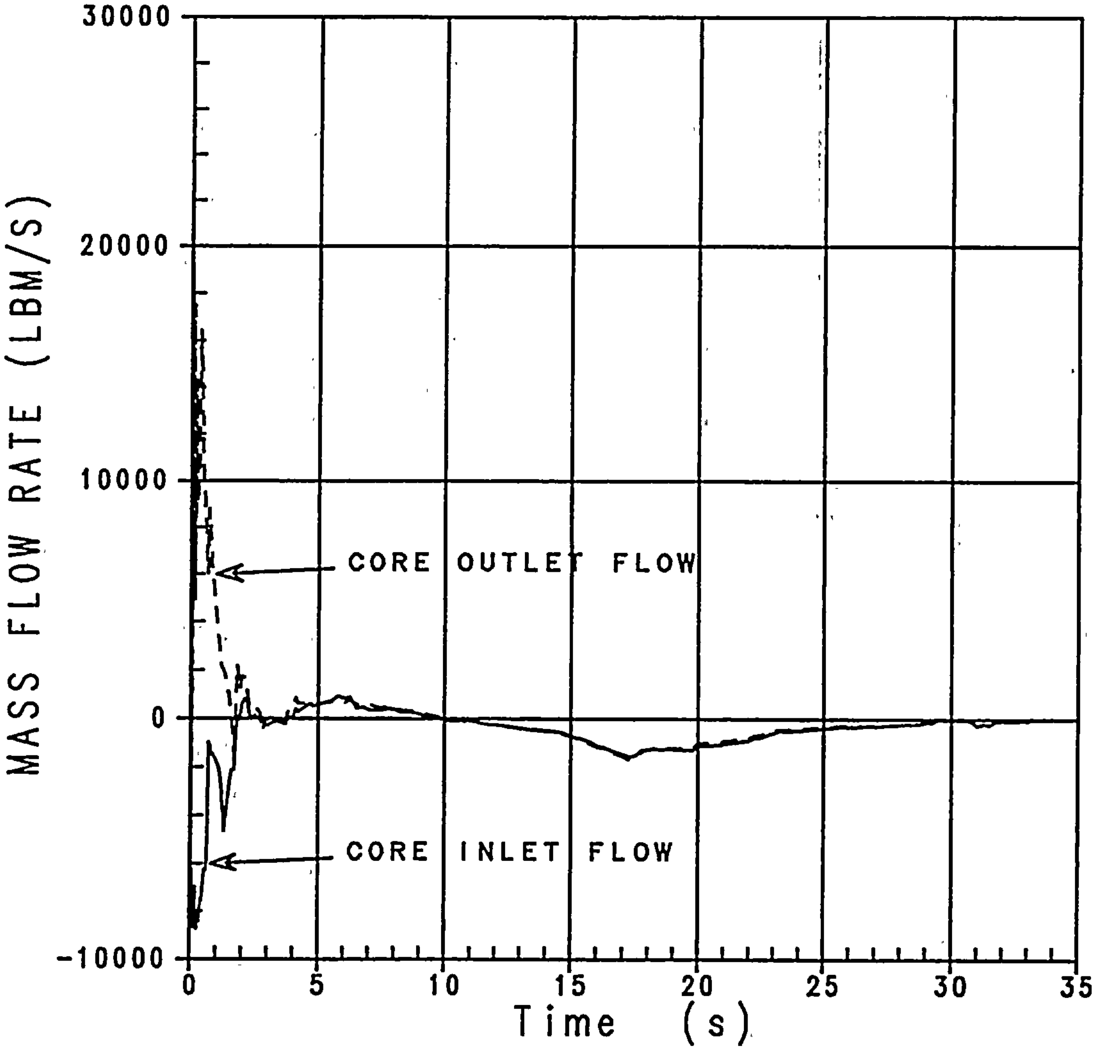

Figure 3.3.1-13: Core Flow During Blowdown for C_D=0.4, Low T_{avg}

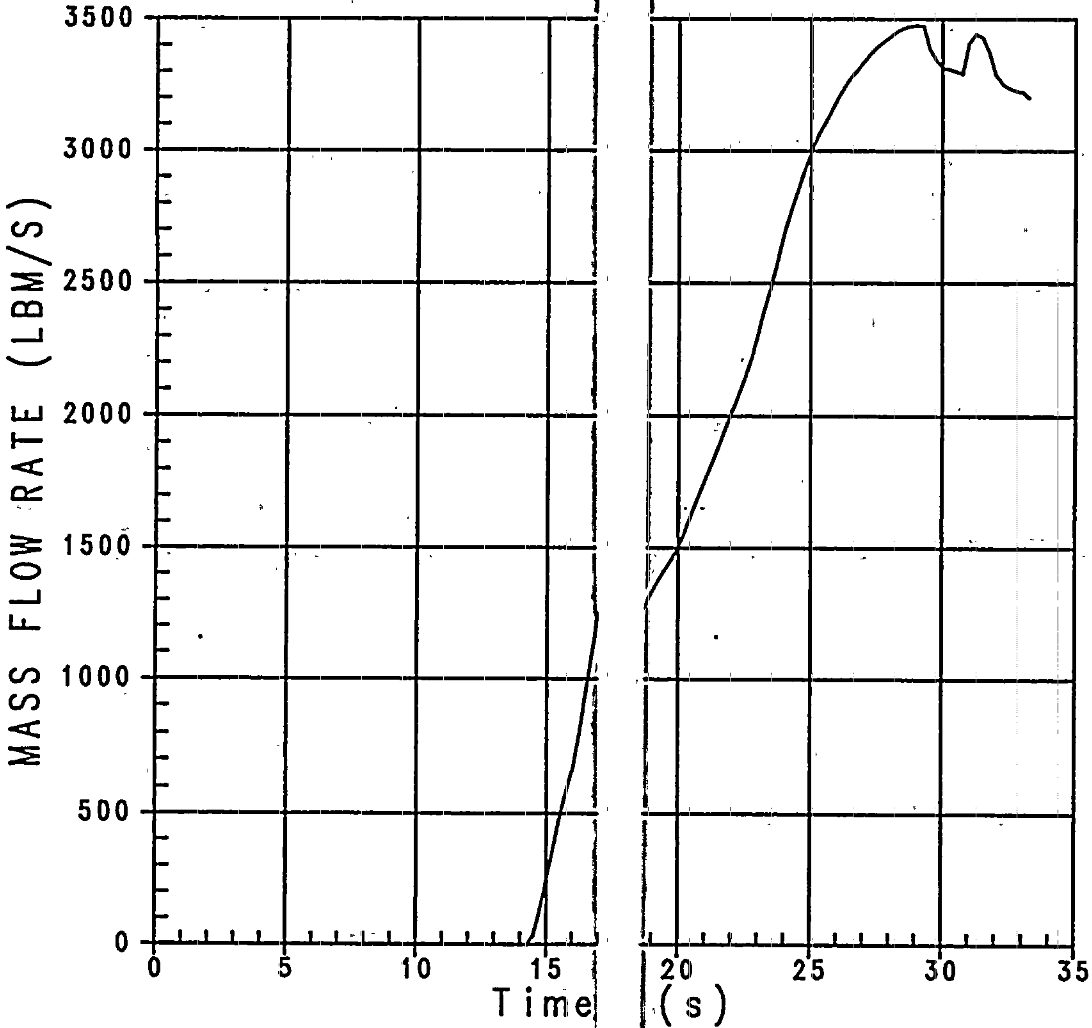

Figure 3.3.1-14: Accumulator Flow During Blowdown for C_D=0.4, Low T_{avg}

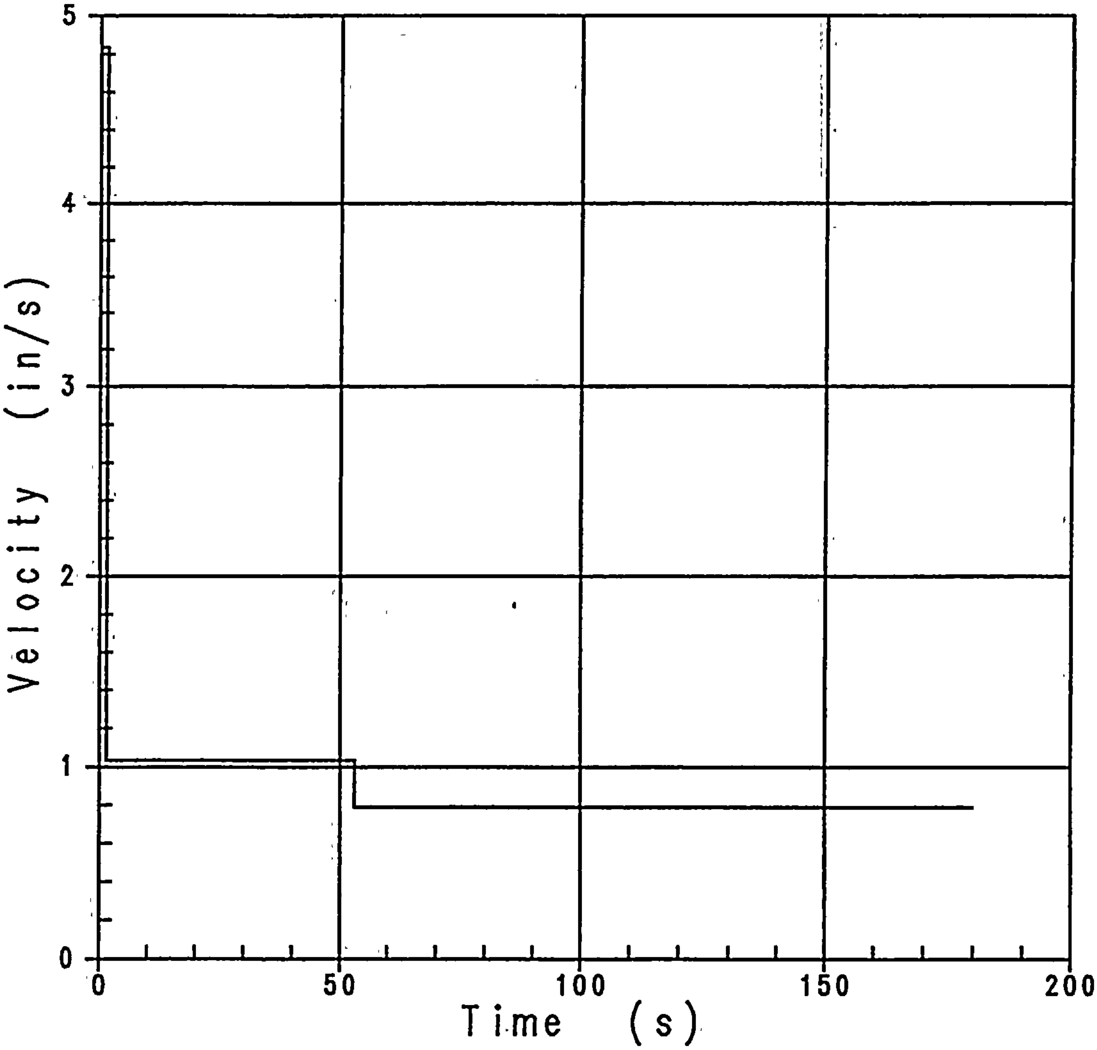

Figure 3.3.1-15: Core Reflooding Rate for C_D=0.4, Low T_{avg}

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.9 seconds after break for C_D=0.4, Low Tavg Case

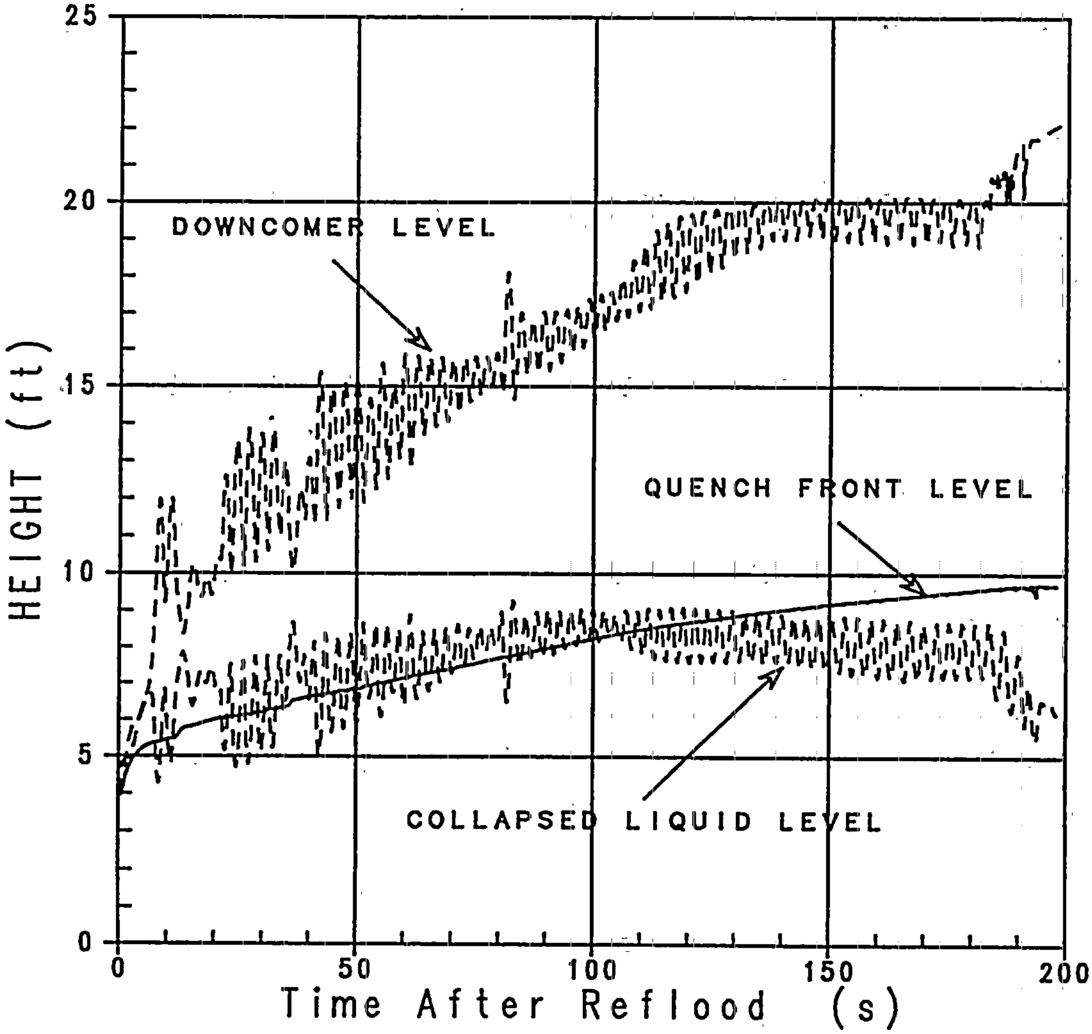

Figure 3.3.1-16: Core and Downcomer Mixture Levels During Reflood for C_D=0.4, Low T_{avg}

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.9 seconds after break for C_D=0.4, Low Tavg Case

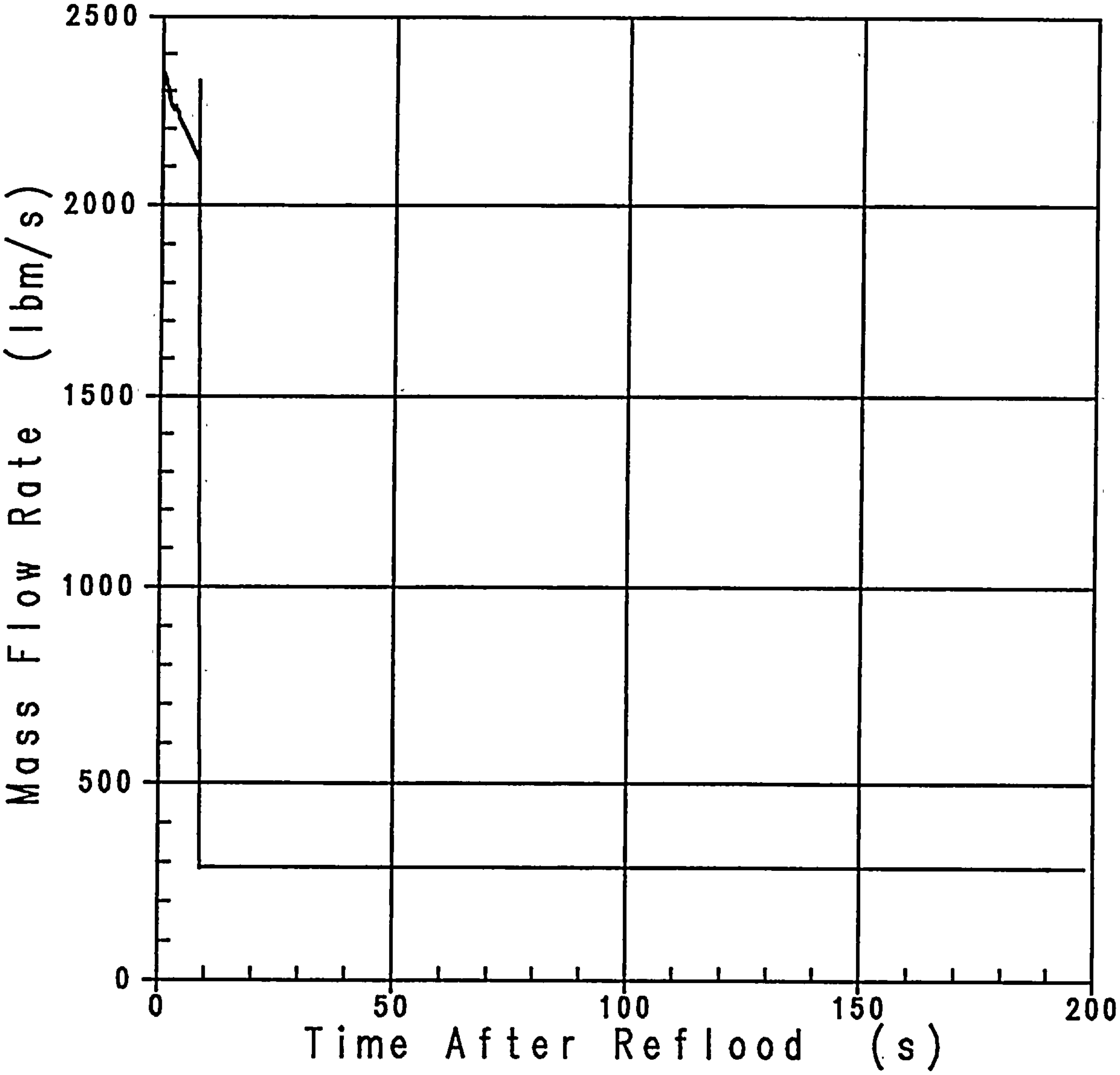

Figure 3.3.1-17: ECCS Flows During Reflood for C_D=0.4, Low T_{avg}

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.9 seconds after break for C_D=0.4, Low Tavg Case

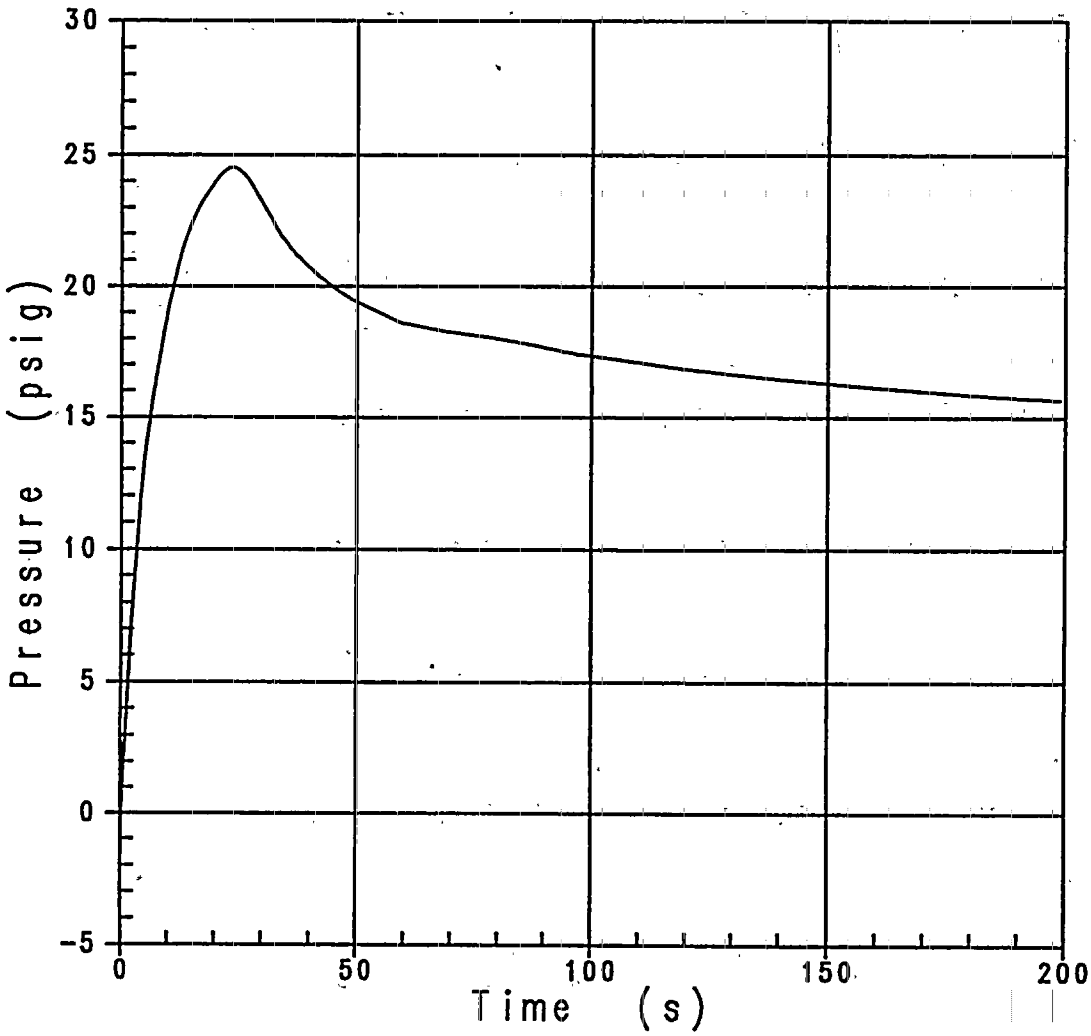

Figure 3.3.1-18: Containment Pressure Transient for C_D=0.4, Low T_{avg}

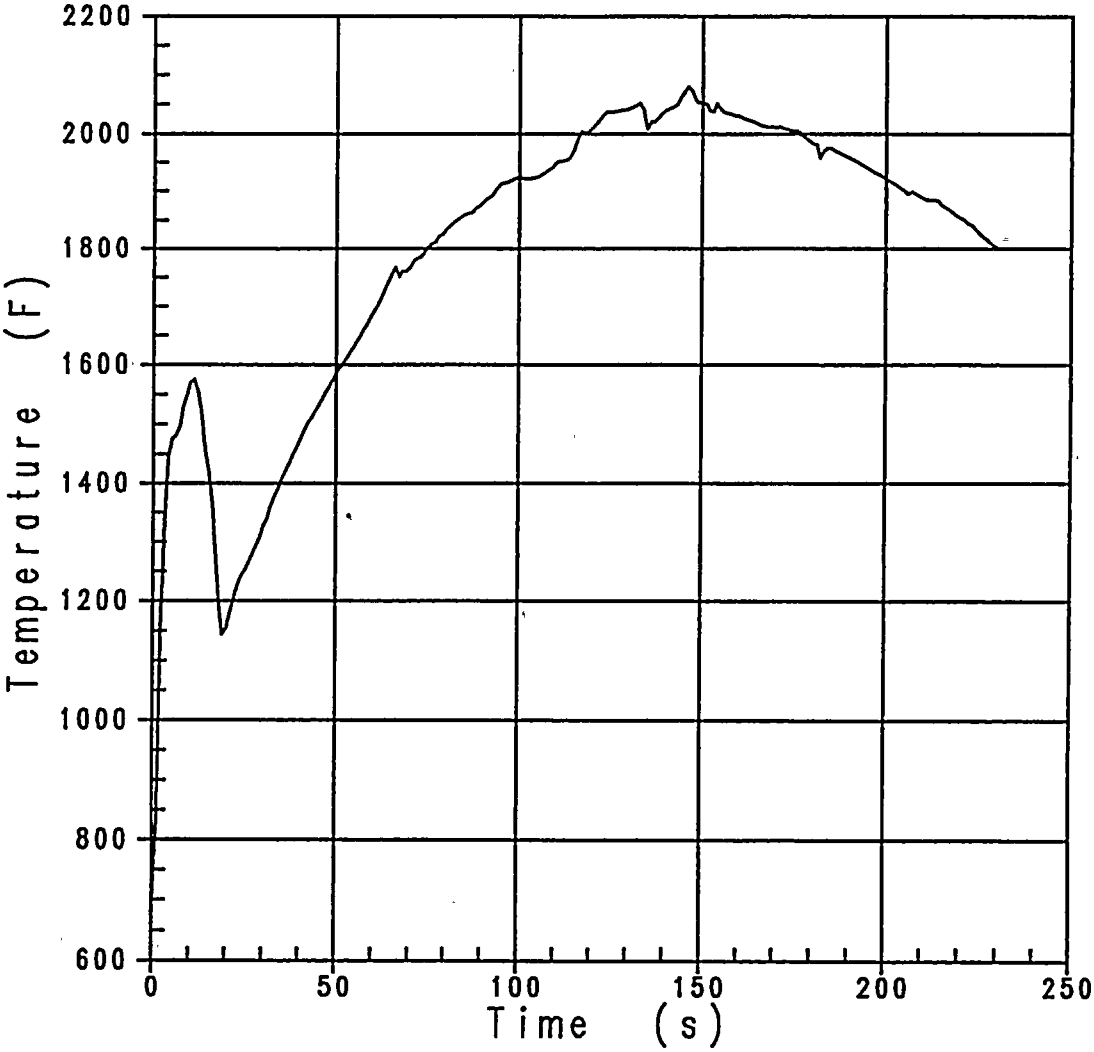

Figure 3.3.1-19: Peak Cladding Temperature for C_D=0.4, High T_{avg}

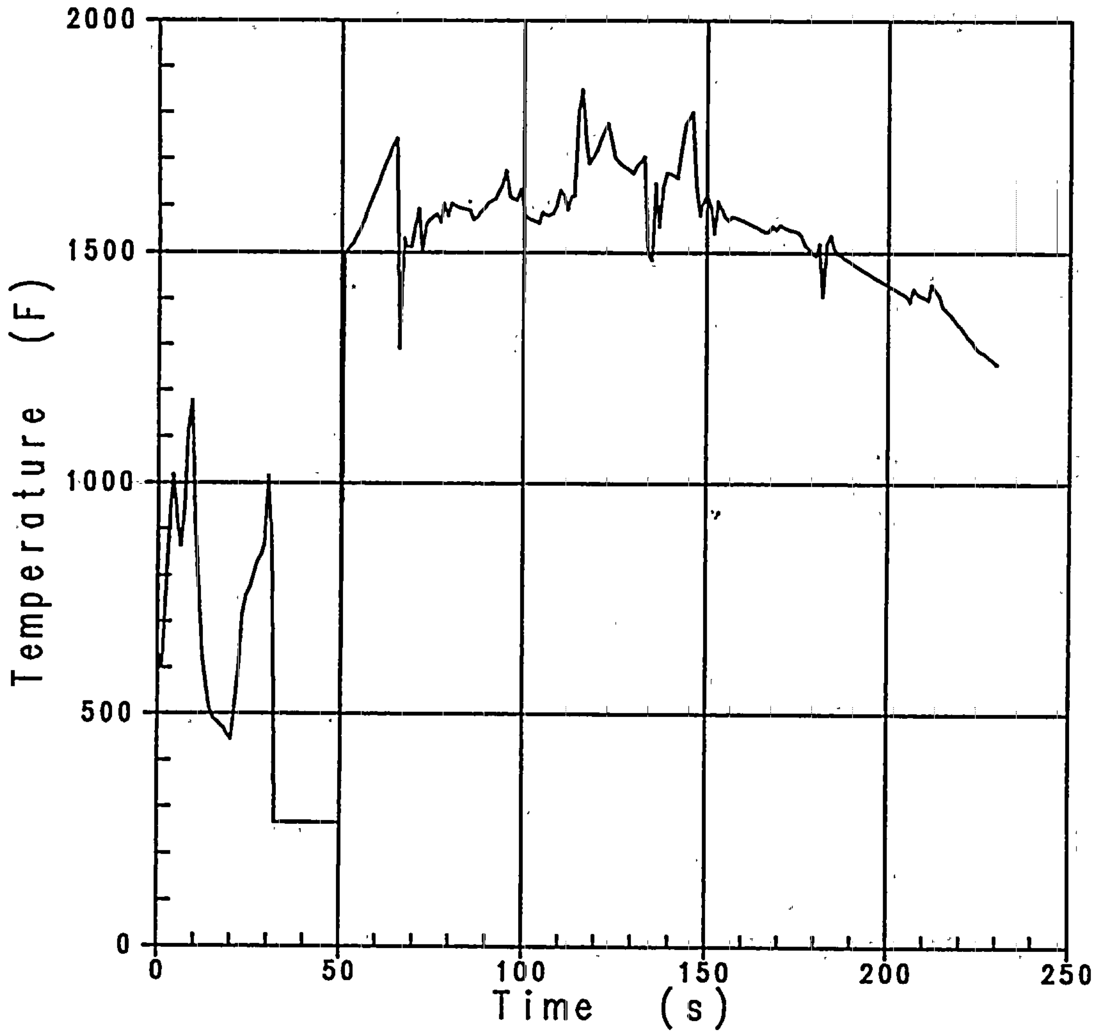

Figure 3.3.1-20: Local Fluid Temperature at PCT Elevation for C_D=0.4, High T_{avg}

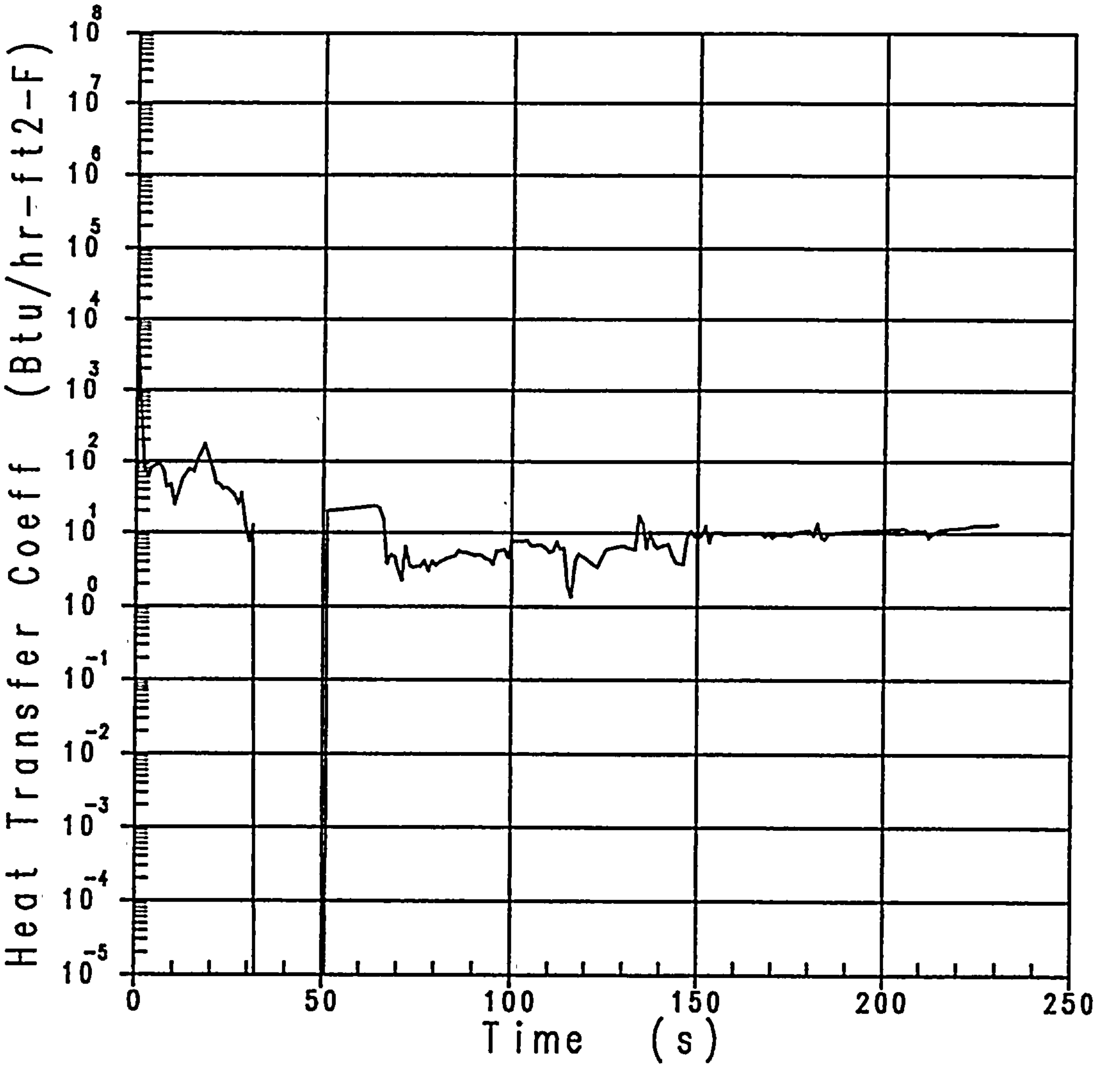

Figure 3.3.1-21: Local Heat Transfer Coefficient at PCT Elevation for C_D=0.4, High T_{avg}

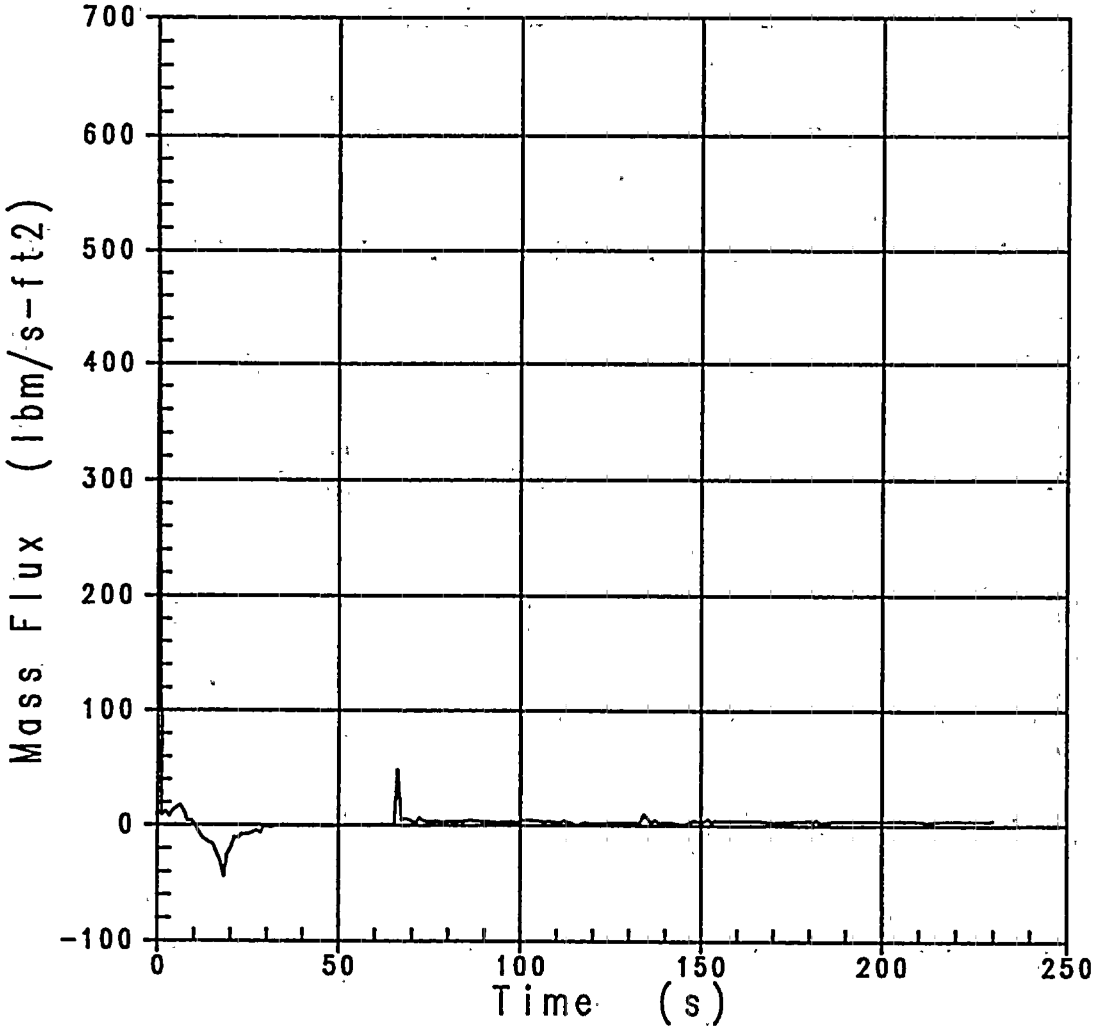

Figure 3.3.1-22: Local Mass Flux at PCT Elevation for C_D=0.4, High T_{avg}

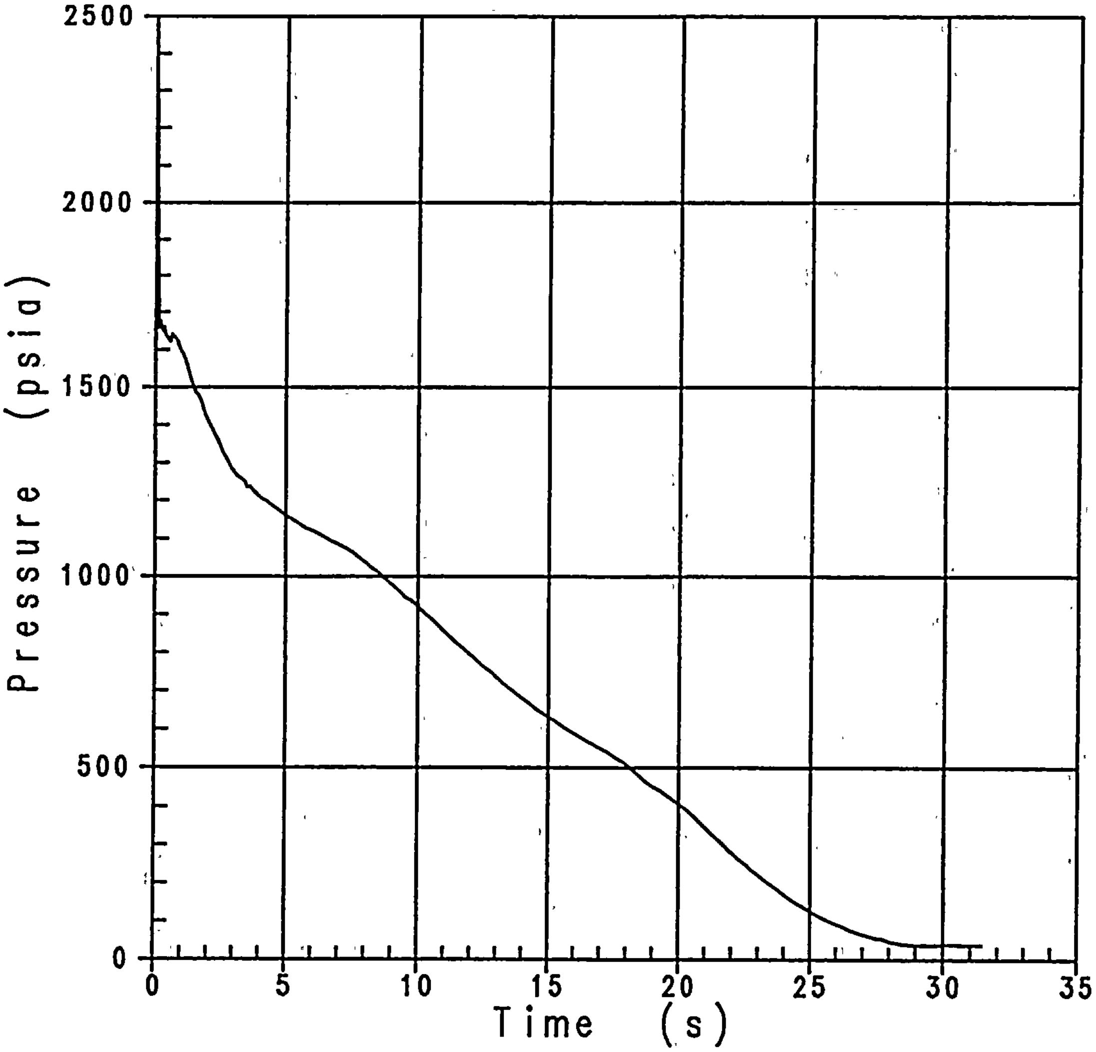

Figure 3.3.1-23: RCS Pressure During Blowdown for C_D=0.4, High T_{avg}

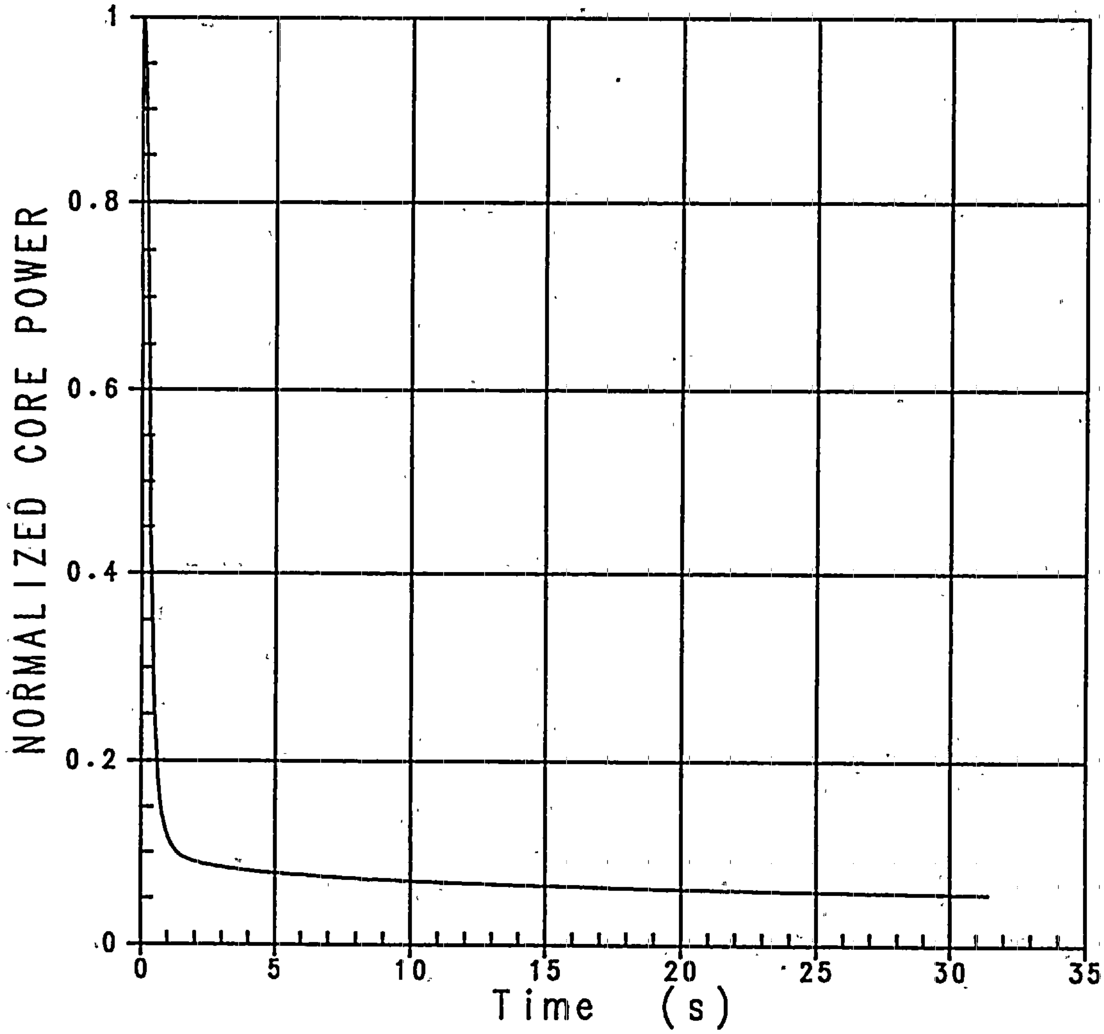

Figure 3.3.1-24: Decay Heat During Blowdown for C_D=0.4, High T_{avg}

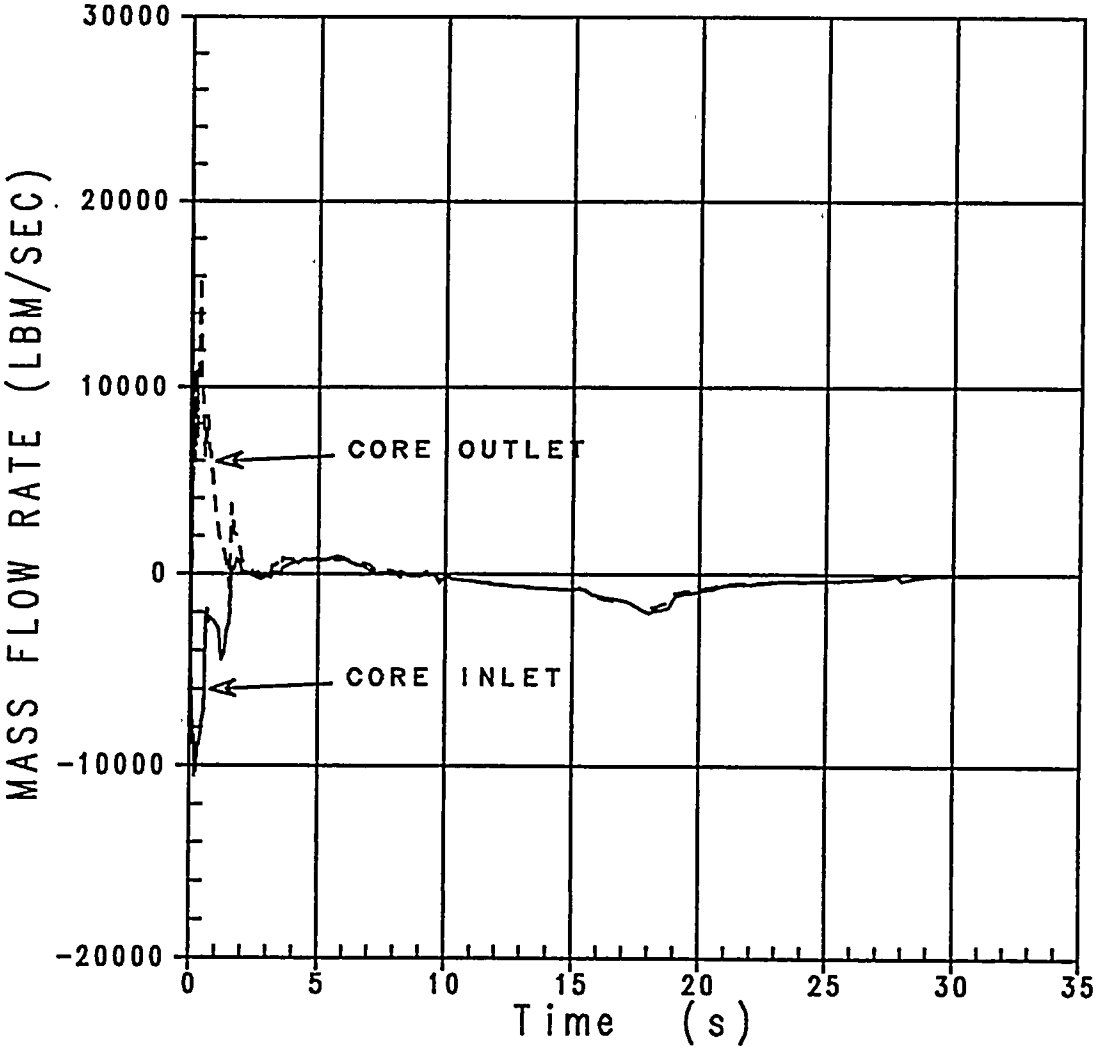

Figure 3.3.1-25: Core Flow During Blowdown for C_D=0.4, High T_{avg}

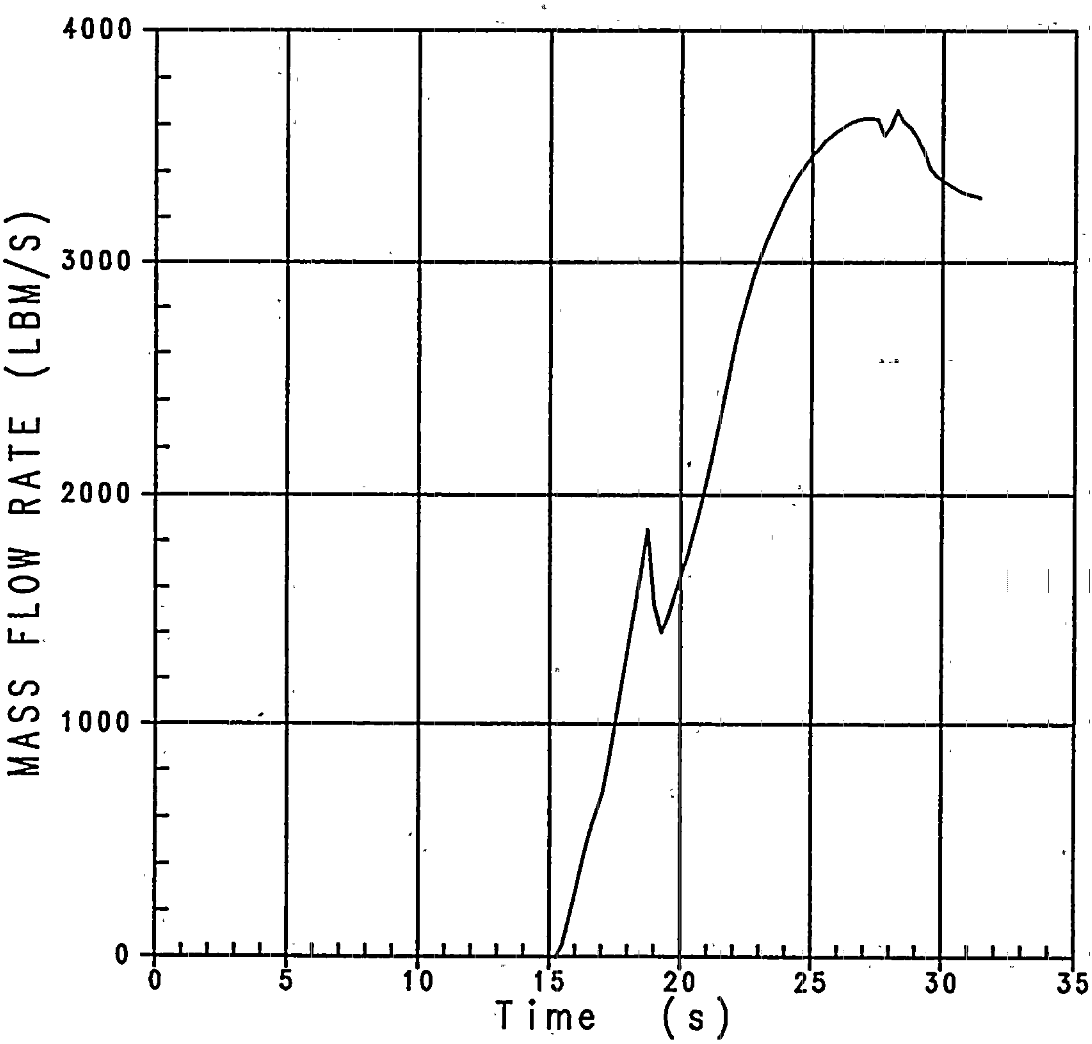

Figure 3.3.1-26: Accumulator Flow During Blowdown for C_D=0.4, High T_{avg}

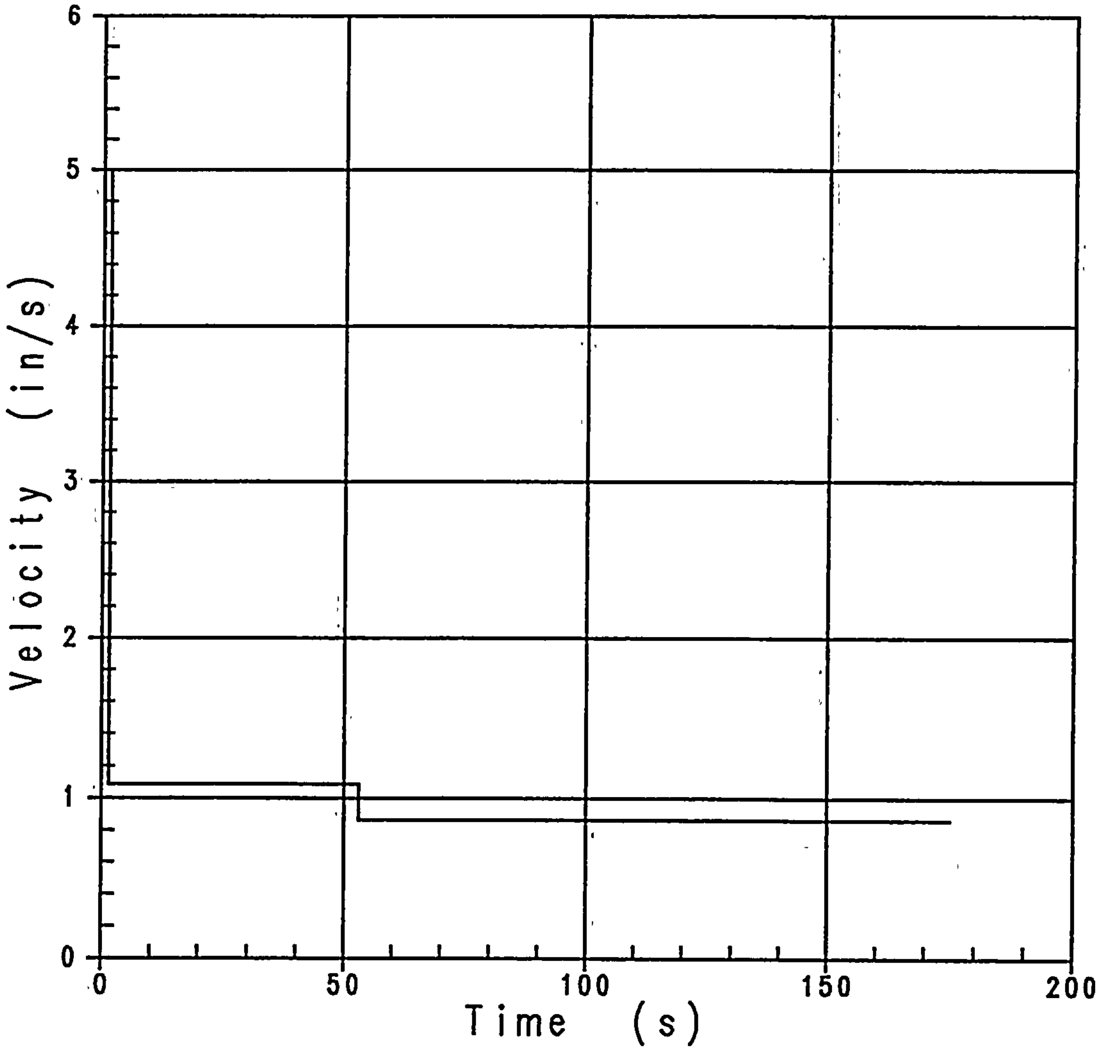

Figure 3.3.1-27: Core Reflooding Rate for C_D=0.4, High T_{avg}

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 50.7 seconds after break for C_D=0.4, High Tavg Case

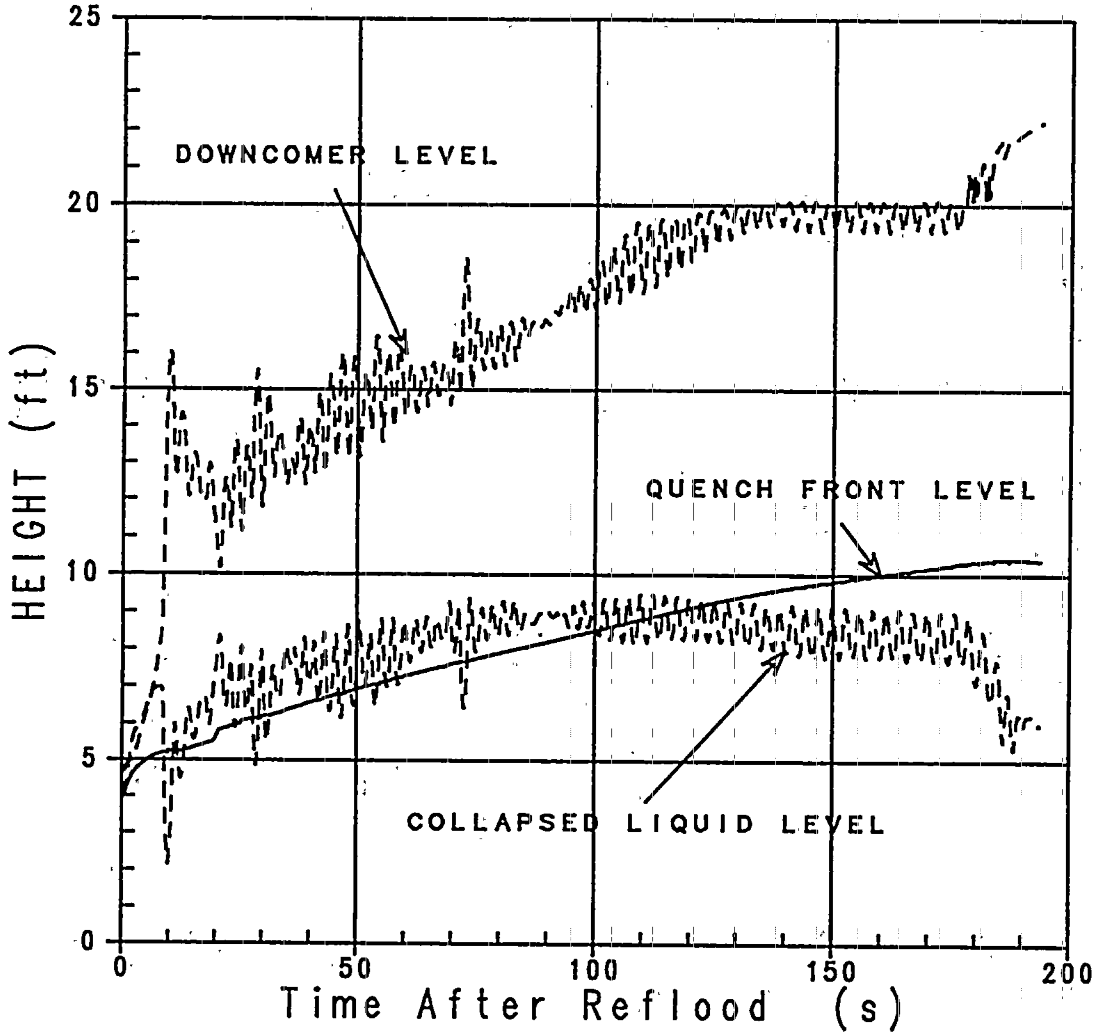

Figure 3.3.1-28: Core and Downcomer Mixture Level During Reflood for C_D=0.4, High T_{avg}

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 50.7 seconds after break for C_D=0.4, High Tavg Case

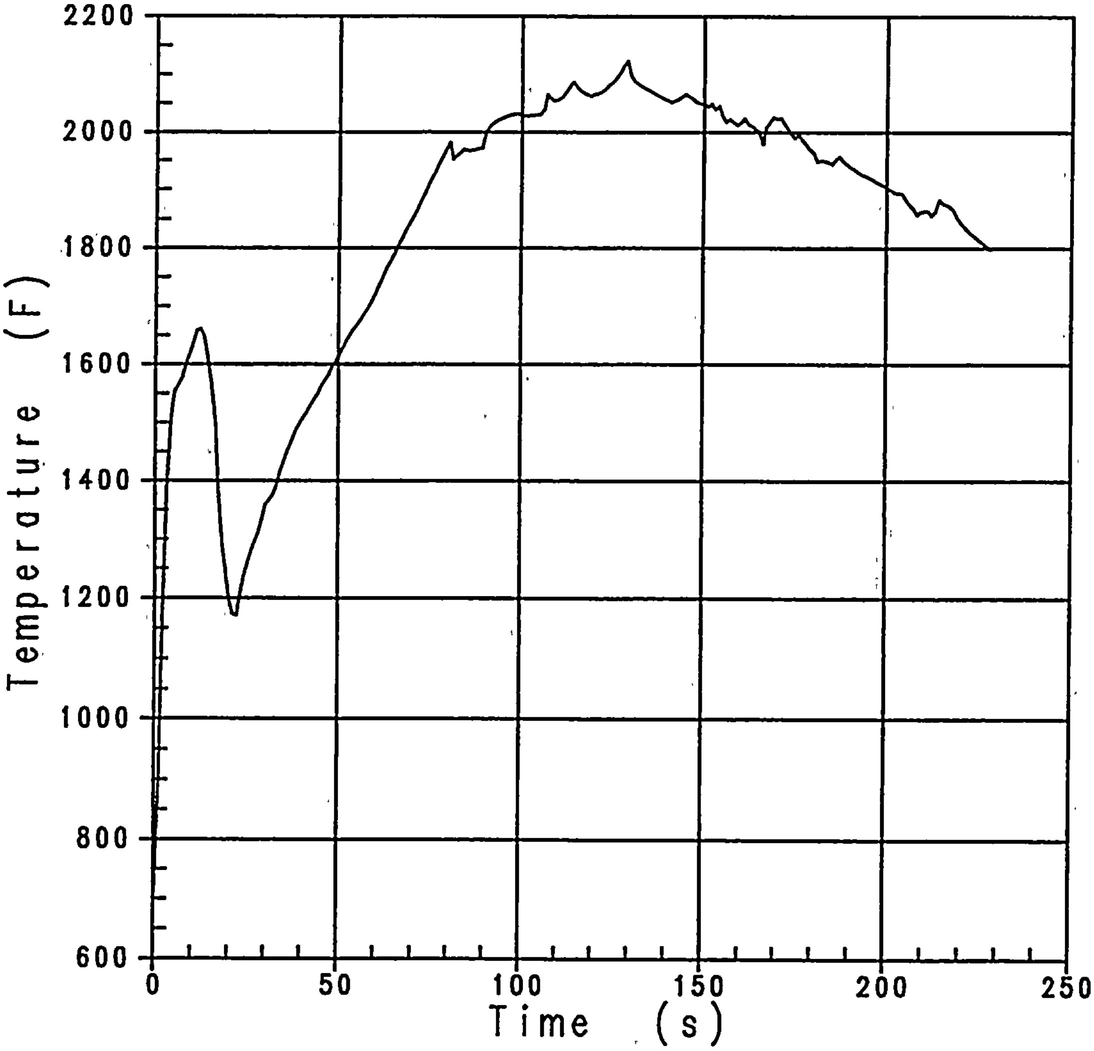

Figure 3.3.1-29: Peak Cladding Temperature for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

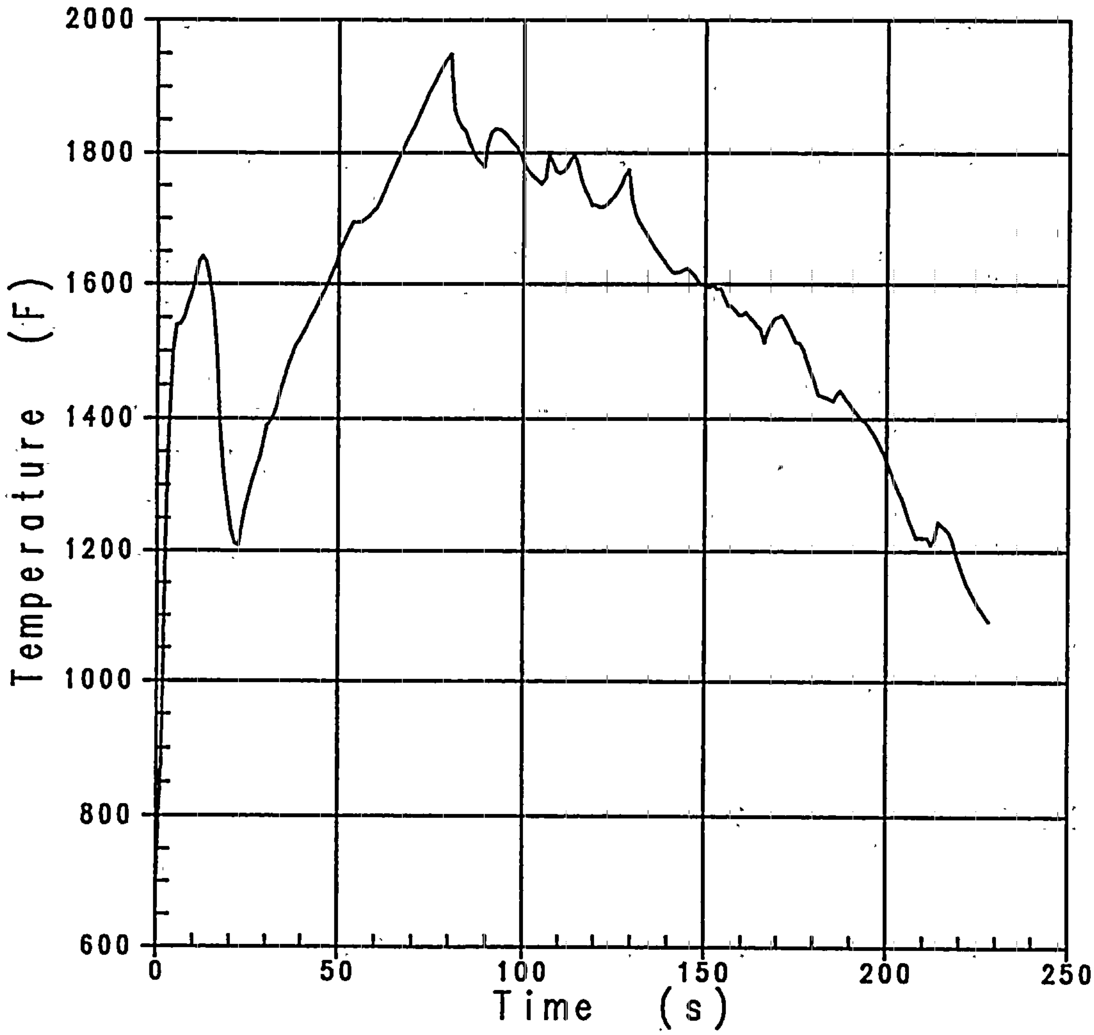

Figure 3.3.1-30: Cladding Temperature at Fuel Rod Burst Location for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

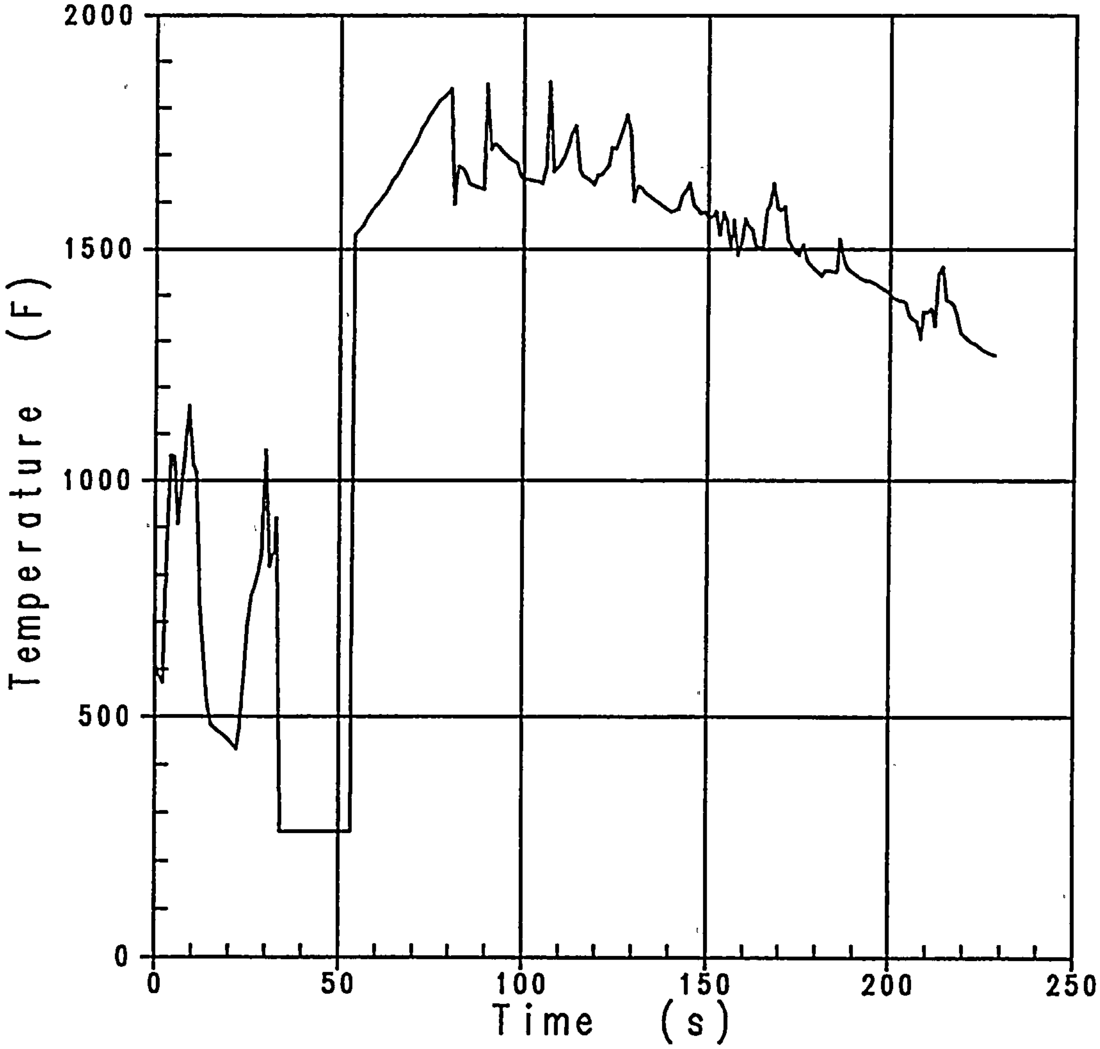

Figure 3.3.1-31: Local Fluid Temperature at PCT Elevation for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

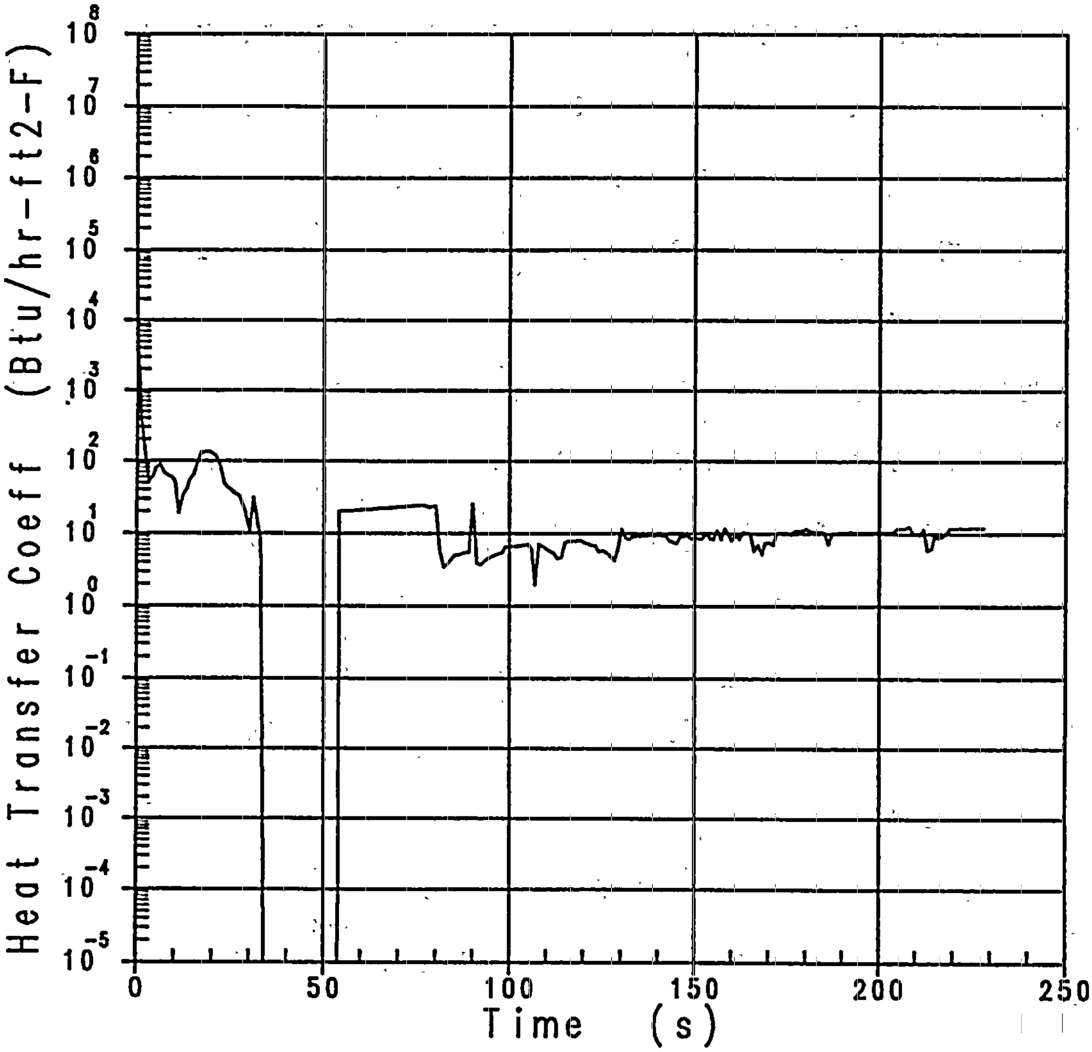

Figure 3.3.1-32: Local Heat Transfer Coefficient at PCT Elevation for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

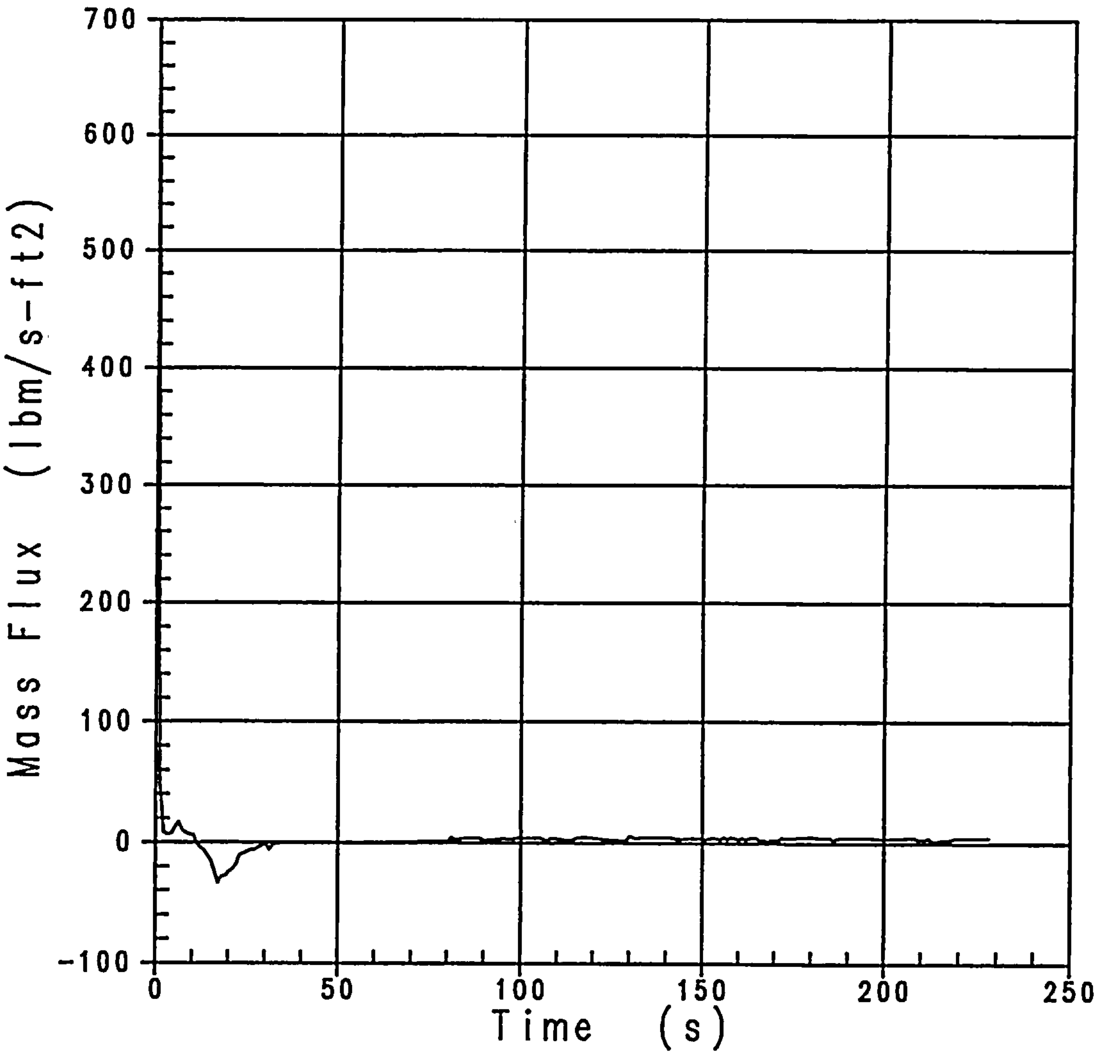

Figure 3.3.1-33: Local Mass Flux at PCT Elevation for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

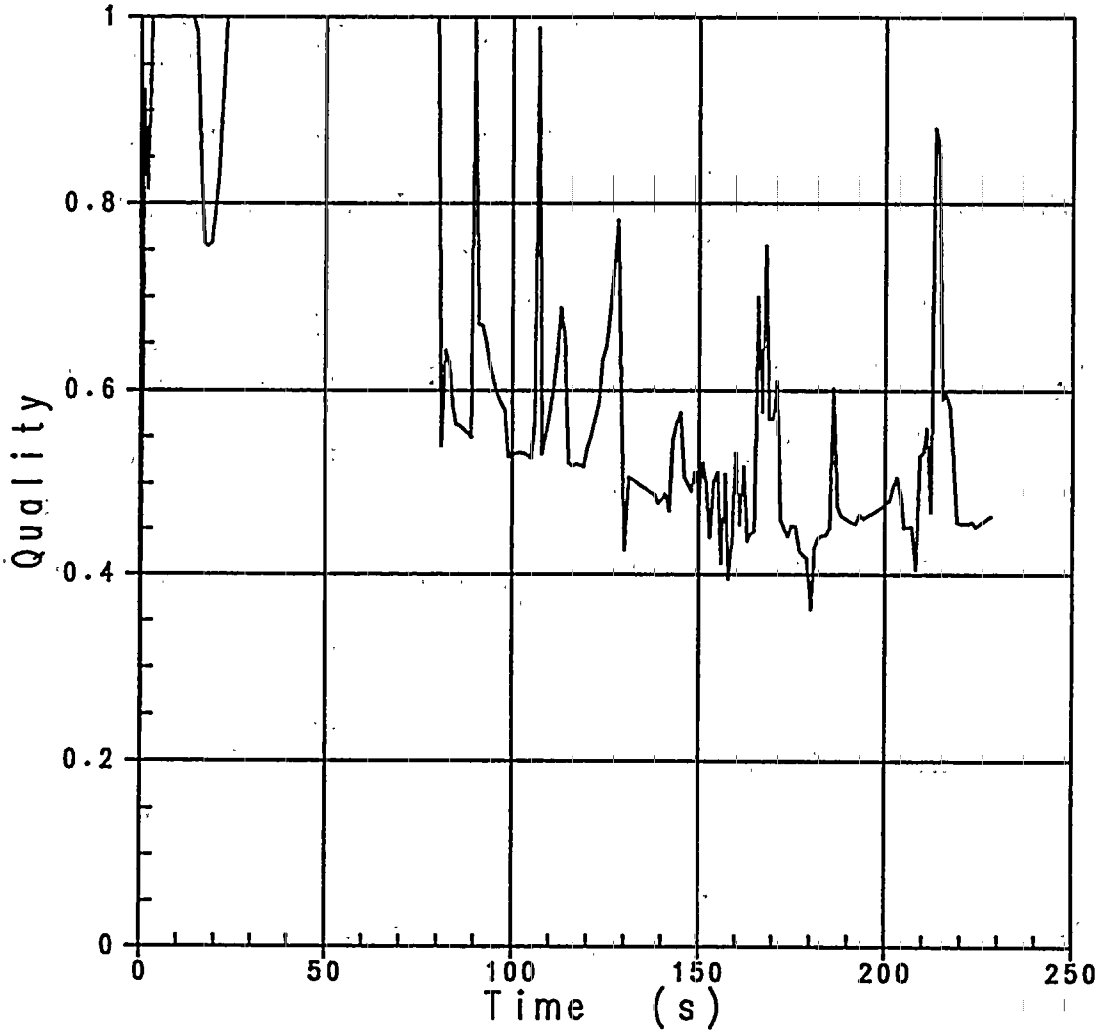

Figure 3.3.1-34: Local Quality at PCT Elevation for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

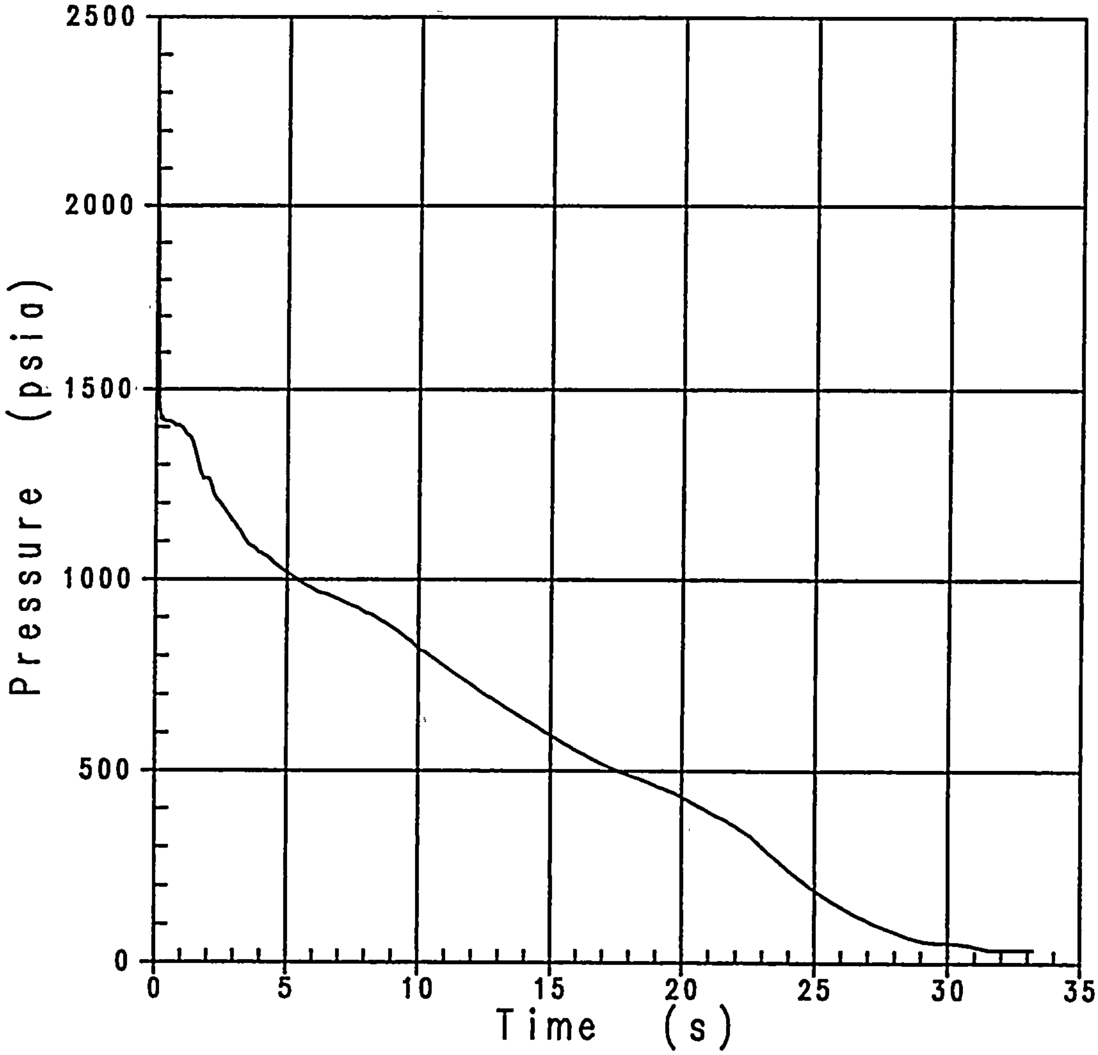

Figure 3.3.1-35: RCS Pressure During Blowdown for C_D=0.4, Low T_{avg},
8.5 ft Skewed Power Shape

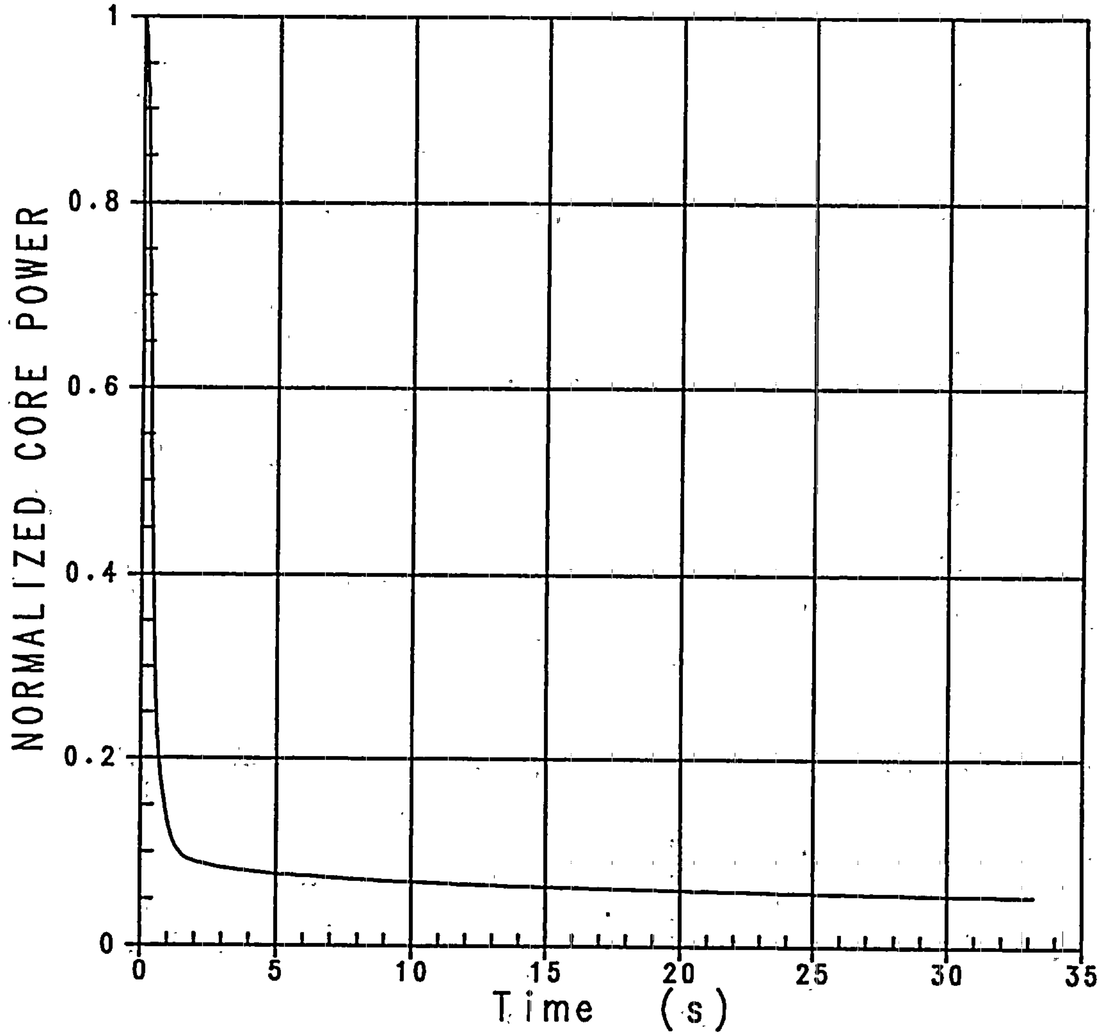

Figure 3.3.1-36: Decay Heat During Blowdown for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

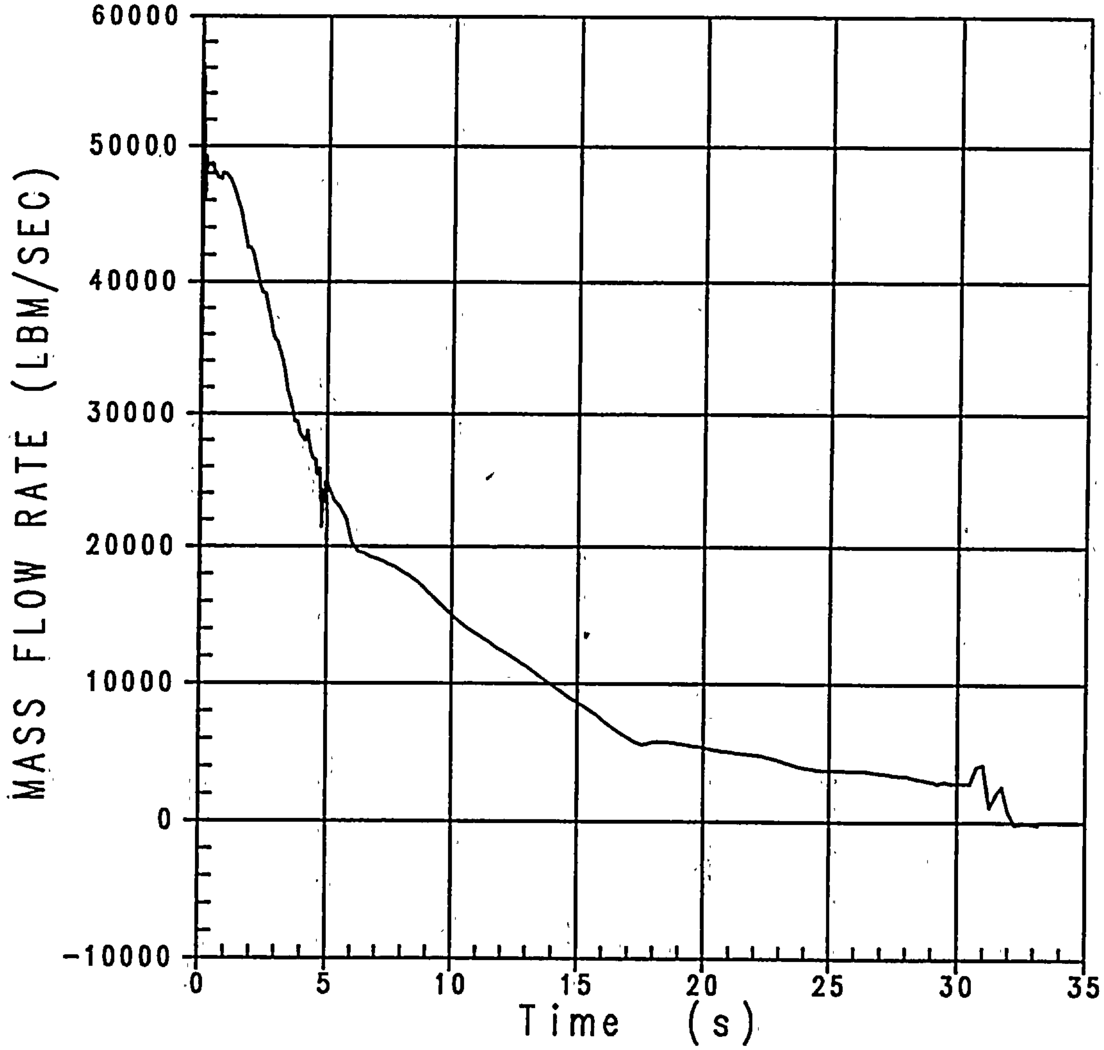

Figure 3.3.1-37: Break Flow During Blowdown for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

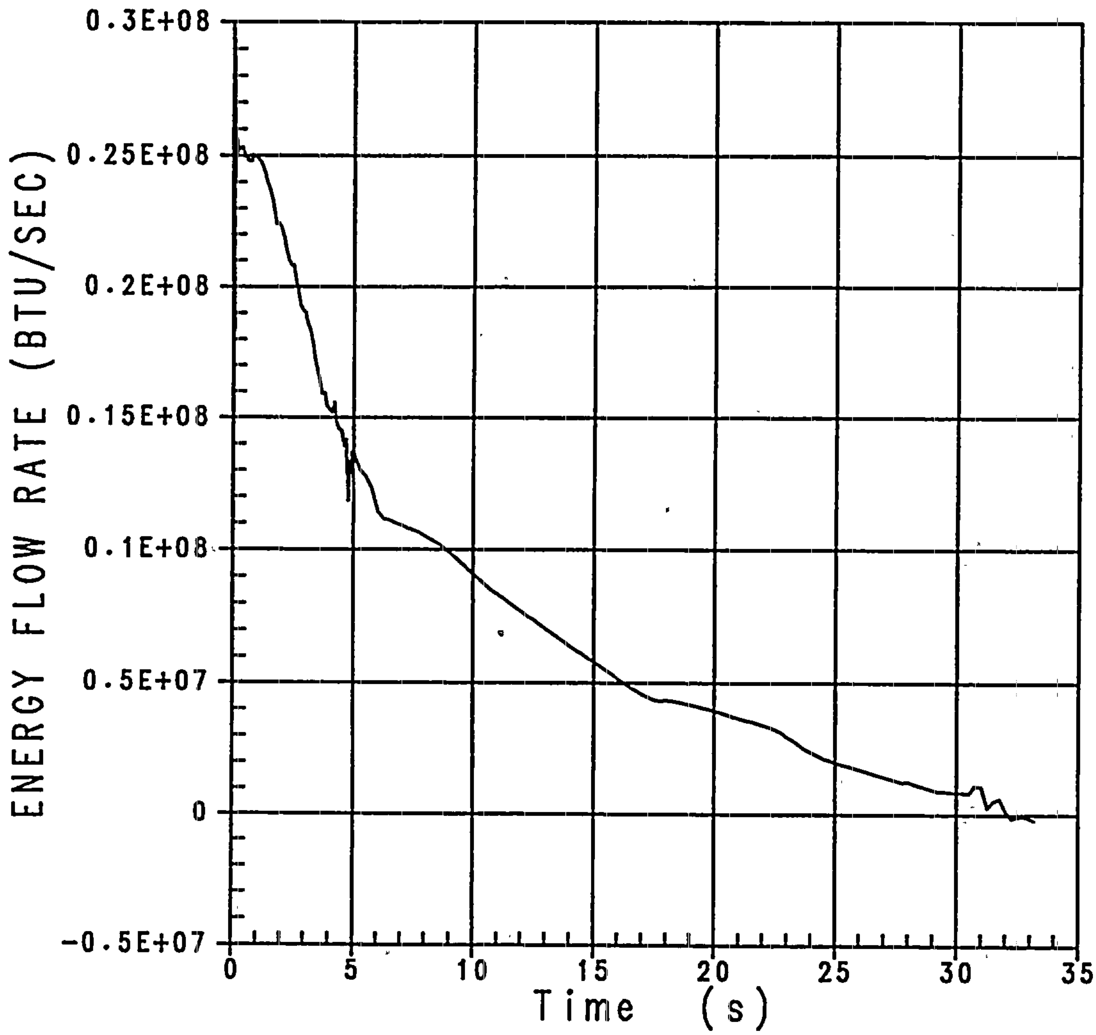

Figure 3.3.1-38: Break Energy During Blowdown for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

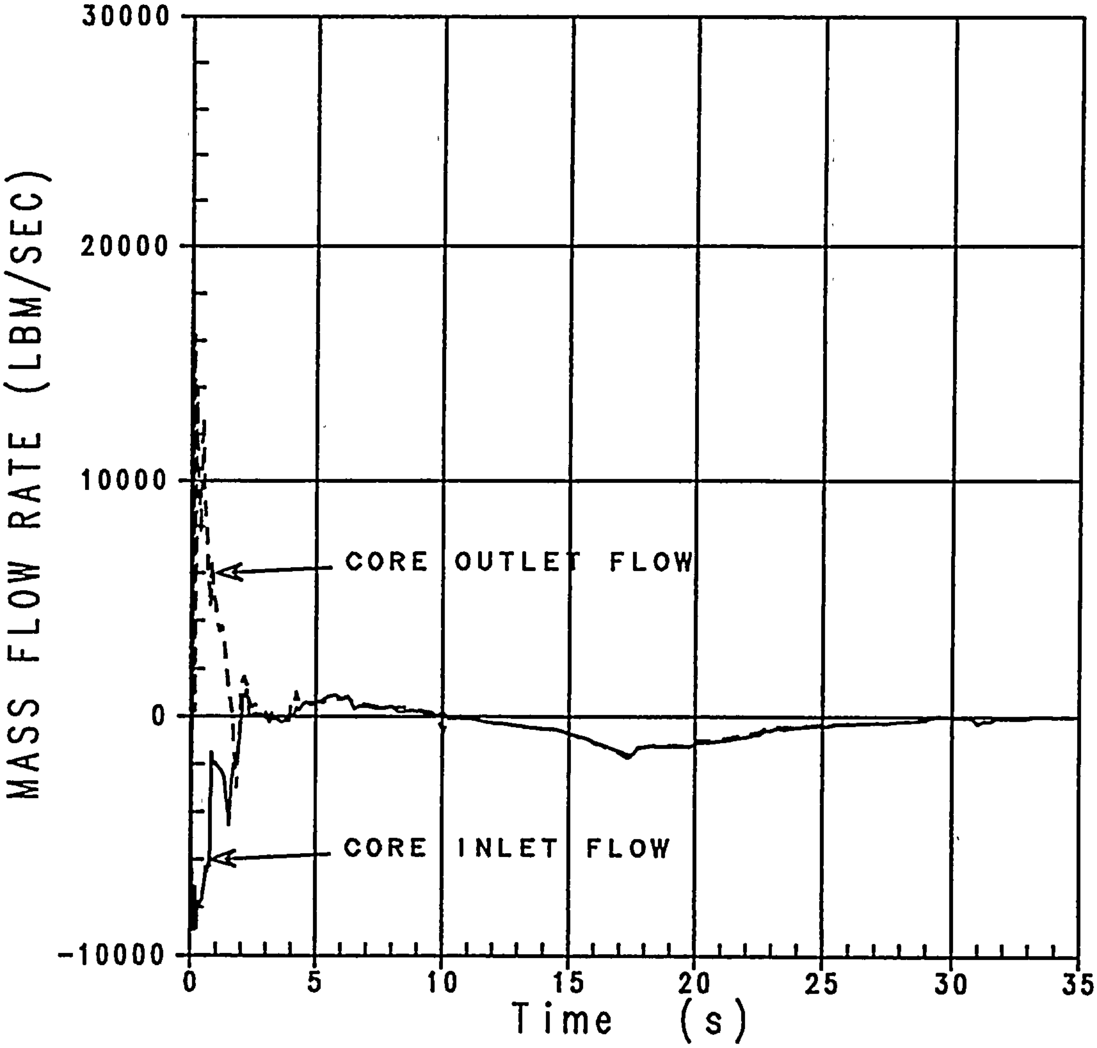

Figure 3.3.1-39: Core Flow During Blowdown for C_D=0.4, Low T_{avg},
8.5 ft Skewed Power Shape

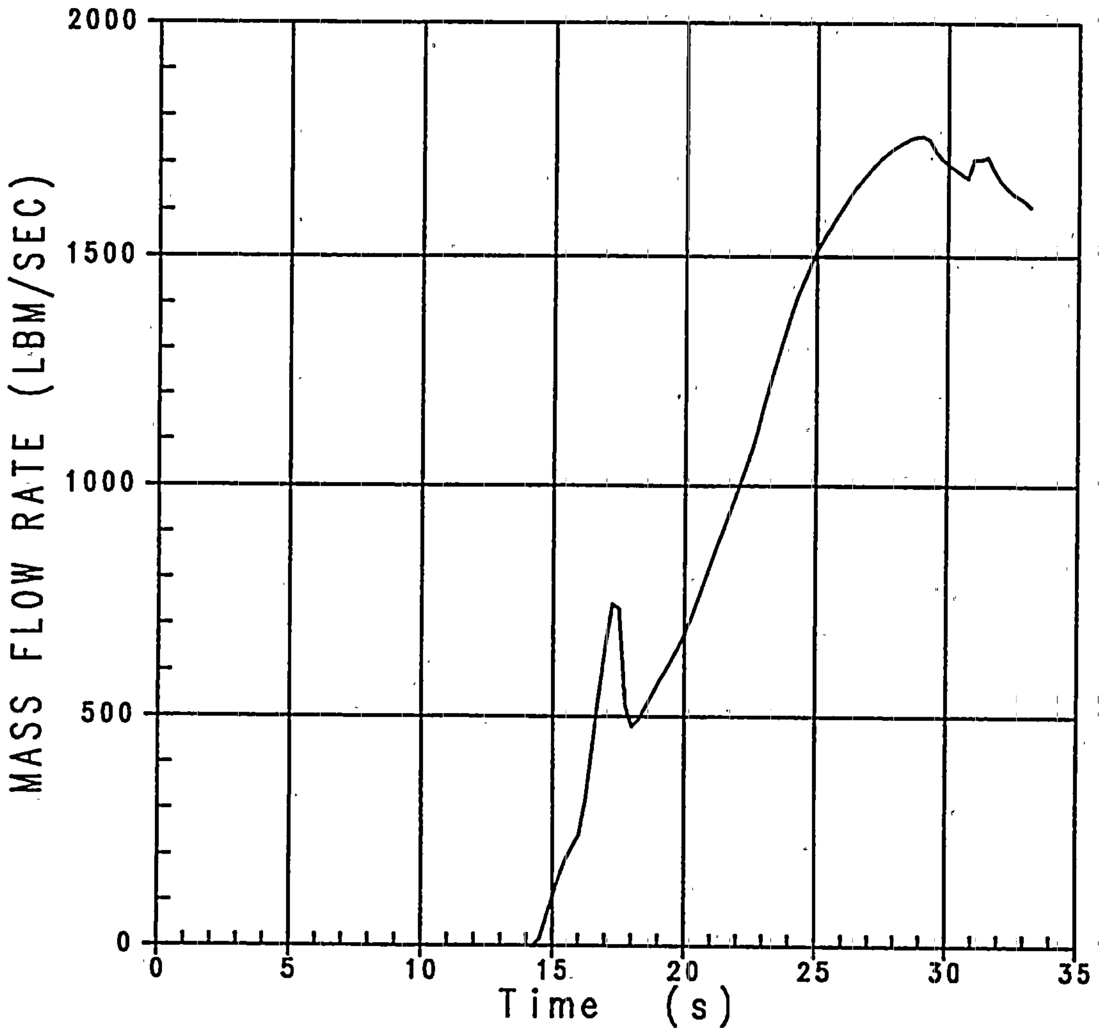

Figure 3.3.1-40: Accumulator Flow During Blowdown for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

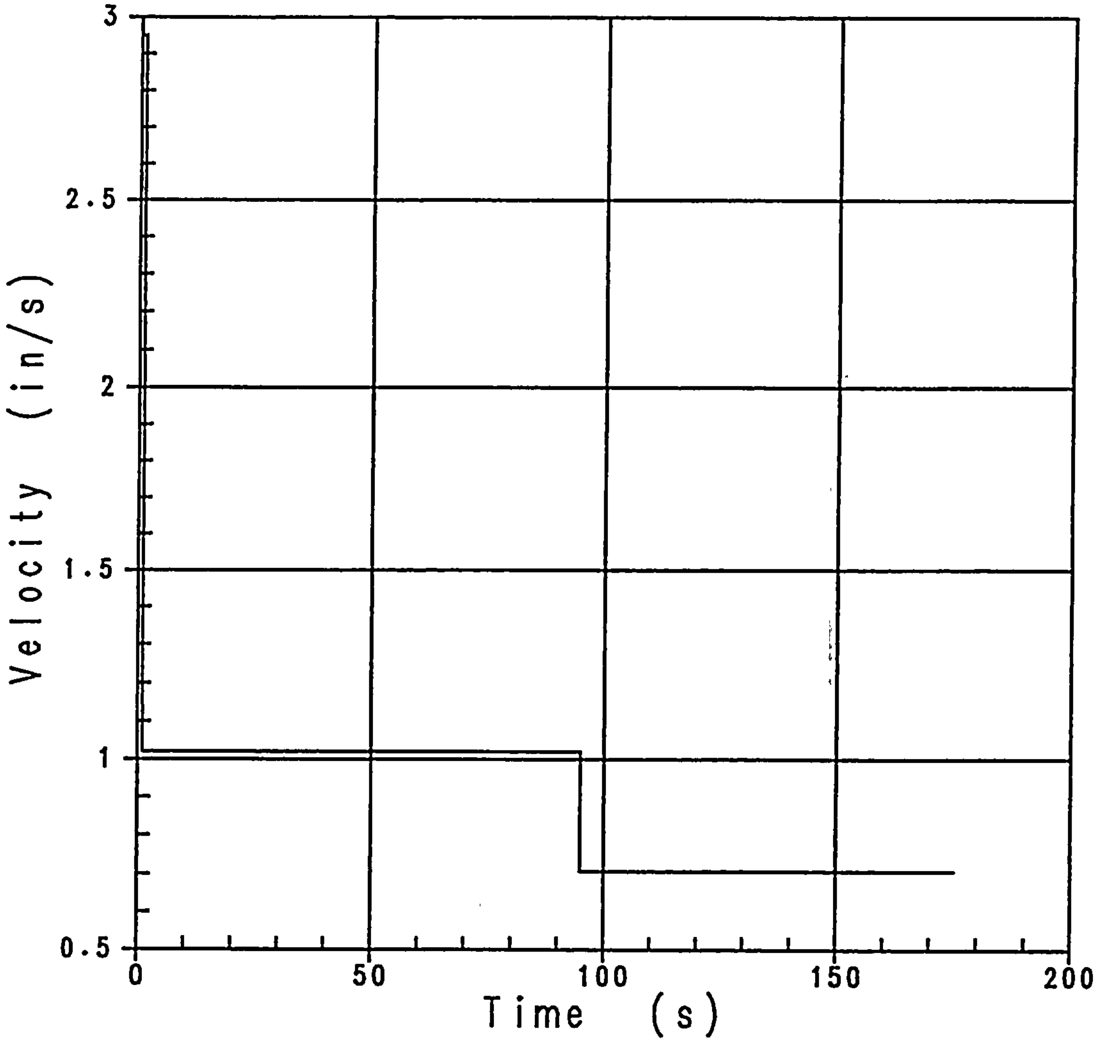

Figure 3.3.1-41: Core Reflooding Rate for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.8 seconds after break for C_D=0.4, Low Tavg, 8.5 ft Skewed Power Shape Case

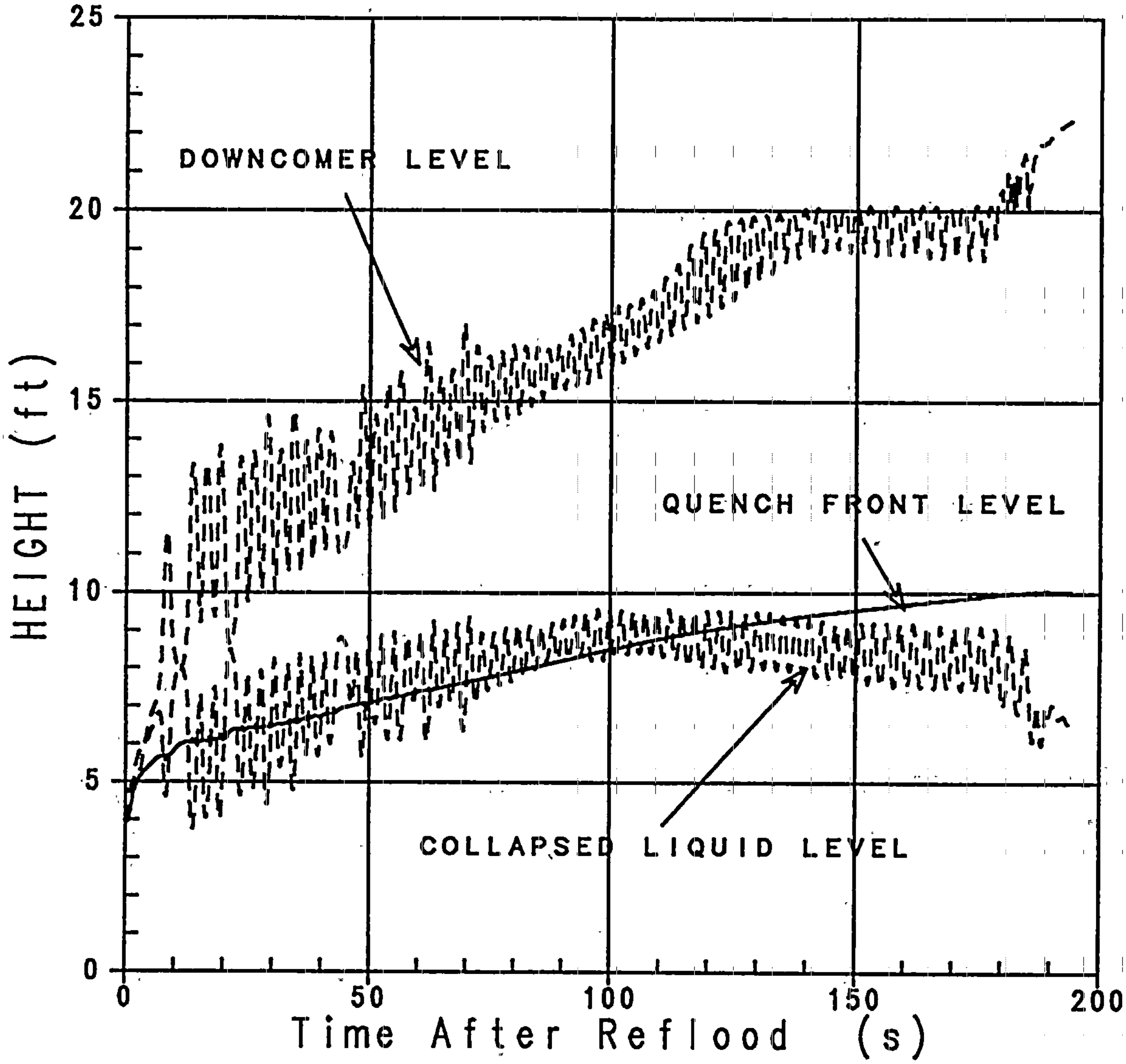

Figure 3.3.1-42: Core and Downcomer Mixture Levels During Reflood for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.8 seconds after break for C_D= 0.4, Low Tavg, 8.5 ft Skewed Power Shape Case

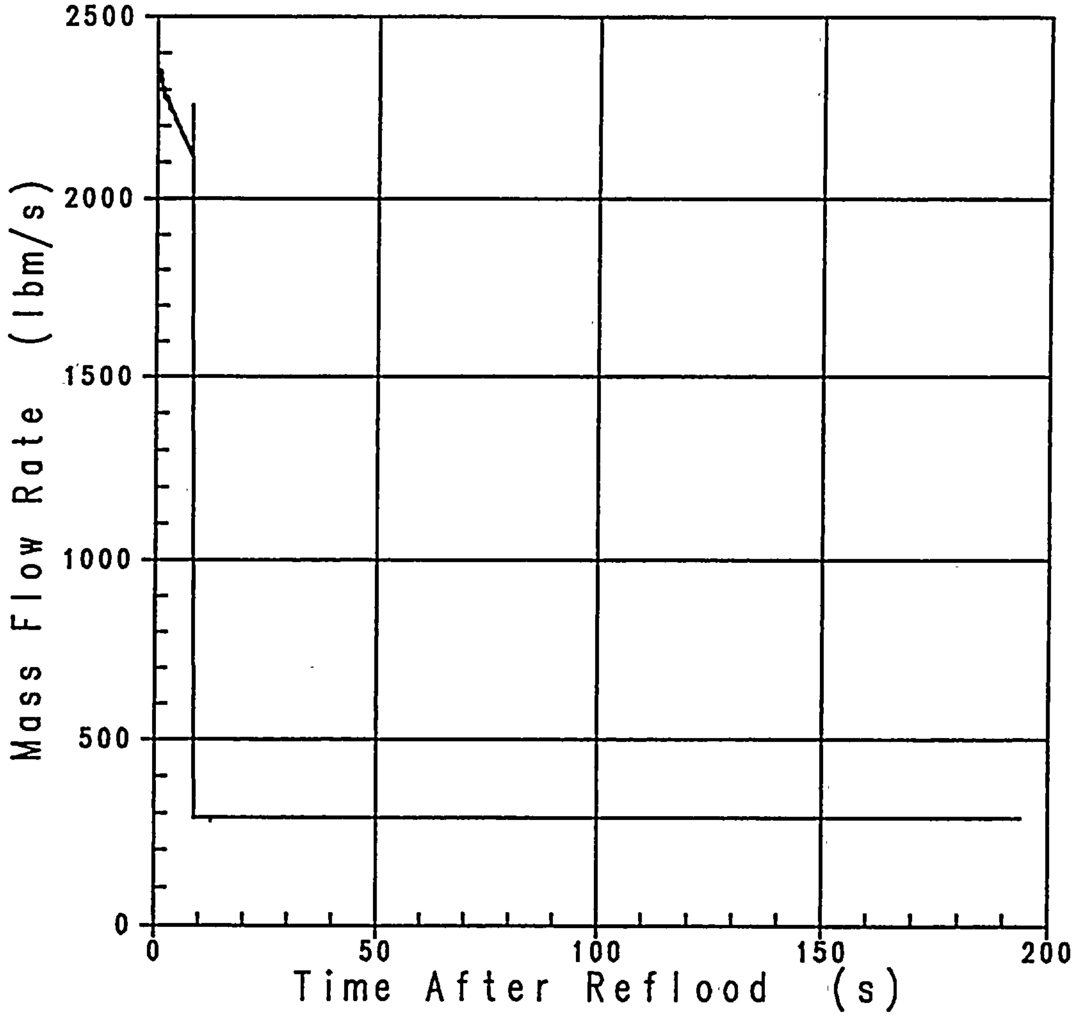

Figure 3.3.1-43: ECCS Flows During Reflood for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

Time	= 0.0 seconds is Bottom Of Core Recovery Time
Time from Initiation of Event	= 53.8 seconds after break for C_D=0.4, Low Tavg, 8.5 ft Skewed Power Shape Case

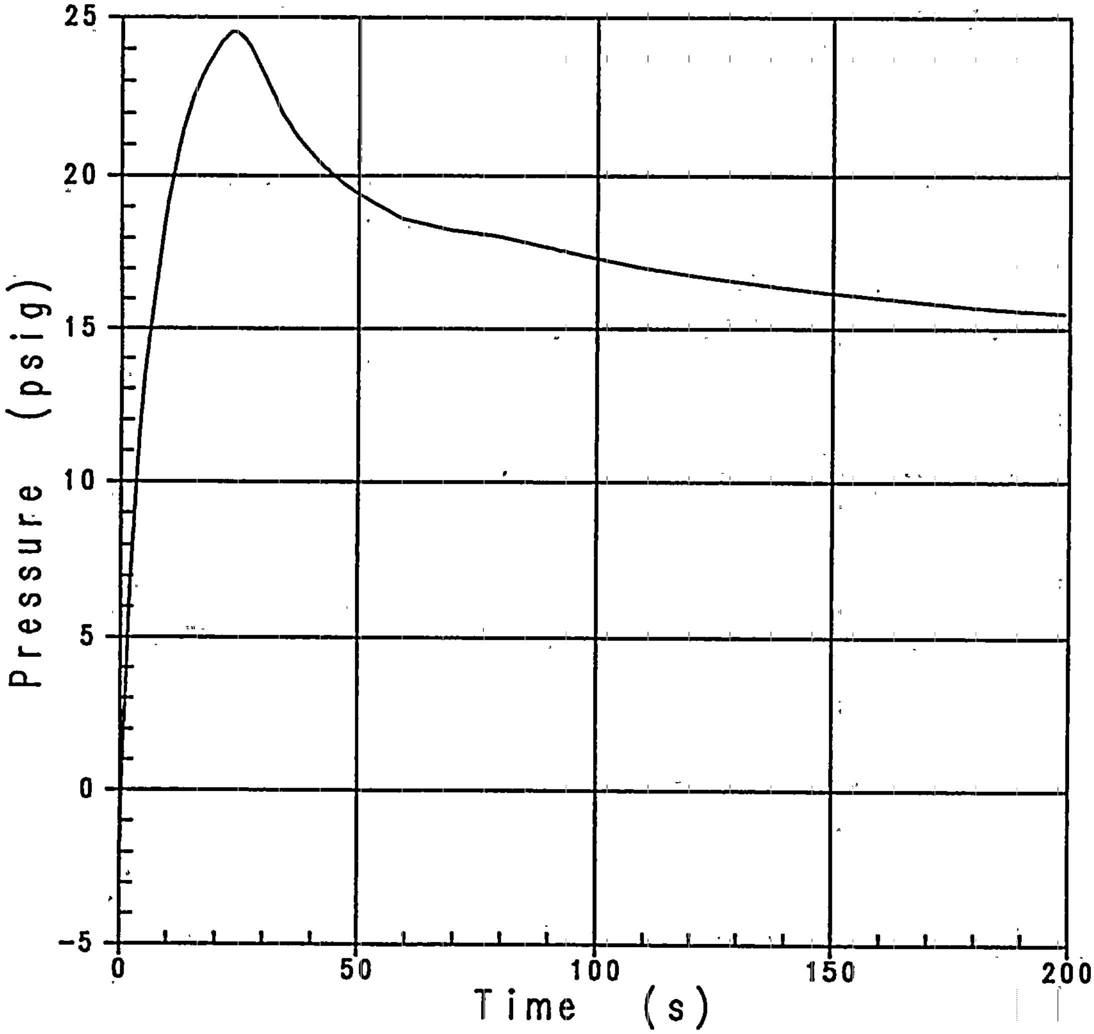

Figure 3.3.1-44: Containment Pressure Transient for C_D=0.4, Low T_{avg}, 8.5 ft Skewed Power Shape

3.3.2 Small Break LOCA

3.3.2.1 Introduction

This section contains information regarding the small break Loss-of-Coolant Accident (LOCA) analysis and evaluations performed in support of the uprating program for Turkey Point Units 3 and 4. The purpose of analyzing the small break LOCA is to demonstrate that conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the uprating. Important input assumptions, as well as analytical models and analysis methodology for the small break LOCA, are contained in subsequent sections. Analysis results are provided in the form of tables and figures, as well as a more detailed description of the limiting transient. It was determined that no design or regulatory limit related to the small break LOCA would be exceeded due to the uprated power and assumed plant parameters. The SBLOCA was previously submitted under FPL letter L-95-193, dated July 26, 1995.

3.3.2.2 Input Parameters and Assumptions

The following important plant conditions and features are listed in Table 3.3.2-1. Several additional considerations that are not identified in Table 3.3.2-1 are discussed below:

Figure 3.3.2-1 depicts the hot rod axial power shape modeled in the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is + 20%). Such a distribution is limiting for small break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The chosen power shape has been conservatively scaled to a flat K(Z) envelope based on the peaking factors given above.

Figure 3.3.2-2 provides the degraded HHSI flow versus pressure curve modeled in the small break LOCA analysis. The flow from one HHSI pump only is assumed in this analysis.

3.3.2.3 Description of Analyses/Evaluations Performed

<u>Analytical Model</u>

For small breaks, the NOTRUMP computer code (References 2 and 3) is employed to calculate the transient depressurization of the reactor coolant system (RCS), as well as to describe the mass and energy release of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these advanced features are: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA

Emergency Core Cooling System (ECCS) Evaluation Model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 4).

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a small break LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with a version of the LOCTA-IV code (Reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions (see Figure 3.3.2-3).

Analysis

A spectrum of 2-inch, 3-inch, and 4-inch equivalent diameter cold leg breaks was performed using the analytical model described above. A sensitivity of the limiting transient to the RCS vessel average temperature was also performed.

The most limiting single active failure assumed for a small break LOCA is that of an emergency power train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. This means that credit may be taken for at most two high head safety injection (SI) pumps and one low head, or residual heat removal (RHR), pump. However, in the analysis of the small break LOCA presented here, only the minimum delivered ECCS flow from a single high head SI pump with degraded flow was assumed.

The small break LOCA analysis performed for the Turkey Point Units 3 and 4 uprating program utilizes the NRC-approved NOTRUMP Evaluation Model (References 2 and 3), with appropriate modifications to model pumped SI and accumulator injection in the broken loop as well as an improved condensation model (COSI) for the pumped SI into the broken and intact loops (References 6 and 7).

The small break LOCA analysis performed for the Turkey Point uprating program assumes SI is delivered to both the intact and broken loops at the RCS backpressure.

Prior to break initiation, the plant is assumed to be in a full power (102%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Table 3.3.2-1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves, which were modeled with 3 percent accumulation and 3 percent tolerance.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1805 psia, is reached. LOOP is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1615 psia, is reached. Safety injection is delayed 35 seconds after the occurrence of the low pressure condition. This delay accounts for signal initiation, diesel generator start up and emergency power bus loading consistent with the assumed loss of offsite power coincident with reactor trip, as well as the time involved in aligning the valves and bringing the HHSI pump up to full speed. These countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. (However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical - refer to Section 3.3.5). In addition, credit is taken in the small break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position. A rod drop time of 3 seconds was assumed while also considering an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 5 seconds from the time of reactor trip signal to full rod insertion was used in the small break LOCA analysis.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the small break transient (prior to the assumed loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient enough to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a partial period of core uncovery occurs. Ultimately, the small break transient analysis is terminated when the ECCS flow provided to the RCS exceeds the break flow rate.

The core heat removal mechanisms associated with the small break transient include not only the break itself and the injected ECCS water, but also that heat transferred from the RCS to the steam generator secondary side. Main Feedwater (MFW) is assumed to be isolated coincident with the safety injection signal, and the MFW pumps coast down to 0% flow in 10 seconds. A continuous supply of makeup water is also provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal occurs coincident with the safety injection signal, resulting in the assumed delivery of full AFW system flow 120 seconds following the signal. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 600 psig, as in the case of the limiting 3-inch break and the 4-inch break, the cold leg accumulators begin to inject borated water into the reactor coolant loops. In the case of the 2-inch break however, the vessel mixture level is recovered without the aid of accumulator injection.

Evaluations

Upon completion of the small break LOCA analysis, an evaluation was performed for automatic containment spray actuation during small break LOCA. This evaluation accounts for the fact that Turkey Point Units 3 and 4 may be subject to SI interruption for up to 2 minutes while switching over to cold leg recirculation. The results of this evaluation are discussed in Section 3.3.2.5.

3.3.2.4 Acceptance Criteria for Analyses / Evaluations

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F,

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation,

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling,

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Criteria 1 through 3 are explicitly covered by the small break LOCA analysis at uprated conditions.

For criterion 4, the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the PCT criterion of 10 CFR 50.46 and consequently, demonstrate the core remains amenable to cooling.

For criterion 5, Long-Term Core Cooling (LTCC) considerations are not directly applicable to the small break LOCA transient, but are assessed in Section 3.3.5 as part of the evaluation of ECCS performance.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA.

3.3.2.5 Results

In order to determine the conditions that produced the most limiting small break LOCA case (as determined by the highest calculated peak cladding temperature), a total of four cases were examined. These cases included the investigation of variables including break size and RCS temperature to ensure that the most severe postulated small break LOCA event was analyzed. The following discussions provide insight into the analyzed conditions.

First, a break spectrum based on high RCS T_{avg} was performed, as this was expected to yield more limiting PCT results than low RCS T_{avg}. The limiting break for the Turkey Point Units was found to be a 3-inch diameter cold leg break. The results of Reference 8 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations. The PCT attained during the transient was 1688°F (refer to Table 3.3.2-2). Inherent in the limiting small break analysis are several input assumptions (see Section 3.3.2.2 and Table 3.3.2-1), while Table 3.3.2-3 provides the key transient event times.

A summary of the transient response for the limiting high T_{avg} 3-inch break case is shown in Figures 3.3.2-4 through 3.3.2-12. These figures present the response of the following parameters:

- RCS Pressure Transient,
- Core Mixture Level,
- Peak Cladding Temperature,
- Top Core Node Vapor Temperature,
- Safety Injection Mass Flow Rate for the Intact and Broken Loops,
- Cold Leg Break Mass Flow Rate,
- Hot Spot Rod Surface Heat Transfer Coefficient, and
- Hot Spot Fluid Temperature.

Upon initiation of the limiting 3-inch break, there is a slow depressurization of the RCS (see Figure 3.3.2-4). During the initial period of the small break transient, the effect of the break flow rate is not sufficient to overcome the flow rate maintained by the reactor coolant pumps as they coast down. As such, normal upward flow is maintained through the core and core heat is adequately removed. Following reactor trip, the removal of the heat generated as a result of fission products decay is accomplished via a two-phase mixture level covering the core. From the core mixture level and cladding temperature transient plots for the 3-inch break calculations given in Figures 3.3.2-5 and 3.3.2-6, respectively, it is seen that the peak cladding temperature occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 3.3.2-7).

A comparison of the flow provided by the safety injection system to the intact and broken loops to the total cold leg break mass flow rate at the end of the transient (as given in Figures 3.3.2-8, 3.3.2-9 and 3.3.2-10, respectively), shows that at the time the transient was terminated, the total safety injection flow rate that was delivered to the intact and broken loops exceeds the mass flow rate out the break.

In addition, the inner vessel core mixture level has recovered the top of the core (Figure 3.3.2-5). Figures 3.3.2-11 and 3.3.2-12 provide additional information on the hot rod surface heat transfer coefficient at the hot spot and fluid temperature at the hot spot, respectively.

There is no longer a concern of exceeding the 10 CFR 50.46 criteria as described in Section 3.3.2.4 since:

1. The RCS pressure is gradually decaying, and
2. The net mass inventory is increasing.

As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel cladding temperatures will continue to decline. The 3-inch high T_{avg} small break LOCA transient is terminated.

Additional Break Cases

Studies documented in Reference 8 have determined that the limiting small-break transient occurs for breaks of less than 10 inches in diameter. To ensure that the 3-inch diameter break was the most limiting, calculations were also performed with break equivalent diameters of 2 inches and 4 inches. The results of each of these cases are given in Tables 3.3.2-2 and 3.3.2-3. Plots of the following parameters are given in Figures 3.3.2-13 through 3.3.2-15 for the 2-inch break case and Figures 3.3.2-16 through 3.3.2-18 for the 4-inch break.

1. RCS Pressure Transient,
2. Core Mixture Level, and
3. Peak Cladding Temperature.

The PCTs for the 2-inch and 4-inch breaks were 1656°F and 1583°F, respectively (see Table 3.3.2-2). The PCTs for each of these cases was calculated to be less than that for the 3-inch break case based on high T_{avg} conditions.

Limiting Temperature Conditions

Reduced operating temperature typically results in a PCT benefit for the small break LOCA. However, due to competing effects and the complex nature of small break LOCA transients, there have been some instances where more limiting results have been observed for the reduced operating temperature case. For this reason, a small break LOCA transient based on a lower bound RCS vessel average temperature was performed.

The temperature window analyzed was based on a nominal vessel average temperature of 574.2°F, with ± 3°F for an operating window and ± 8.5°F to bound uncertainties. The break spectrum was performed at the high vessel average temperature, as this case was expected to yield limiting results. Then, a sensitivity analysis for the low vessel average temperature was performed based on the limiting 3-inch break case from the break spectrum analyses previously described.

Plots of the following parameters are given in Figures 3.3.2-19 through 3.3.2-21 for the 3-inch break case at low T_{avg} conditions:

1. RCS Pressure Transient,
2. Core Mixture Level, and
3. Peak Cladding Temperature.

The PCT for the 3-inch break case based on low vessel average temperature was 1619°F (see Table 3.2-2). Therefore, the PCT for this case was calculated to be less than that for the 3-inch break case with high vessel average temperature conditions.

Evaluations

The evaluation for containment spray actuation in small break LOCA resulted in no change to the predicted small break LOCA PCT for the various cases analyzed.

3.3.2.6 Conclusions

A break spectrum supporting the high nominal vessel average temperature was performed. Peak cladding temperatures of 1656°F, 1688°F, and 1583°F were calculated for the 2-inch, 3-inch, and 4-inch cold leg breaks, respectively, thus identifying the 3-inch equivalent diameter break as limiting. A sensitivity to low nominal vessel average temperature was performed. The calculated peak cladding temperature was 1619°F for the Low Tavg case. Therefore, the 3-inch equivalent diameter cold leg break, high nominal vessel average temperature, is the limiting case.

The analyses presented in this section show that the high head safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperatures below the required limit of 10 CFR 50.46 which is defined in Section 3.3.2.4.

Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break Loss-of-Coolant Accident.

3.3.2.7 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P.E., "NOTRUMP - A Nodal Transient Small Break and General Network Code,"WCAP-10079-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (non-proprietary), August 1985.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

5. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (proprietary) and WCAP-8305 (non-proprietary), June 1974.

6. Thompson, C. M, et al., "Addendum to the Westinghouse Small Break LOCA Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and the COSI Condensation Model", WCAP-10054-P, Addendum 2 (proprietary) and WCAP-10081-NP (non-proprietary), August 1994.

7. Shimeck, D. J., "COSI SI/Steam Condensation Experiment Analysis", WCAP-11767-P-A (proprietary), and WCAP-11372-NP-A (non-proprietary), March 1988.

8. Rupprecht, S. D. et al, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", WCAP-11145-P-A (proprietary), October 1986.

Table 3.3.2-1

Input Parameters Used in the Small Break LOCA Analysis

Parameter	High Tavg	(Low Tavg)
Reactor core rated thermal power[1], (MWt)	2300	
Peak linear power[1,2], (kw/ft)	14.9	
Total peaking factor (F_Q^T) at peak[2]	2.50	
Power shape[2]	See Figure 3.3.2-1	
$F_{\Delta H}$	1.70	
Fuel[3]	15x15 DRFA	
Accumulator water volume, nominal (ft^3/acc.)	892	
Accumulator tank volume, nominal (ft^3/acc.)	1200	
Accumulator gas pressure, minimum (psig)	600	
Pumped safety injection flow	See Figure 3.3.2-2	
Steam generator tube plugging level (%)[4]	20	
Thermal Design Flow/loop, (gpm)	85,000	
Vessel average temperature w/ uncertainties, (°F)	585.7	(562.7)
Reactor coolant pressure w/ uncertainties, (psia)	2320	
Min. aux. feedwater flowrate/loop, (lb/sec)[5]	9.26	

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1 Two percent is added to this power to account for calorimetric error. Reactor coolant pump heat is not modeled in the SBLOCA analyses.

2 This represents a power shape corresponding to a one-line segment peaking factor envelope, K(z), based on $F_Q^T = 2.50$.

3 DRFA fuel type modeled in the small break LOCA analysis.

4 Maximum plugging level in any one or all steam generators.

5 Flowrates per steam generator.

Table 3.3.2-2

Small Break LOCA Analysis Fuel Cladding Results

Break Spectrum, (High T_{avg})

	2-inch	3-inch	4-inch
Peak Cladding Temperature (°F)	1656	1688	1583
Peak Cladding Temperature Location (ft)*	11.75	11.75	11.50
Peak Cladding Temperature Time (sec)	2627	1188	668
Local Zr/H_2O Reaction, Max (%)	2.0188	1.5535	0.6679
Local Zr/H_2O Reaction Location (ft)*	11.75	11.50	11.25
Total Zr/H_2O Reaction (%)	< 1.0	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A

Results for the limiting 3-inch break size

	High Tavg	Low Tavg
Peak Cladding Temperature (°F)	1688	1619
Peak Cladding Temperature Location (ft)*	11.75	11.50
Peak Cladding Temperature Time (sec)	1188	1229
Local Zr/H_2O Reaction, Max (%)	1.5535	1.1034
Local Zr/H_2O Reaction Location (ft)*	11.50	11.50
Total Zr/H_2O Reaction (%)	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst
Hot Rod Burst Location (ft)*	N/A	N/A

* From bottom of active fuel

Table 3.3.2-3

Small Break LOCA Analysis Time Sequence of Events

Break Spectrum, (High T_{avg})

	2-inch	3-inch	4-inch
Break Occurs (sec)	0.0	0.0	0.0
Reactor Trip Signal (sec)	40.6	17.0	10.4
Safety Injection Signal (sec)	58.9	30.4	21.4
Top Of Core Uncovered (sec)	1402	482	278[1]
Accumulator Injection Begins (sec)	N/A	1040	525
Peak Clad Temperature Occurs (sec)	2627	1188	668
Top Of Core Covered (sec)	4554	2363	965

Results for the limiting 3-inch break size

	High Tavg	Low Tavg
Break Occurs (sec)	0.0	0.0
Reactor Trip Signal (sec)	17.0	14.4
Safety Injection Signal (sec)	30.4	21.8
Top Of Core Uncovered (sec)	482	526
Accumulator Injection Begins (sec)	1040	1086
Peak Clad Temperature Occurs (sec)	1188	1229
Top Of Core Covered (sec)	2363	2343

[1]Momentary core uncovery occurred at 213 seconds during prelude to loop seal clearing. The beginning of the subsequent extended core uncovery at 278 seconds is the time listed.

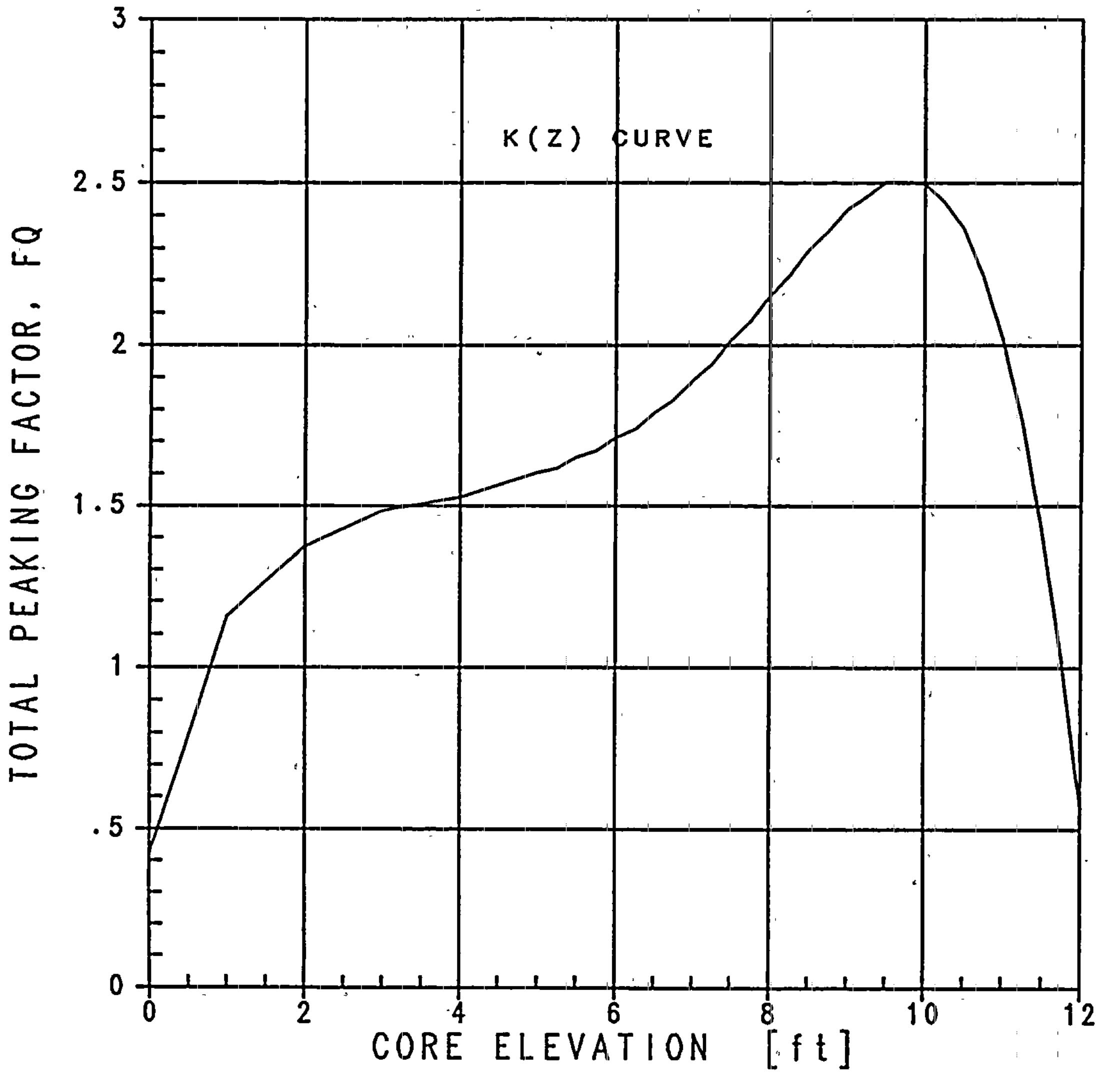

Figure 3.3.2-1: Small Break Hot Rod Power Shape

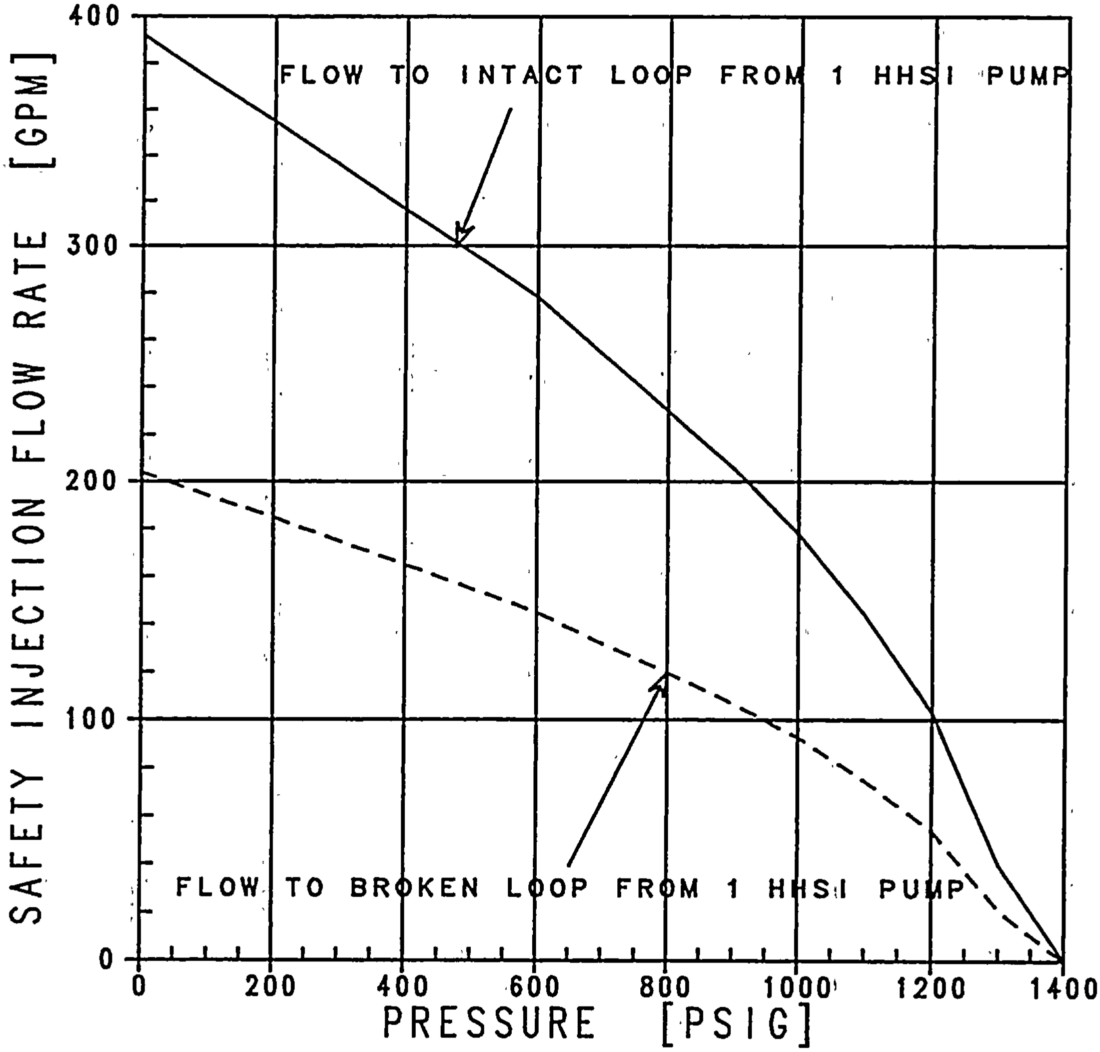

Figure 3.3.2-2: Small Break Pumped Safety Injection Flow Rate - 1 HHSI Pump

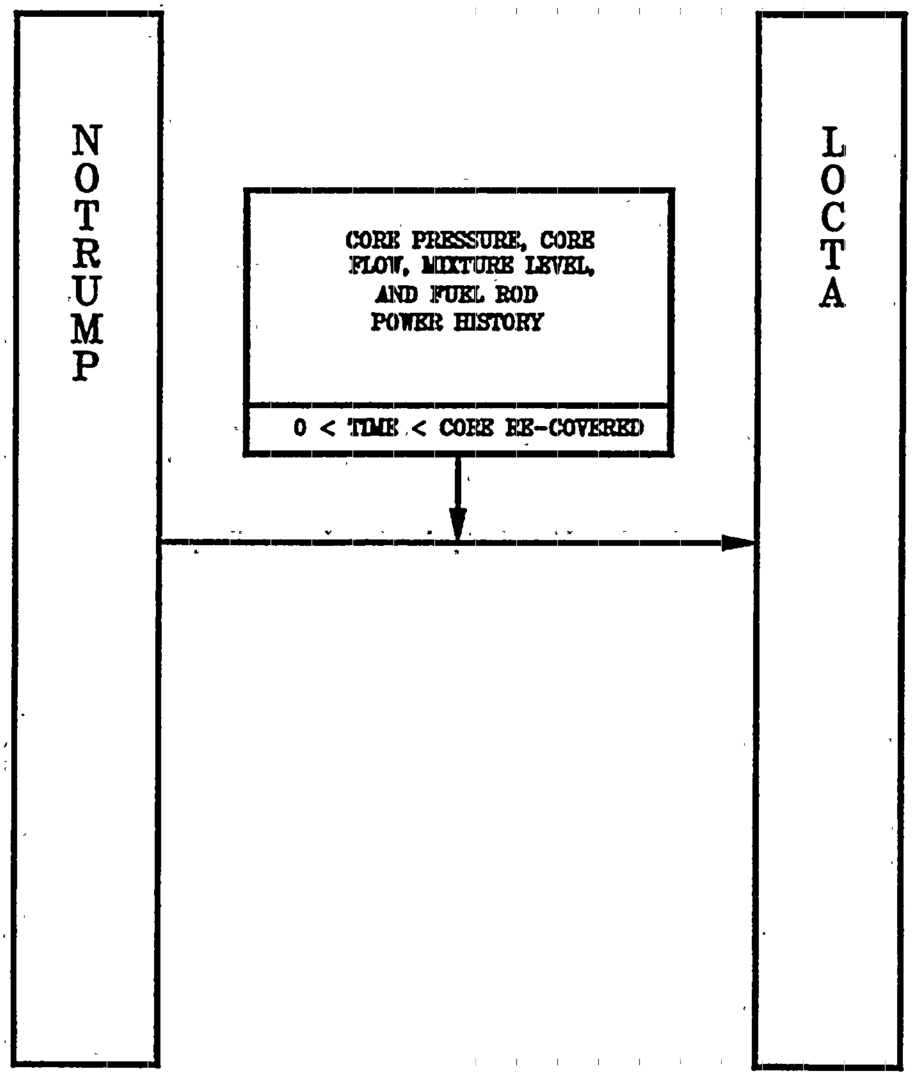

Figure 3.3.2-3: Code Interface Description for the Small Break LOCA Model

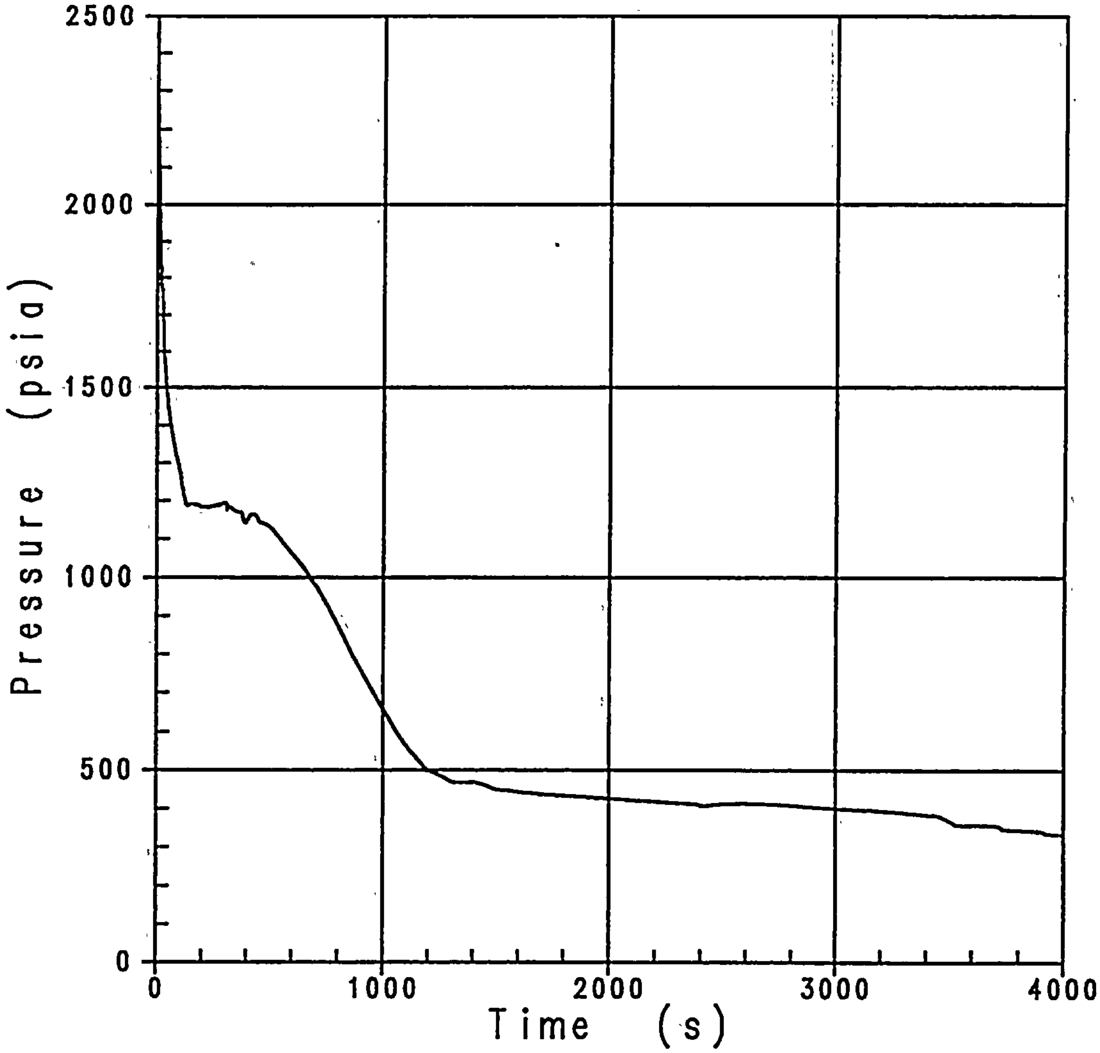

Figure 3.3.2-4: RCS Depressurization Transient, Limiting 3-Inch Break, High T_{avg}

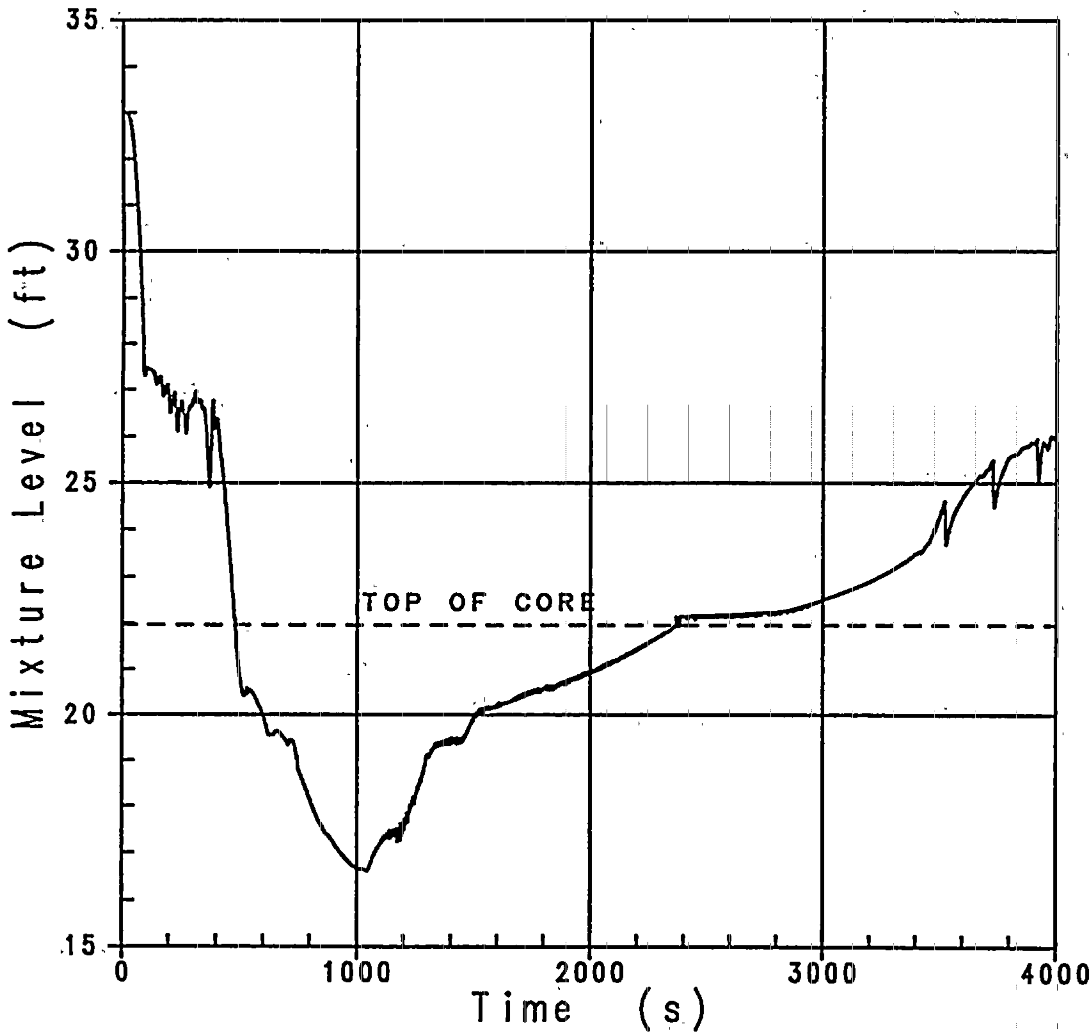

Figure 3.3.2-5: Core Mixture Level, 3-Inch Break, High T_{avg}

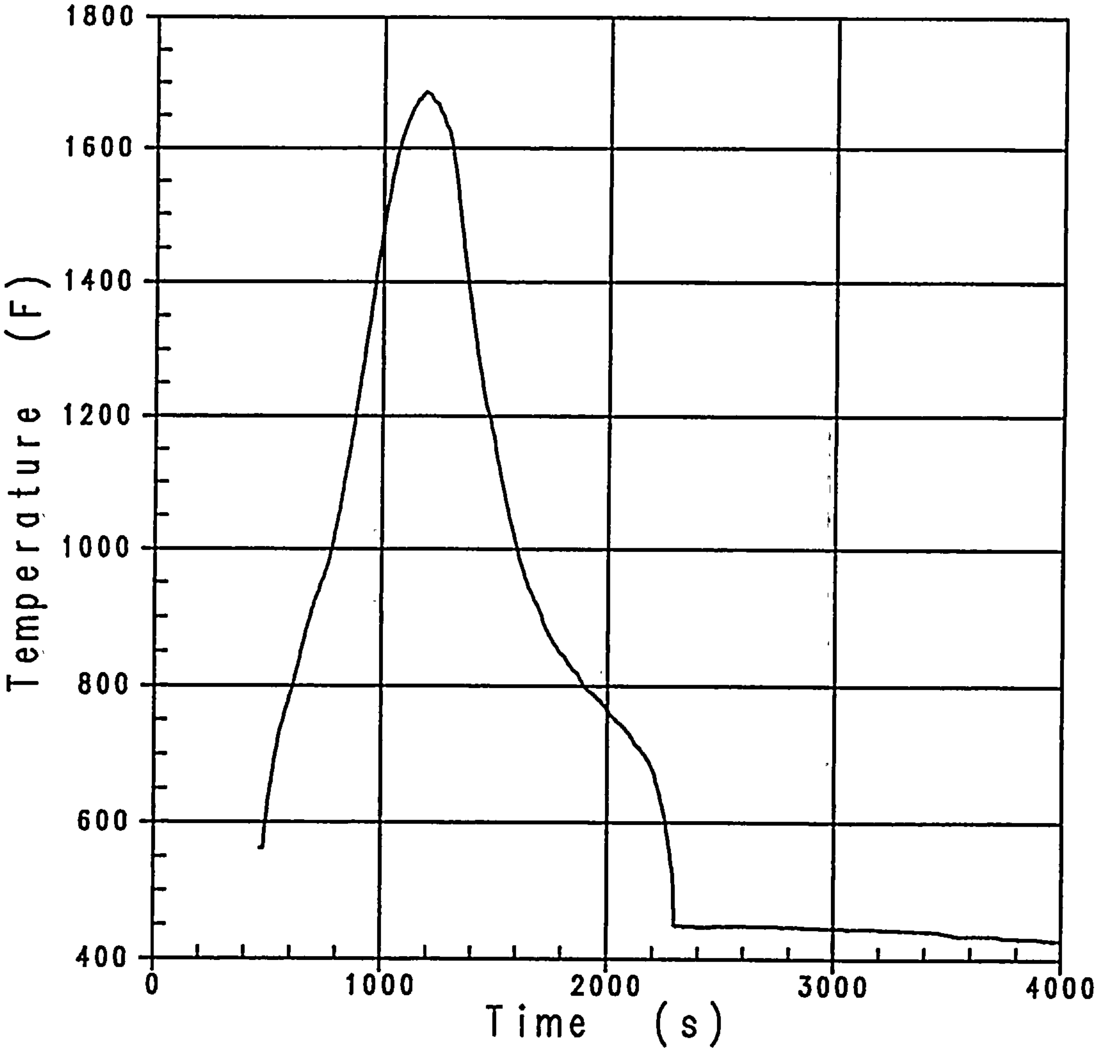

Figure 3.3.2-6: Peak Cladding Temperature - Hot Rod, 3-Inch Break, High T_{avg}

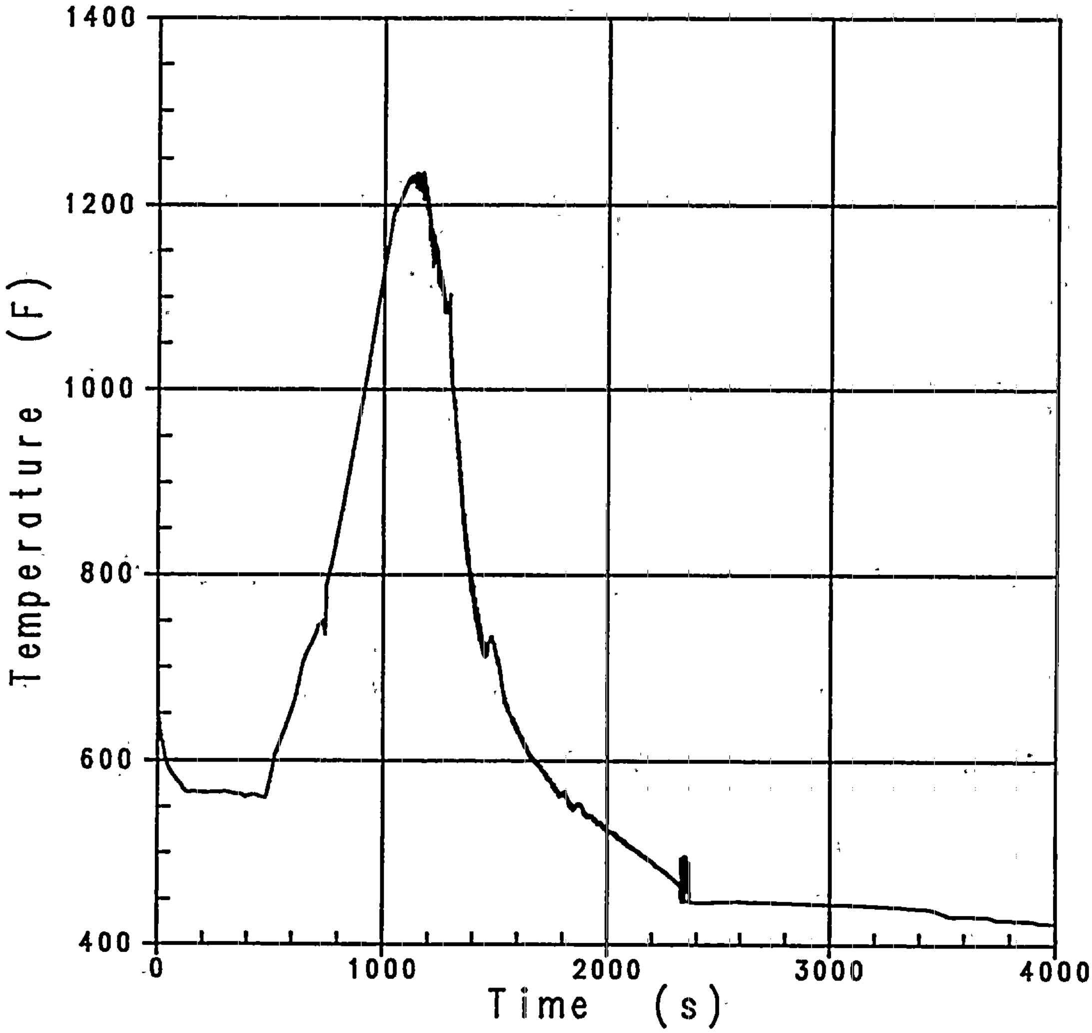

Figure 3.3.2-7: Top Core Node Vapor Temperature, 3-Inch Break, High T_{avg}

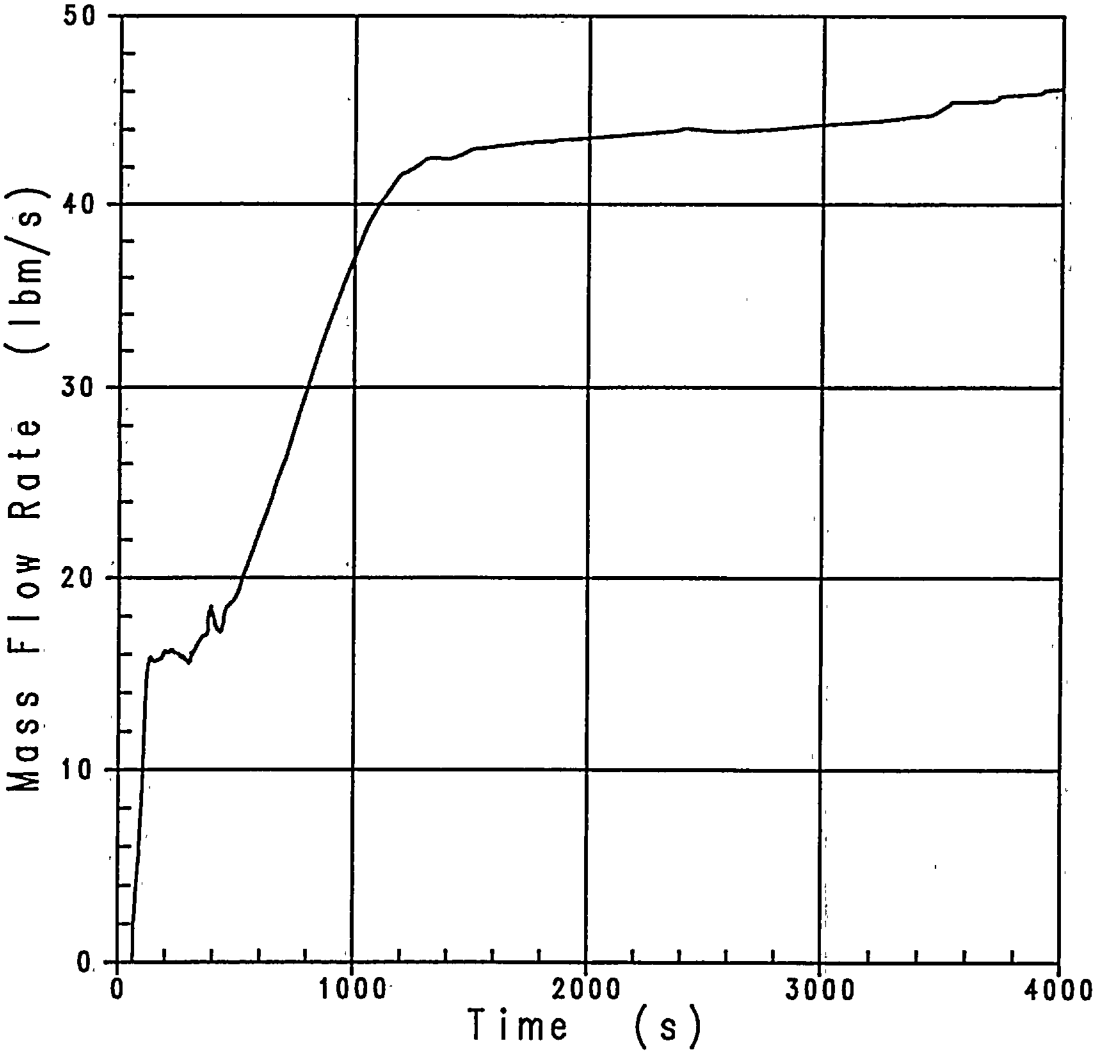

Figure 3.3.2-8: ECCS Pumped Safety Injection - Intact Loop, 3-Inch Break, High T_{avg}

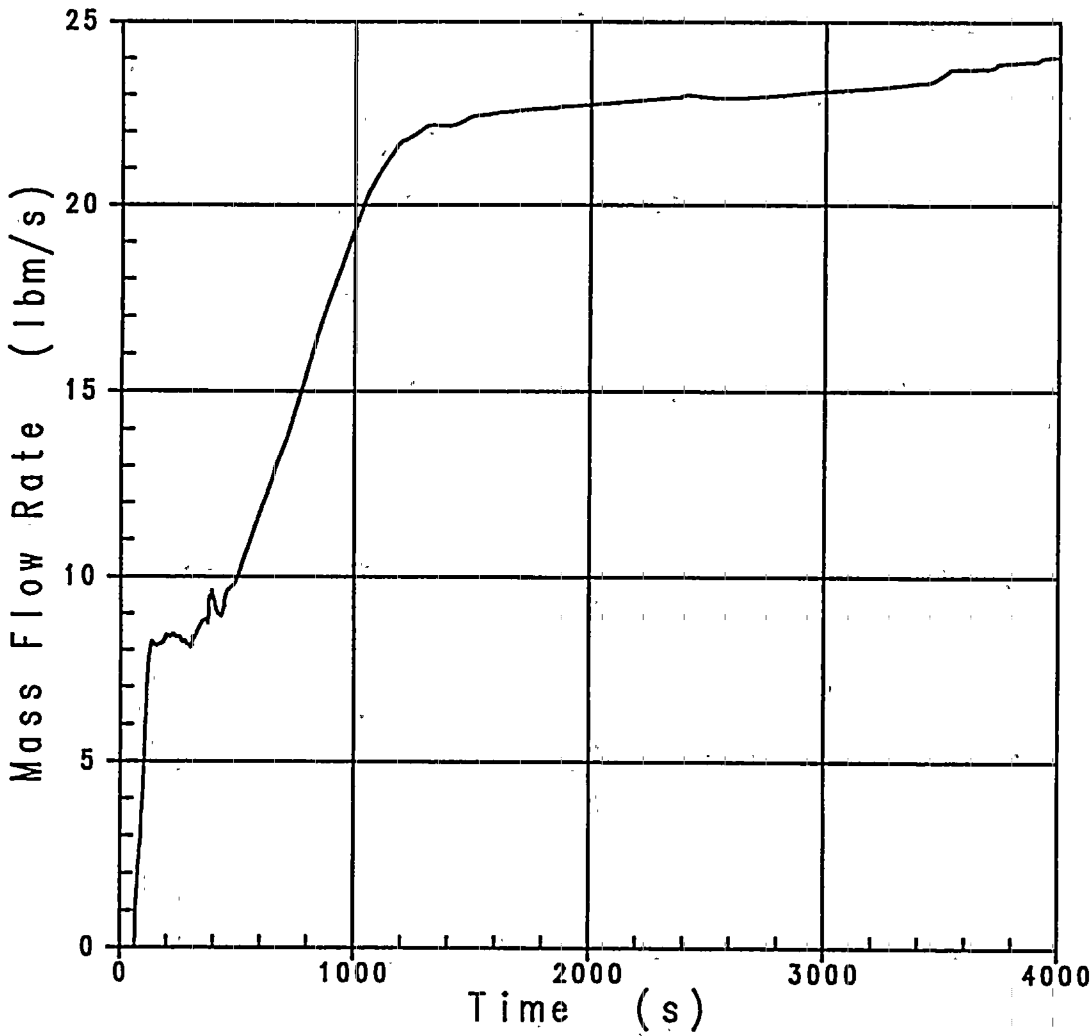

Figure 3.3.2-9: ECCS Pumped Safety Injection - Broken Loop, 3-Inch Break, High T_{avg}

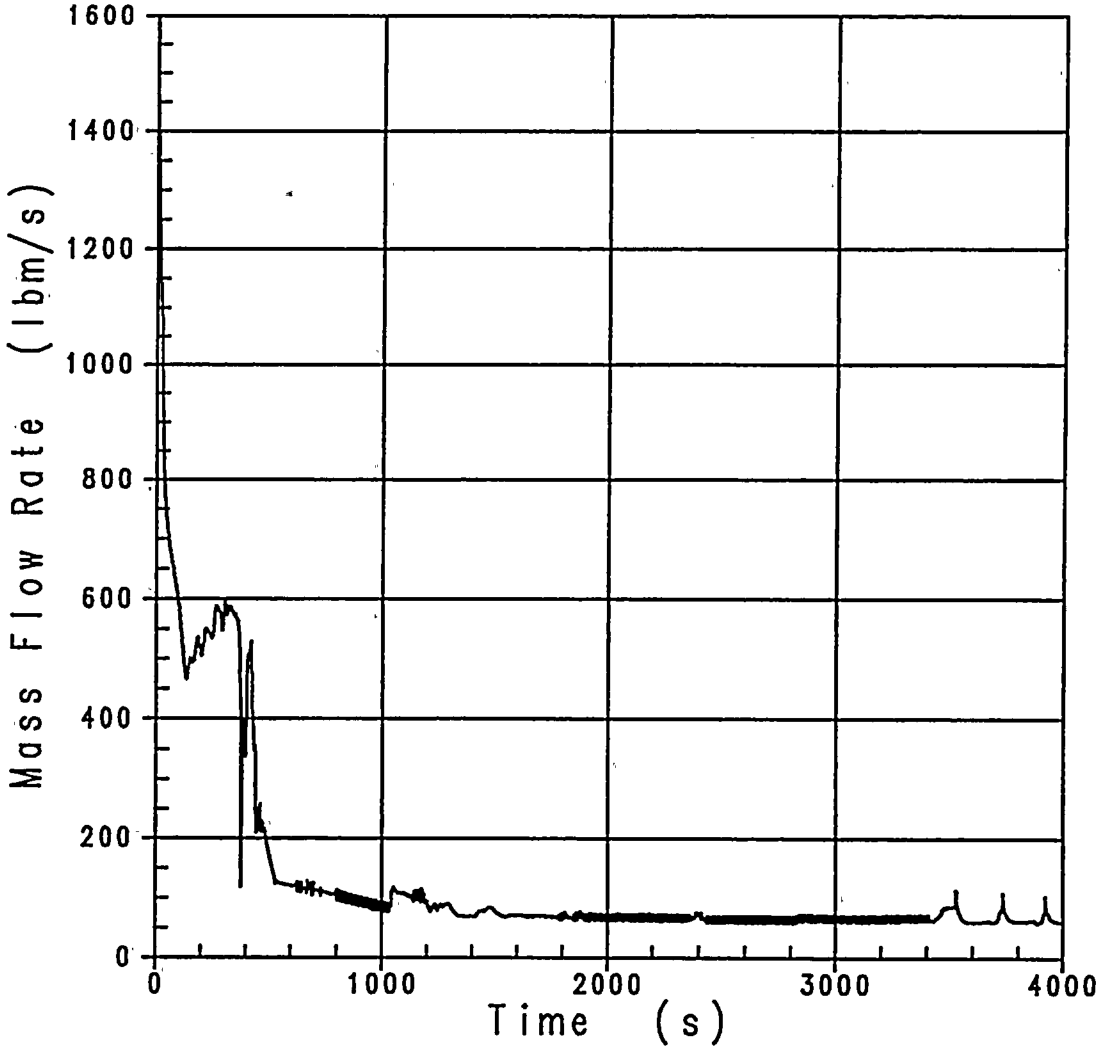

Figure 3.3.2-10: Cold Leg Break Mass Flow, 3-Inch Break, High T_{avg}

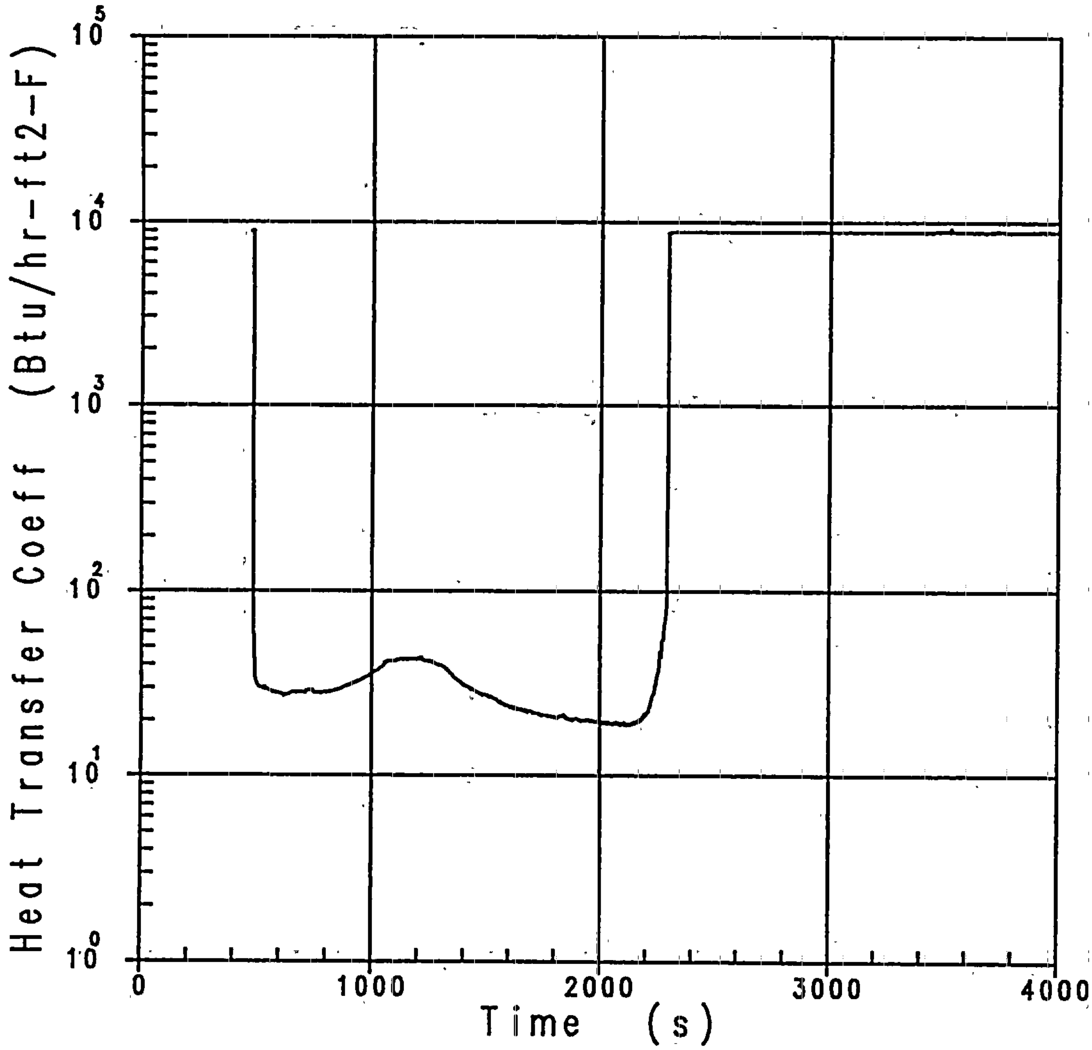

Figure 3.3.2-11: **Hot Rod Surface Heat Transfer Coefficient - Hot Spot, 3-Inch Break, High T_{avg}**

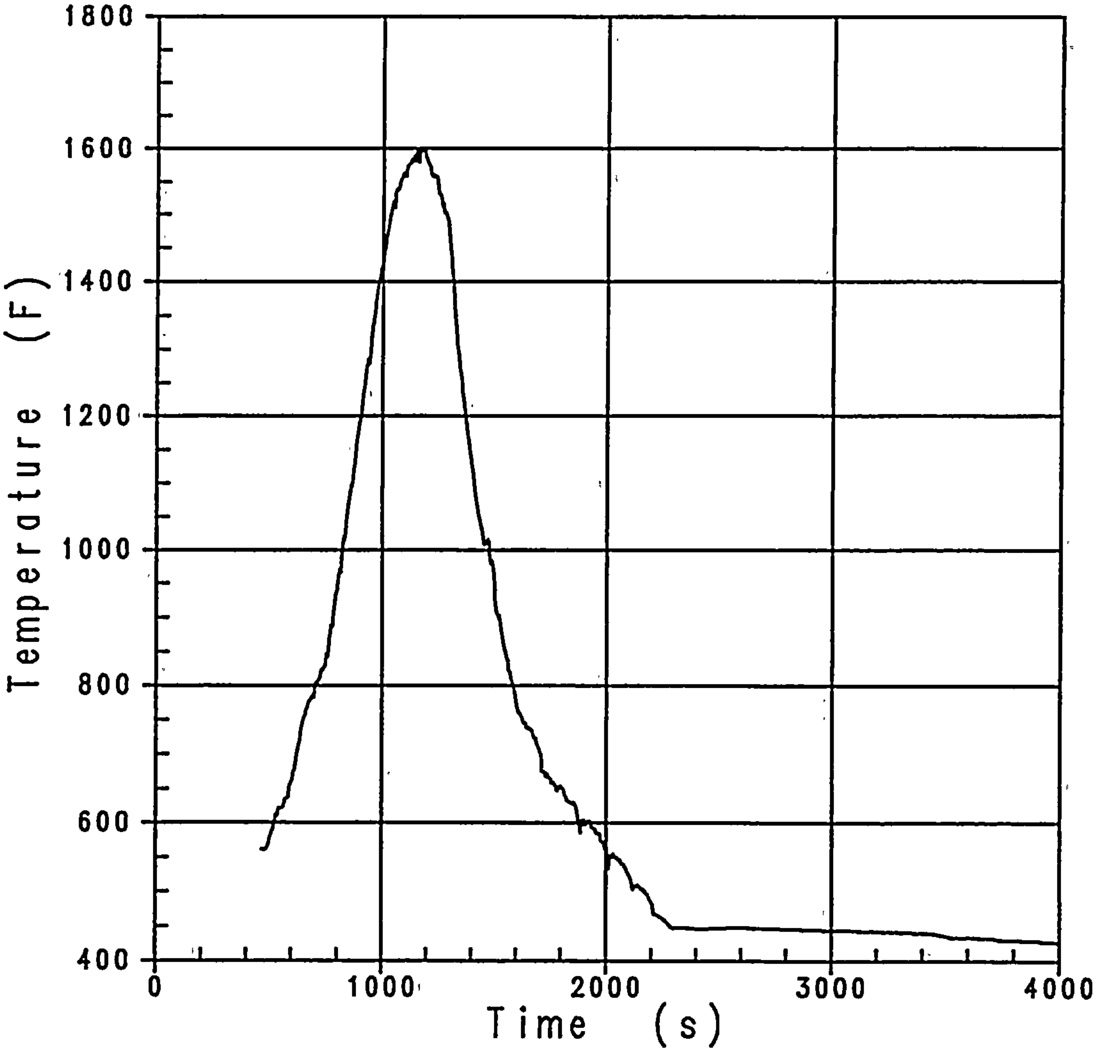

Figure 3.3.2-12: Fluid Temperature - Hot Spot, 3-Inch Break, High T_{avg}

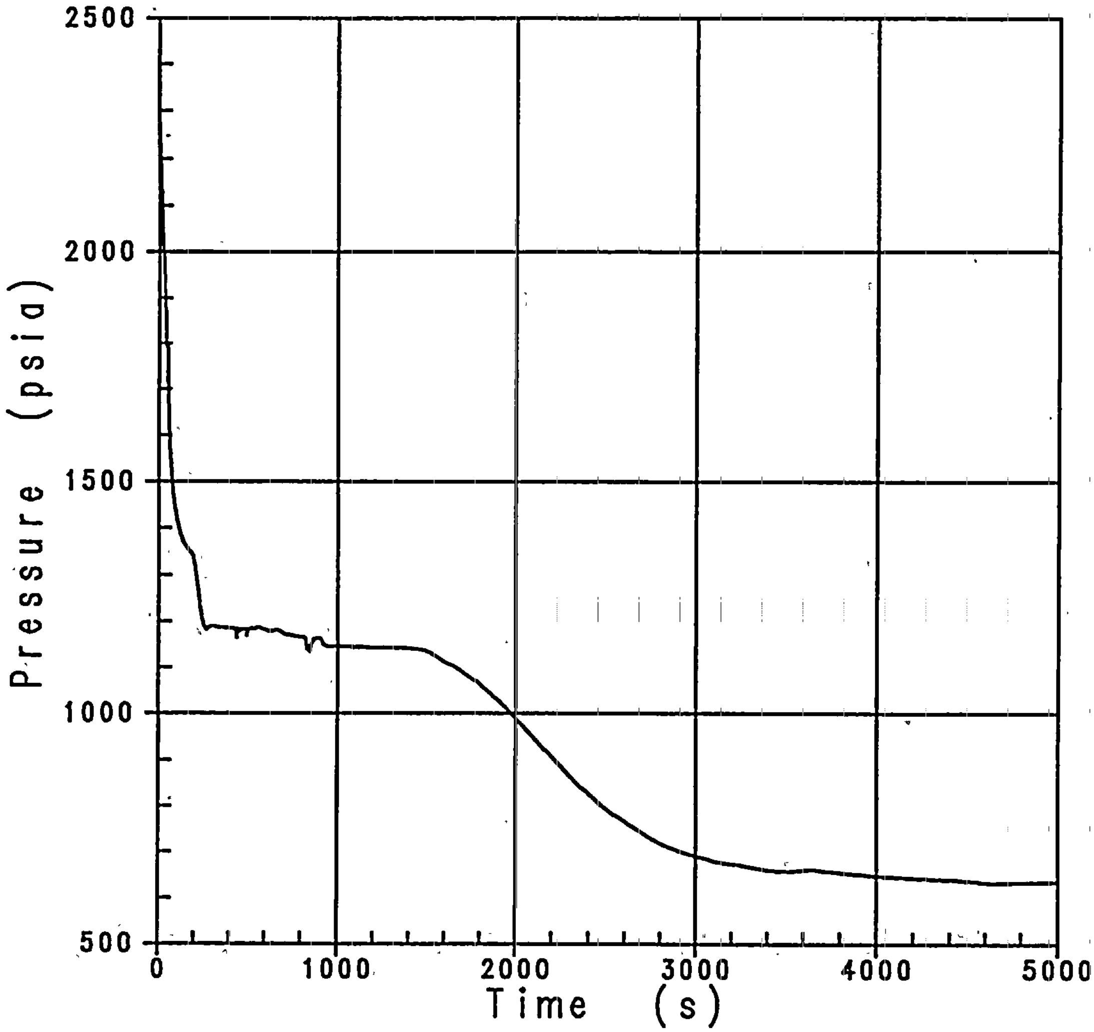

Figure 3.3.2-13: RCS Depressurization Transient, 2-Inch Break, High T_{avg}

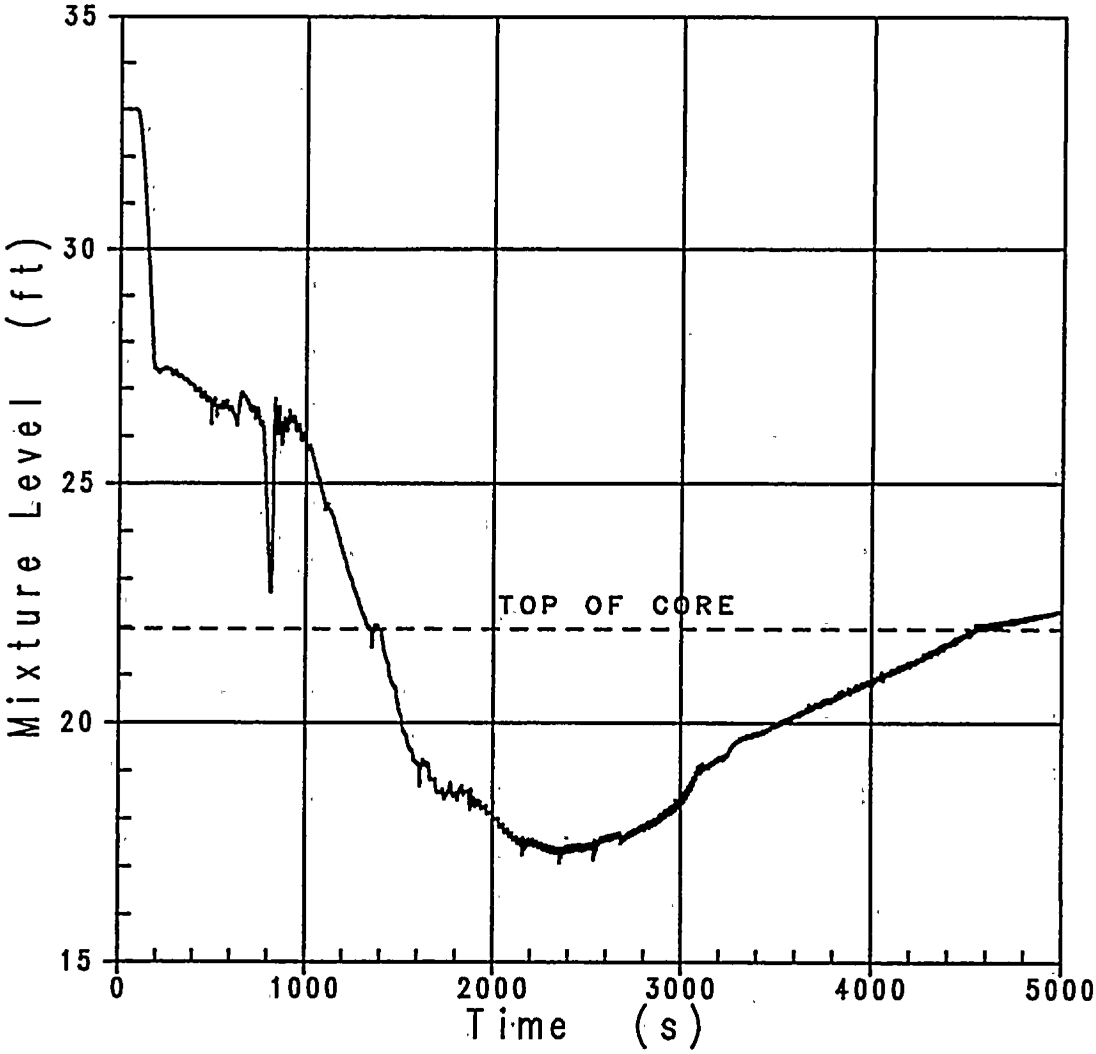

Figure 3.3.2-14: Core Mixture Level, 2-Inch Break, High T_{avg}

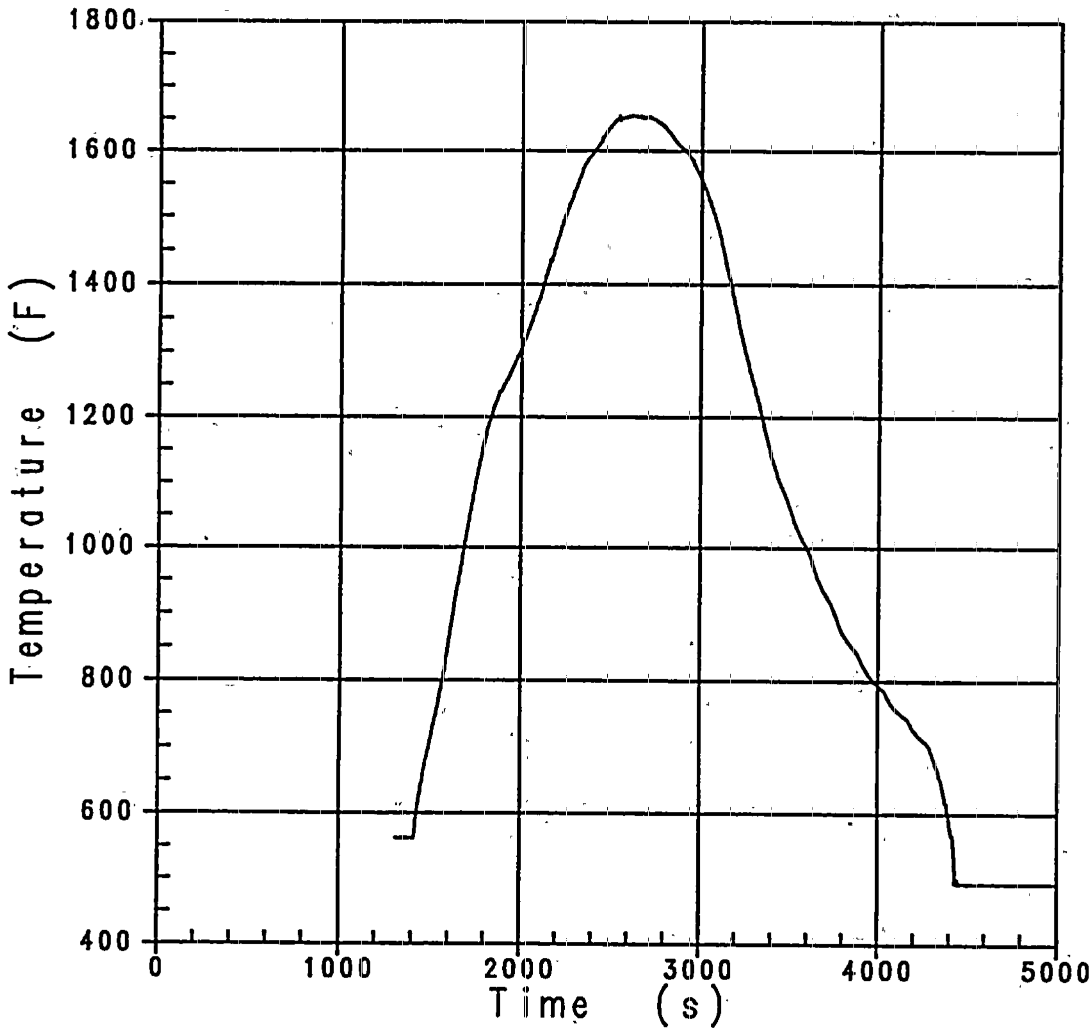

Figure 3.3.2-15: Peak Cladding Temperature - Hot Rod, 2-Inch Break, High T_{avg}

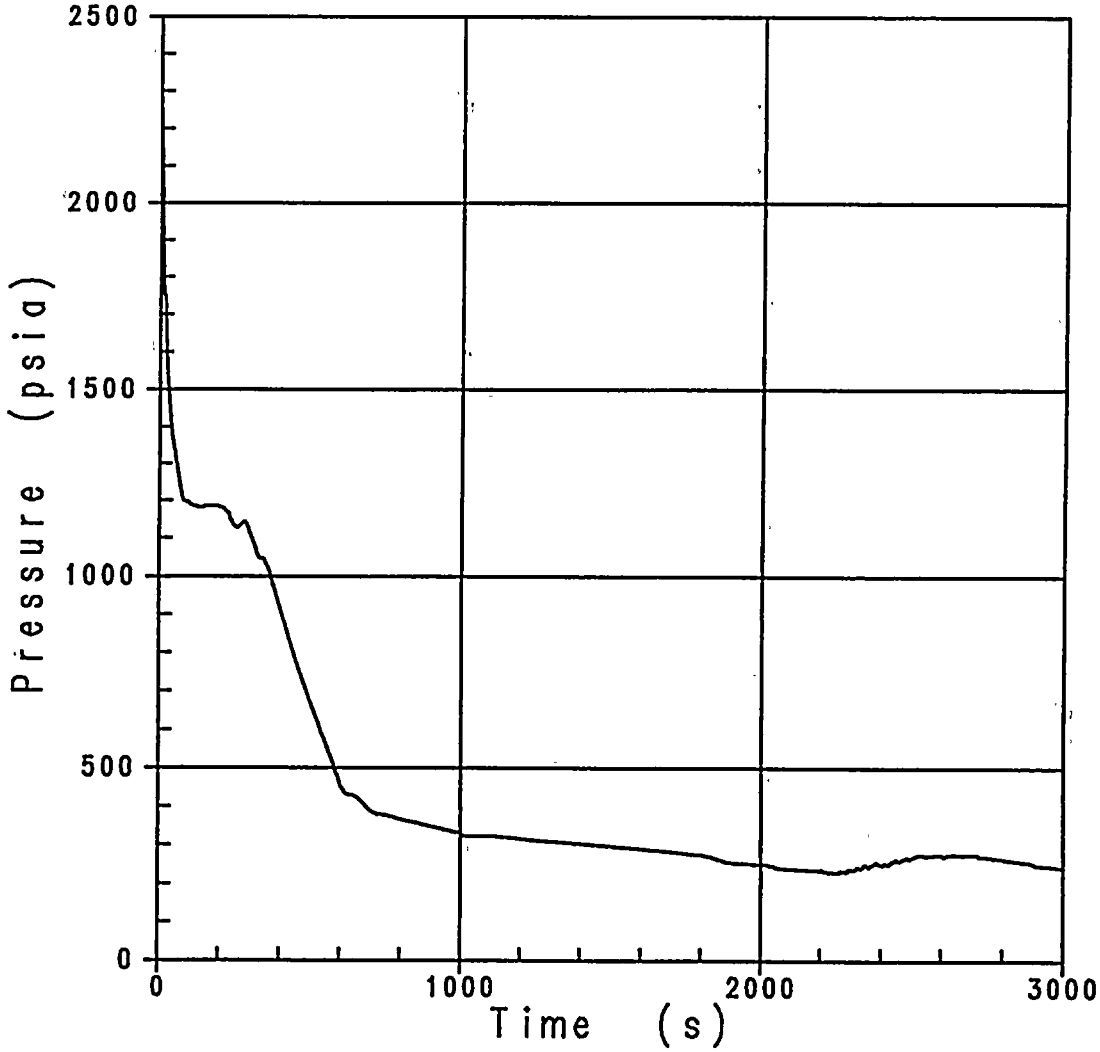

Figure 3.3.2-16: RCS Depressurization Transient, 4-Inch Break, High T_{avg}

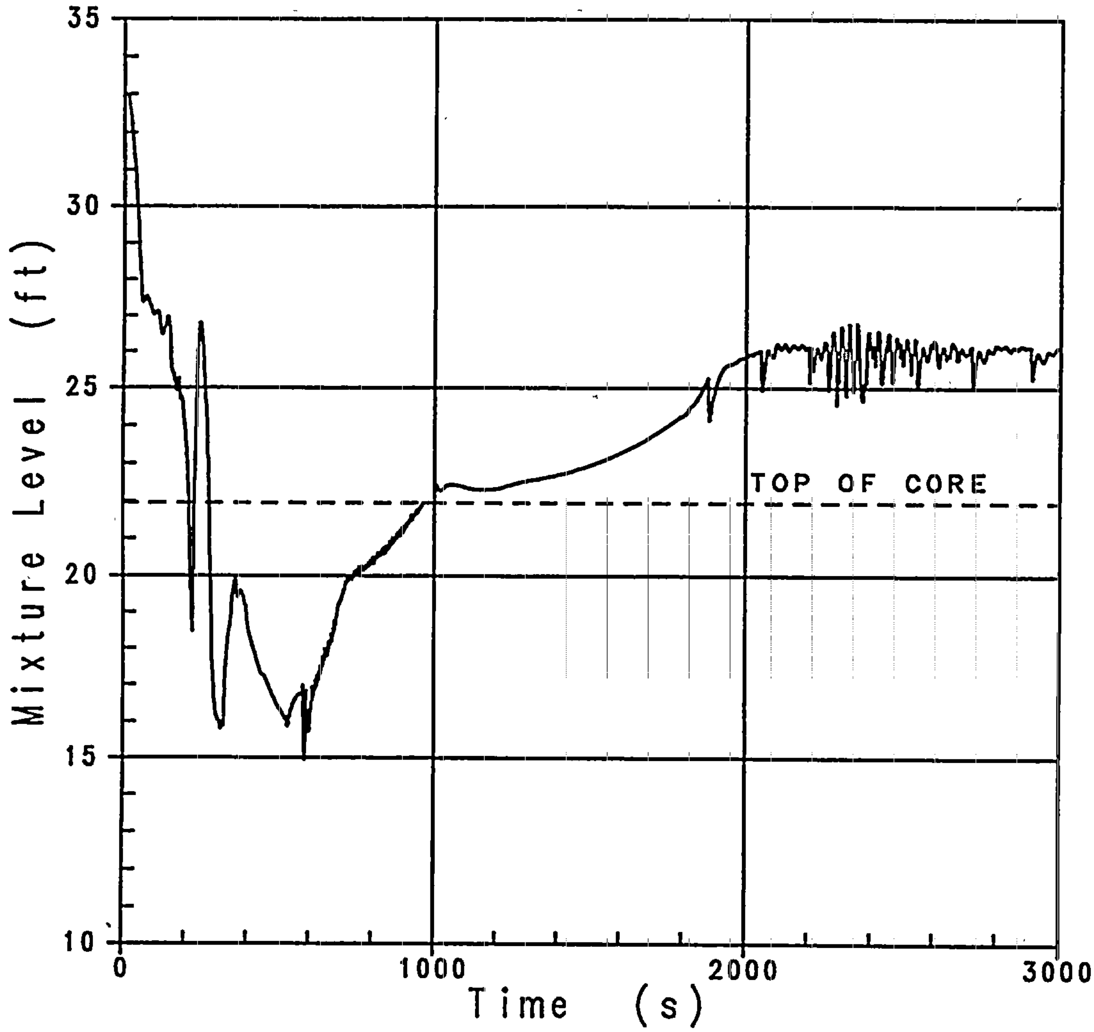

Figure 3.3.2-17: Core Mixture Level, 4-Inch Break, High T_{avg}

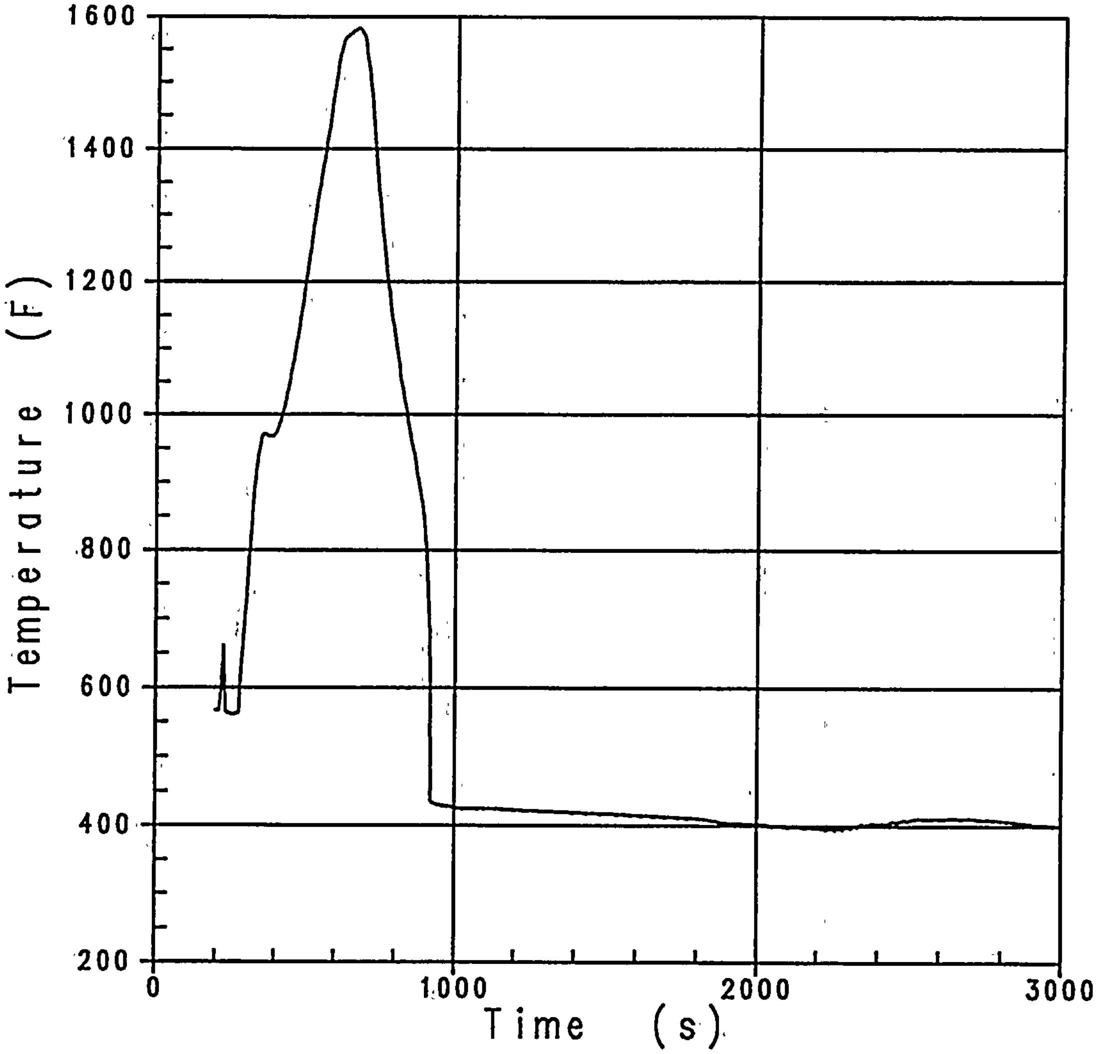

Figure 3.3.2-18: Peak Cladding Temperature - Hot Rod, 4-Inch Break, High T_{avg}

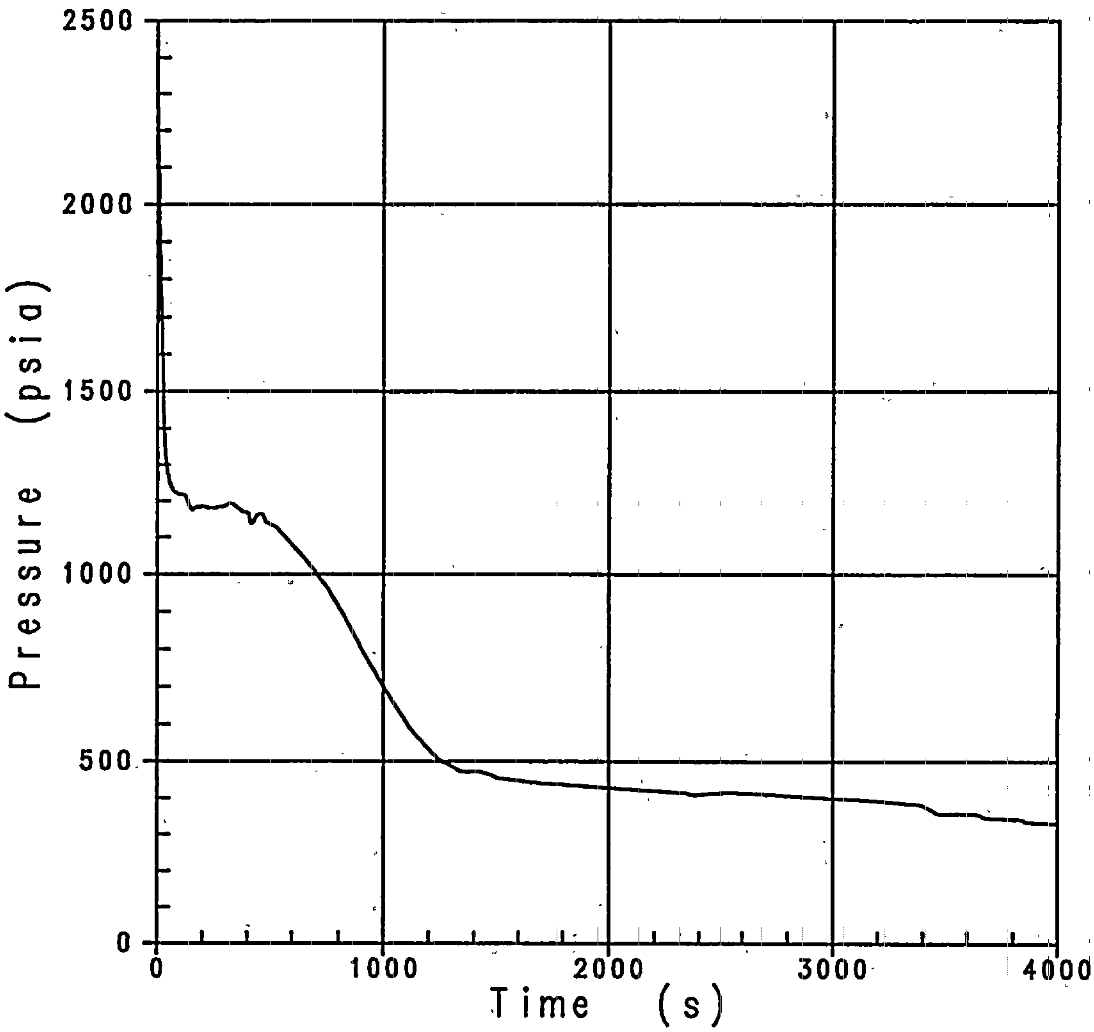

Figure 3.3.2-19: RCS Depressurization Transient, 3-Inch Break, Low T_{avg}

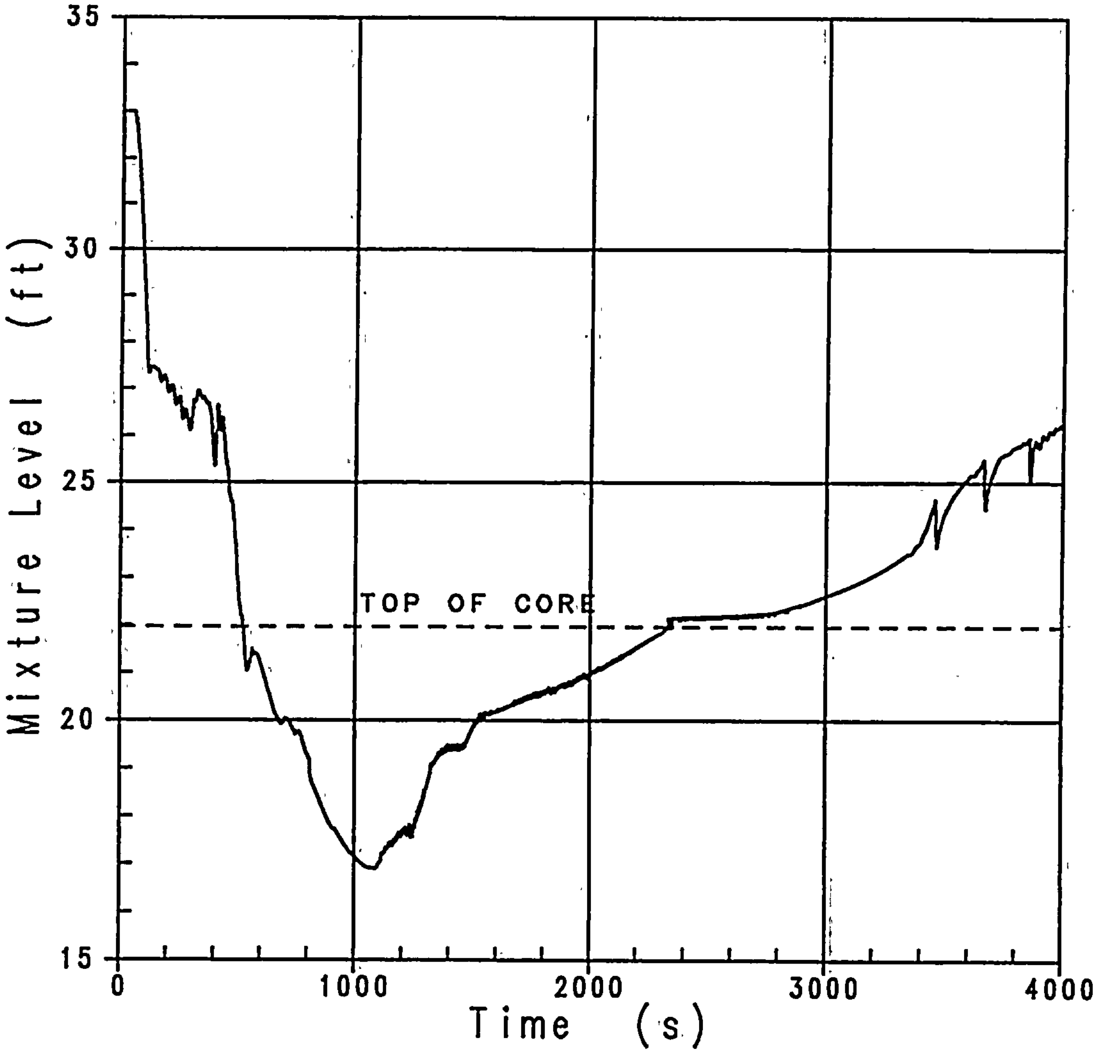

Figure 3.3.2-20: Core Mixture Level, 3-Inch Break, Low T_{avg}

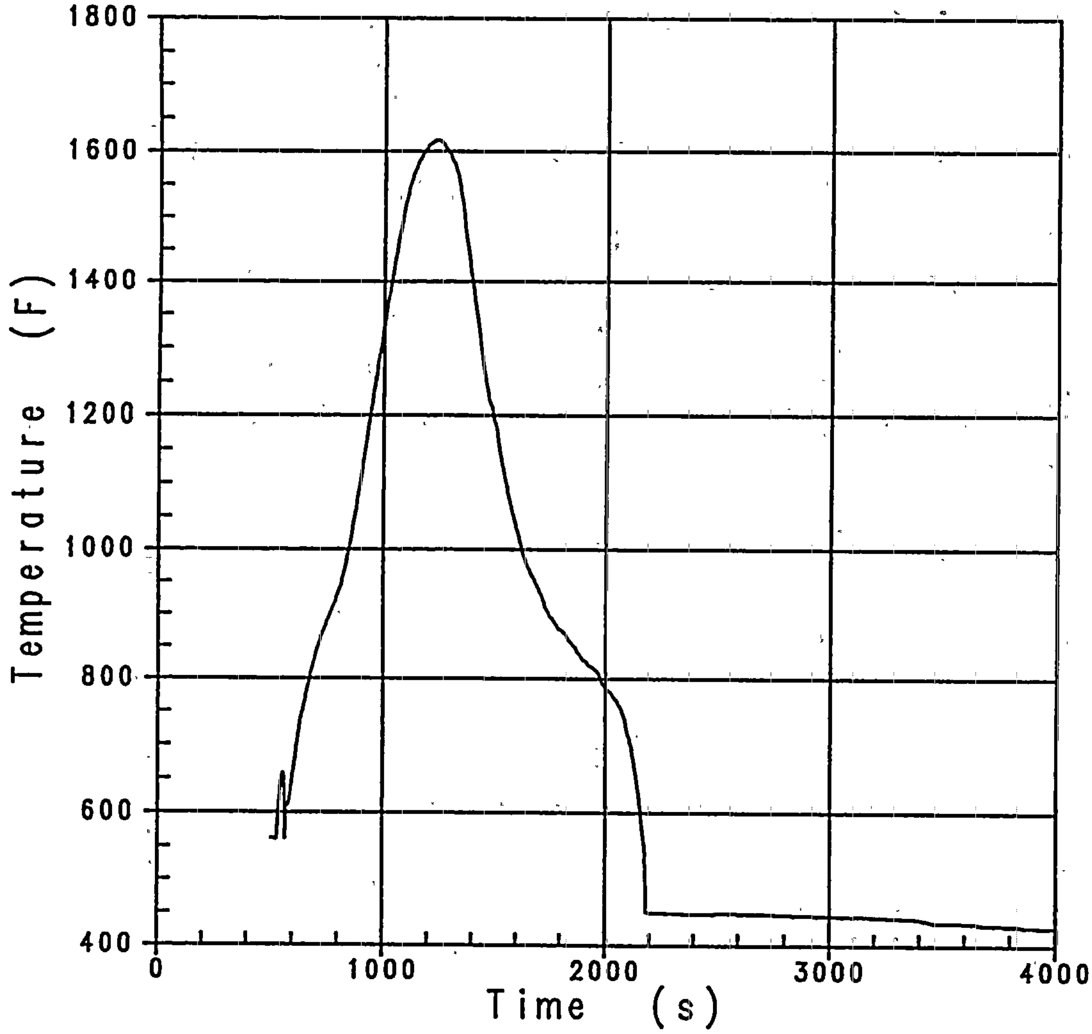

Figure 3.3.2-21: Peak Cladding Temperature - Hot Rod, 3-Inch Break, Low T_{avg}

3.3.3 LOCA Hydraulic Forces

3.3.3.1 Introduction

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and hydraulic loads that occur on Reactor Coolant System (RCS) components as a result of a postulated loss-of-coolant accident (LOCA). In general, LOCA hydraulic forces increase with an increase in RCS coolant density and, consequently, LOCA hydraulic forces increase with lower RCS temperatures. The lower RCS temperatures associated with the plant uprate requires that RCS components be evaluated relative to the higher forces associated with the reduced RCS temperatures.

The hydraulic forcing functions and loads that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The limiting break location and area varies with the RCS component under consideration but historically the limiting postulated breaks are a limited displacement reactor pressure vessel (RPV) inlet/outlet nozzle break or a double-ended guillotine (DEG) reactor coolant pump (RCP)/steam generator (SG) inlet/outlet nozzle break. The NRC's recent revision to GDC-4 allows main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping". This exemption is generally referred to as "leak-before-break" licensing. For Turkey Point, the applicability of a leak-before-break design basis was approved in (Reference 1) and was subsequently incorporated into the Turkey Point UFSAR in Revision 7 (July 1989). In addition, the NRC recently approved the base case LBB methodology for 2208 MWt (Reference 3). Previous UFSAR Turkey Point LOCA forces analyses did not take credit for the leak-before-break licensing basis. For the plant uprate, leak-before-break credit is used to evaluate the increased LOCA hydraulic forces.

Leak-before-break licensing allows RCS components to be evaluated for LOCA integrity considering the next most limiting auxiliary line breaks. For Turkey Point, the next most limiting auxiliary line breaks are the pressurizer surge line break (98.35 in^2) on the hot leg and the accumulator line break (60.19 in^2) on the cold leg. Postulated residual heat removal (RHR) auxiliary line breaks are bounded by the accumulator line break.

3.3.3.2 Input Parameters and Analysis Assumptions

The LOCA hydraulic forces analysis incorporates initial RCS condition uncertainties due to process measurement accuracy, instrumentation error, analog-to-digital signal processing, and environmental effects on transmitters. For LOCA hydraulic forces, a higher initial pressure is conservative so the uncertainty in pressurizer pressure is added to the nominal RCS pressure; since lower RCS temperatures are conservative, the maximum temperature uncertainty is subtracted from the RCS temperatures corresponding to the plant uprating conditions.

Steam generator and loop hydraulic forces are evaluated on the basis of established LOCA forces sensitivities to break size/location and RCS thermal/hydraulic conditions. The intent of the evaluations is to demonstrate that the increase in LOCA SG/loop hydraulic forces due to changes in RCS temperatures and pressure can be offset by the less severe accumulator line and pressurizer surge line breaks postulated under leak-before-break licensing. Note that the analyses of record assumed double-ended guillotine breaks which can be ignored in favor of these limiting auxiliary line breaks.

Pressure vessel/internals forces are analyzed (as opposed to evaluated) using the NRC approved MULTIFLEX 1.0 (Reference 2) computer code since the analysis of record already considered branch line breaks (as allowed under leak-before-break licensing). Consequently, no break area/location margin is available to offset the increase in vessel/internals hydraulic forces due to the plant uprating, therefore LOCA forces were calculated to show acceptable results.

3.3.3.3 LOCA Forces Analysis Acceptance Criteria and Results

3.3.3.3.1 Reactor Vessel and Vessel Internals

Vessel and vessel internals LOCA hydraulic forcing functions were generated using two postulated auxiliary line breaks. An accumulator line break was analyzed using a flexible beam core barrel MULTIFLEX model (for fluid-structure interaction) and a pressurizer surge line break was analyzed using the more conservative rigid core barrel model. Using these auxiliary line breaks and the new RCS conditions, the vessel/internals LOCA hydraulic forces were computed and the results (horizontal and vertical LOCA hydraulic forces) were used for the structural analysis.

The results of this analysis were compared with the previous (analysis of record) LOCA hydraulic forces analysis which supported the implementation of the Debris Resistant Fuel Assembly (DRFA) at Turkey Point. The pipe break considered in the prior analysis was an accumulator line break; the pressurizer branch line break was not considered. Comparing peak horizontal forces on the core barrel, reactor vessel, and thermal shield, it was apparent that the differences between the analysis of record and the present analysis were minimal (typically less than 5%) up to 100 msec for the accumulator line break. After 100 msec, the peak horizontal forces were somewhat greater (typically 10-20%) for the present analysis although the peak forces for both analyses were decaying with time. While LOCA horizontal forces at the uprated conditions were expected to increase throughout the transient according to established sensitivities, the results were judged to be acceptable in light of the coupling of the structural and hydraulic systems and relatively small break area (accumulator branch line versus DEG). With regards to the vertical forces on reactor internals, the change in forces was reasonable and consistent with the revised plant operating conditions, namely, colder fluid temperatures, lower thermal design flow, and higher initial RCS pressure (due to greater uncertainty in pressurizer pressure).

3.3.3.3.2 RCS Loop Piping and Steam Generators

Hydraulic forcing functions on the RCS loop piping and steam generators were evaluated using established LOCA forces sensitivities to changes in RCS temperatures and reduced break area associated with leak-before-break licensing; LOCA loop and steam generator forces were last analyzed assuming postulated DEG pipe breaks. As a result of the plant uprating, RCS temperatures were reduced in comparison to the analyses of record, resulting in an increase in loop and steam generator forces. However, the increase in LOCA loop/SG forces due to lower RCS temperatures was offset by less severe accumulator and pressurizer surge line breaks postulated under leak-before-break licensing. Therefore, it was concluded that the leak-before-break credit offsets the increase in loop/SG forces due to lower temperatures and that the analyses of record forcing functions remain bounding for these components.

3.3.3.4 Conclusion

The LOCA hydraulic forces analysis for Turkey Point in support of the plant uprating incorporated a 8°F reduction in T_{avg}, which bounds the uprating low-temperature conditions shown in Table 2.1-1.

The forces analysis of the reactor vessel/internals was based on the MULTIFLEX (Reference 2) computer code and associated post-processors. The postulated break locations included two limiting branch line breaks, i.e, the accumulator and pressurizer surge lines, as allowed under leak-before-break licensing. The MULTIFLEX analysis assumed bounding uprated conditions and incorporated plant initial condition uncertainties. The results of the analysis, namely, horizontal and vertical LOCA hydraulic forces, were stored on computer files for access by the cognizant structural analysts within Westinghouse.

For the RCS loop piping and steam generators, evaluations were performed using established sensitivities to show that the existing forces (double ended guillotine breaks as described in the UFSAR) remain bounding due to the reduction in effective break area as allowed under leak-before-break licensing.

3.3.3.5 References

1. NRC Letter, from G. E. Edison (NRC) to W. F. Conway (FPL), "Turkey Point Units 3 and 4 - Generic Letter 84-04, Asymmetric LOCA Loads", dated November 28, 1988.

2. Takeuchi, K., et. al., "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics", WCAP-8708-PA-V1 (Proprietary), WCAP-8709-A (Non-Proprietary), September, 1977.

3. NRC Letter from R. P. Croteau (NRC) to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak Before Break Methodology for Reactor Coolant System Piping," dated June 23, 1995.

3.3.4 Hot Leg Switchover

Post-LOCA Hot Leg Switchover (HLSO) time is calculated for inclusion in emergency operating procedures to ensure there is limited boron precipitation in the reactor vessel following boiling in the core after a cold leg break LOCA. This calculation is dependent upon power level and the various boron concentrations of the RCS and ECCS.

The HLSO calculation is performed to show the acceptance criteria of 10 CFR 50.46 continue to be met for the increase in core power from 2200 MWt to 2300 MWt. Specifically, a new HLSO time is established at uprated conditions to show that boron concentrations will not build up to a point such that boron precipitation occurs. Excessive boron precipitation may result in a change in core geometry which is not amenable to cooling or reduced heat transfer capability such that heat can not be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Currently, a HLSO time of 18 hours is calculated for the Turkey Point Units based on a core power of 2200 MWt. Although the boron concentrations of the RCS and ECCS are not changing as a result of the uprating, the increase in the core power to 2300 MWt necessitates a recalculation of the HLSO time and hot leg recirculation minimum required flow. The increase in core power will reduce the HLSO time from the current value.

The new HLSO time based on an uprated core power 2300 MWt is 12 hours. Since the HLSO time has been reduced, a revised hot leg recirculation minimum required flow was calculated. Based on plant specific criteria established by Westinghouse, sufficient flow must be delivered to the core during the hot leg recirculation phase such that 1.67 times core boiloff is available at the revised HLSO time. The revised hot leg recirculation minimum flow requirement is 33 lbm/sec. This hot leg recirculation minimum flow has been shown to be available. Finally, a revised hot leg / cold leg recirculation cycling time has been calculated based on uprated conditions. The new requirements for cycling between hot leg injection and cold leg injection post-LOCA is 12 hours after initially switching over to hot leg recirculation and every 24 hours after that.

In conclusion, a new HLSO time, minimum flow requirement for hot leg recirculation and cycling time have been established for the uprating project. It has been shown that, for the uprated conditions, the core geometry will remain amenable to cooling and decay heat can be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

3.3.5 Post-LOCA Long Term Core Cooling

The Westinghouse licensing position for satisfying the requirements of 10 CFR 50.46 Paragraph (b) Item (5), "Long-Term Cooling", is documented in Reference 1. The Westinghouse position is that the core will remain subcritical post-LOCA by borated water from the various ECCS water sources residing in the RCS and containment sump. Since credit for control rods insertion is not taken for Large Break LOCA, the borated ECCS water provided by the accumulators and RWST must have a sufficiently high boron concentration that, when mixed with other sources of borated and non-borated water, the core will remain subcritical assuming all control rods out.

Although uprated power is not part of this calculation, the Tavg range will have an affect on the fluid masses used in the calculation. During post-LOCA long term cooling, the safety injection flow is drawn from the containment sump following switchover from the RWST. The calculations performed by Westinghouse to determine the containment sump boron concentration include the water mass of the RCS. Since the Tavg range will lower the RCS operating temperature, which will increase the density of the fluid, there is a potential for the post-LOCA sump boron concentration to decrease. However, the effect of this density change on RCS water mass is relatively small, and within the accuracy of the calculation. In addition, the RWST water mass, which is more important in the calculation, is unaffected by this Tavg range. Therefore, the Tavg range has a negligible effect on the post-LOCA sump boron concentration calculation.

In conclusion, the uprated conditions including Tavg range have been considered and it is concluded that the core will remain subcritical post-LOCA and that decay heat can be removed for the extended period of time required by the long-lived radioactivity remaining. The revised post-LOCA long term core cooling boron limit curve is used to qualify the fuel on a cycle-by-cycle basis during the fuel reload process.

3.3.5.1 Reference

1. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339 (Non-Proprietary), July 1974.

3.4 STEAM GENERATOR TUBE RUPTURE

3.4.1 Identification of Causes and Accident Description

The complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser, the atmospheric dump valves (ADV) or safety relief valves (SRV). In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming of the SGs following the accident. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from this release.

The purpose for performing SGTR event analysis is to establish the offsite doses resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator (SG) and subsequent release of radioactivity to the atmosphere. Acceptance criteria for offsite doses are expressed as maximum allowed whole-body and thyroid doses at the exclusion area boundary and low population zone. The primary thermal/hydraulic parameters which affect the calculation of offsite doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator and the amount of steam released from the ruptured steam generator to the atmosphere.

The event analyzed is the double-ended rupture of a single steam generator tube as documented in UFSAR, Rev. 12 (Section 14.2.4). It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the reactor coolant system (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. Loss of offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric dump valves and/or safety valves. Following SI actuation, it is assumed that the RCS pressure stabilizes at the value where the SI and break flow rates are equal. The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR, at which time it is assumed that the operators have completed the actions necessary to terminate the break flow and the steam release from the ruptured steam generator.

After 30 minutes, it is assumed in the UFSAR analysis that steam is released only from the intact steam generators in order to dissipate the core decay heat and to subsequently cool the plant down to the residual heat removal (RHR) System operating conditions. During post-SGTR cooldown, the pressure in the affected steam generator is assumed to be decreased by the backfill method (ES 3.1) which is the preferred approach since it minimizes the radioactivity released to the atmosphere. Use of alternate post-SGTR cooldown procedures ES 3.2 (steam generator blowdown) or ES 3.3 (atmospheric steam dump) would result in an increase in the offsite doses, however, the increase is expected to remain within the 10 CFR 100 acceptance criteria. For Turkey Point Units 3 and 4, it is assumed that plant cooldown to RHR operating conditions is accomplished within 24 hours after initiation of the

SGTR and that steam releases are terminated at this time. A primary and secondary side mass and energy balance is used to calculate the steam release and feedwater flow for the intact steam generators from 0 to 2 hours and from 2 to 24 hours.

3.4.2 Input Parameters and Assumptions

A steam generator tube rupture (SGTR) thermal/hydraulic analysis for offsite activity release has been performed. The SGTR analysis incorporates a ± 3°F Tavg window about the current licensed Tavg of 574.2°F as part of the plant uprating effort. Plant secondary side conditions (e.g., steam pressure, flow, temperature) are based on (1) 0% steam generator tube plugging (SGTP) to reflect expected conditions at the uprated power level with the steam generators in their current condition (< 1% SGTP) and (2) 20% SGTP to reflect lower steam pressure and temperature at the maximum tube plugging condition. The SGTR analysis incorporates a total T_{avg} reduction of 8°F, which bounds the uprating conditions for low T_{avg} provided in Table 2.1-1.

The offsite doses following a steam generator tube rupture (SGTR) reflect the uprated power level of 2346 MWt and both pre-accident iodine spike and accident initiated iodine spikes (Reference 1). The assumptions used in the SGTR dose analysis are summarized in Table 3.4-2.

3.4.2.1 High Head Safety Injection (HHSI) and Charging Flow Rates

At Turkey Point, the charging (positive displacement) pumps automatically trip upon generation of an "SI" signal. However, plant Emergency Operating Procedures (EOP) instruct the operator to restart the positive displacement pumps (PDP) to establish charging flow. Consideration of charging pumps in operation concurrent with HHSI pumps increases total injection flow delivery to the RCS. A greater injection flow rate results in a greater RCS equilibrium pressure and, consequently, higher break flow. Thus, it is conservative to use the combined (HHSI + PDP) maximum injection flow rates in the SGTR analysis. For Turkey Point, a maximum charging pump flow capacity of 100 gpm is assumed which is added to the maximum (all four pumps operating) HHSI flow rate at each RCS pressure point.

3.4.2.2 RHR Cut-in Time

Twenty-four hours is conservatively assumed for the RHR cut-in time based on the RCS heat load and RHR heat removal capacity. This affects the duration of long term steam releases from the intact steam generators to the atmosphere following termination of the break flow. The effect of RHR cut-in time on long term doses, however, is not significant since the radiation emitted from the intact steam generators is small relative to that released by the ruptured steam generator.

3.4.2.3 Miscellaneous Parameters

The following parameters are assumed in the analysis:

- Low pressurizer pressure SI actuation setpoint = 1745 psia
- Lowest SG safety valve reseat pressure = 902 psia – includes 15% MSSV blowdown and 3% tolerance.

3.4.3 Description of Analyses

Multiple cases were analyzed, consistent with all of the parameter cases presented in Section 2.0 of this report.

These cases were individually analyzed in order to determine the steam releases for the offsite dose evaluation between 0 and 30 minutes (break flow termination). A single calculation is performed to calculate long term steam releases from the intact steam generators for the time intervals 0 to 2 hours and 2 to 24 hours (RHR cut-in time).

3.4.4 Acceptance Criteria

The offsite dose limits for a SGTR with a pre-accident iodine spike are the guideline values of 10 CFR 100 (Reference 1). These guideline values are 300 rem thyroid and 25 rem γ-body. For a SGTR with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem γ-body.

3.4.5 Results

The tube rupture break flow, atmospheric steam releases, and feedwater flows for the offsite dose analysis are summarized in Table 3.4-1. Note that the steam release from the ruptured steam generator due to failure of the hydraulic line connecting the radiation monitor to the main steam line is included in Table 3.4-1. This additional steam release is discussed later in this section. Also note that maximum steam release and break flow between 0 and 30 minutes (time of break flow termination) are based on two different SGTR cases: 1) high Tavg, 0% SGTP, and 2) low Tavg case, 20% SGTP (which bounds the uprating conditions). For a SGTR event, the amount of radioactivity released to the atmosphere is directly proportional to the amount of steam released through the ruptured steam generator safety valves. Consequently, the worst radiological consequences result from the SGTR case with the greatest amount of steam released. Likewise, a greater break flow results in greater radiological contamination of the secondary side which in turn results in a greater amount of activity released along with the steam. Maximum break flow and steam release, therefore, represent bounding values which are conservative for an offsite dose evaluation.

The SGTR thermal/hydraulic results for the plant uprating can be compared to the Turkey Point UFSAR (Section 14.2.4) results. The UFSAR (Reference 1) indicates that 79,718 lbm of reactor coolant is discharged into the steam generator and 48,534 lbm of steam are released to atmosphere during the 30-minute period to isolate the affected steam generator. In this analysis, 102,700 lbm (28.8% increase) of reactor coolant are discharged into the steam generator and 55,000 (13.3% increase) lbm of steam are released to atmosphere.

The 28.8% increase in primary-to-secondary break flow can be attributed to (1) a slightly higher RCS equilibrium pressure (1374.9 vs. 1337 psia) and (2) a significantly lower steam generator pressure (902 vs. 1100 psia) following reactor trip. Both factors contribute to a larger primary-to-secondary pressure drop and, hence, larger break flow rate for the plant uprate. Note that the higher RCS equilibrium pressure is due to consideration of the positive displacement charging pump in operation concurrent with the four HHSI pumps; the lower steam generator pressure following reactor trip is due to an increase in the assumed MSSV blowdown to 15% and increase in MSSV tolerance to 3%.

The 13.3% increase in steam released to atmosphere during the 30-minute period to isolate the ruptured steam generator is due to the following factors: (1) 4.5% increase in plant power, (2) greater RCS metal/fluid stored energy due to higher initial Tavg (577.2°F vs. 574.2°F), (3) lower MSSV setpoint as discussed above, and (4) greater primary-to-secondary break flow (102,700 lbm vs. 79,718 lbm).

The SGTR analysis also considered additional atmospheric steam releases from the ruptured steam generator due to failure of the radiation monitor (RAD-6426) line and minor leakages on the secondary steam and/or feed side of the steam generator.

The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary are as follows:

	EB (0-2 Hr)	LPZ (0-24 Hr)
Thyroid: Accident Initiated Spike	6.8 E-2	1.0 E-2
Thyroid: Pre-Accident Spike	4.1 E-1	4.5 E-2
γ-Body	2.0 E-2	2.0 E-3

3.4.6 Conclusion

The SGTR thermal/hydraulic analysis for offsite activity release has been completed in support of the uprating. Based on a primary and secondary side mass and energy balance, the break flow and atmospheric steam releases from the ruptured and intact steam generators were calculated for 30 minutes. After 30 minutes, it was assumed that steam is released only from the intact steam generators in order to dissipate the core decay heat and to subsequently cool the plant down to the RHR Systems operating conditions. For Turkey Point Units 3 and 4, it was assumed that plant cooldown to RHR operating conditions can be accomplished within 24 hours after initiation of the SGTR and that steam releases are terminated at this time. A primary and secondary side mass and energy balance was used to calculate the steam release and feedwater flow for the intact steam generators from 0 to 2 hours and from 2 to 24 hours. In addition, minor leakage due to failure of radiation monitor (RAD-6426) line between 30 minutes (time of break flow termination) and 8.5 hours (time at which operator isolates leakage) was added to the overall steam releases to the atmosphere. The increase in radioactivity released to the atmosphere as a result of this leakage was insignificant in comparison with the total.

The SGTR thermal/hydraulic results for this analysis were compared to the Turkey Point UFSAR, Rev. 12 results. As a result of the plant uprating and associated conditions, primary-to-secondary break flow and steam releases were increased.

The offsite thyroid and γ-body doses for the SGTR are within the acceptance criteria in Section 3.4.4.

3.4.7 Reference

1. Turkey Point Units 3 and 4 UFSAR, Revision 12.

Table 3.4-1

SGTR Thermal/Hydraulic Results for Radiological Analysis

	Time	lbm
Tube Rupture Break Flow	0 – 30 minutes	102,700
Steam Release from Ruptured SG	0 – 30 minutes	55,000
Steam Release from Ruptured SG	0.5 – 8.5 hours[1]	2160
Steam Release from Intact SGs	0 – 2 hours	308,500
Steam Release from Intact SGs	2 – 24 hours	1,731,200
Feedwater Flow to Intact SGs	0 – 2 hours	280,100
Feedwater Flow to Intact SGs	2 – 24 hours	1,769,600
Initial Ruptured SG Water Mass	(at time zero)	83,800
Final Ruptured SG Water Mass	(at ≥ 30 minutes)	96,700

[1] This steam release is due to failure of the hydraulic line connecting radiation monitor RAD-6426 to the main steam line. A leak rate of 270 lbm/hr is assumed.

Table 3.4-2

Assumptions for SGTR Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 µCi/gm of DE I-131
Accident Initiated Spike	1.0 µCi/gpm of DE I-131
Reactor Coolant Iodine Activity Increase Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.6 hours after SGTR
Secondary Coolant Activity Prior to Accident	0.10 µCi/gm of DE I-131
SG Tube Leak Rate for Intact SGs During Accident	500 gpd per SG
Break Flow to Ruptured SG	102,700 lb (0-30 min)
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	24 hr
Offsite Power	Lost
Steam Release from SGs to Environment	
Ruptured SG	55,000 lb (0-30 min) 2,160 lb (0.5 - 8.5 hr)(1)
Intact SGs	308,500 lb (0-2 hr) 1,731,200 lb (2-24 hr)

(1) Due to failure of hydraulic line connecting radiation monitor RAD-6426 to the main steamline. A leak rate of 270 lb/hr is assumed.

3.5 CONTAINMENT INTEGRITY ANALYSES

3.5.1 Main Steam Line Break (MSLB) Mass and Energy (M&E) Releases

3.5.1.1 Identification of Causes and Accident Description

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make a reasonable determination of the single absolute worst case for both containment pressure and temperature evaluations following a steamline break difficult. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the containment response to a secondary system pipe rupture.

In addition to the inside containment analyses performed for containment integrity, an analysis was performed for an outside containment steamline break to determine radiological consequences for the uprated conditions.

3.5.1.2 Input Parameters and Assumptions

The postulated break area can have competing effects on blowdown results. Larger break areas will be more likely to result in large amounts of water being entrained in the blowdown. However, larger breaks also result in earlier generation of protective trip signals following the break and a reduction of both the power production by the plant and the amount of high-energy fluid available to be released to the containment.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectrums of both variables have been evaluated. At plant power levels of 102%, 70%, 30% and 0% of nominal full-load power, four break sizes have been defined. These break areas are defined as the following.

1. A full double-ended rupture (DER) downstream of the flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other.

2. A small break at the steam generator nozzle having an area just larger than that at which water entrainment occurs.

3. A small break at the steam generator nozzle having an area just smaller than that at which water entrainment occurs.

4. A small split rupture that will neither generate a steamline isolation signal from the Westinghouse Engineered Safety Features nor result in water entrainment in the break effluent.

The cases examined in this study were chosen based on the results of the analyses presented in Reference 1 for Turkey Point Units 3 and 4. The most-limiting case with respect to peak containment pressure was analyzed at the uprated power condition. Initial containment conditions for this limiting case were assumed to be +3.0 psig and 130°F. This case was a 1.4 ft^2 (based on the steam nozzle flow-limiter cross-sectioned area) DER at hot-zero-power (HZP) conditions. This DER steamline break was modeled assuming isolation is accomplished by the main steam isolation valve in each intact steamline. The important plant conditions and features that were assumed are discussed in the following paragraphs.

Initial Power Level

Steamline breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during "at-power" operation may be greater than for breaks occurring with the plant in a hot-shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power and have a significant influence on both the rate of blowdown and the amount of moisture entrained in the fluid leaving the break.

Because of the opposing effects (mass versus energy release) of changing power level on steamline break releases, no single power level can be singled out as a worst case initial condition for a steamline break event. Therefore, several different power levels spanning from full- to zero-power conditions have been investigated for Turkey Point Units 3 and 4 as discussed in Reference 1. For this power uprating analysis, only the power level corresponding to the steamline break mass-and-energy releases resulting in the limiting containment pressure response is included.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power. Table 3.5.1-1 identifies the values assumed for RCS pressure, RCS vessel average temperature, pressurizer water volume, steam generator water level, and feedwater enthalpy corresponding to the limiting steamline break case analyzed.

Single-Failure Assumption

To avoid unnecessary conservatism, bounding multiple failure assumptions were not made in the analysis. Only one single failure was considered in the analysis. The Main Steam Isolation Valve Assembly in each steamline consists of the main steam isolation valve (MSIV) and the main steam check valve (MSCV). The MSIV closes upon an isolation signal to terminate steam flow from the associated steam generator. The MSCV is designed to prevent reverse steam flow in the steamline, thus preventing blowdown from more than one steam generator for any break inside containment. However, if the MSCV in the faulted loop is assumed to fail, the intact steam generators would blow down through the break until the MSIVs in the intact loops close. This could result in significant additional mass and energy release to containment. The assumption that both the MSIV and the MSCV in the faulted loop fail exceeds the current UFSAR analysis assumptions. The intent of this assumption is to show that the protection logic which provides a signal to close the MSIVs, and the associated delay time, is adequate to limit the amount of steam mass and energy discharged into containment such that the containment pressure limit is not exceeded. To do this, no credit is taken for the proper functioning of the MSCV in preventing reverse steam flow from the intact steam generators.

Main Feedwater System

Main feedwater flow was conservatively modeled by assuming an initial increase in feedwater flow (until fully isolated) in response to increases in steam flow following initiation of the steamline break. This maximizes the total mass addition prior to feedwater isolation. The steamline break case of Reference 1 which resulted in the limiting containment pressure response occurred from a hot-zero-power condition. During actual plant operation, the main feedwater valves are not in service at power levels up to approximately 15-20% of full power; rather, the 4-inch feedwater bypass valves are used to provide flow to the steam generators. The flows through the 4-inch feedwater bypass valves as a function of steam generator pressure was generated for both the faulted and the intact loops. The feedwater isolation response time was governed by the response time of the feedwater bypass valves and was assumed to be a total of 13 seconds following the safety injection signal.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation valve may flash to steam if the feedwater temperature exceeds the saturation pressure. This unisolable feedwater line volume is an additional source of high-energy fluid that was assumed to be discharged out of the break. The unisolable volume in the feedwater lines are maximized for the faulted loop and minimized for the intact loop. The energy in the unisolable volume is maximized by assuming recirculated feedwater from the condenser rather than "cold" water from the demineralized water storage tank. The following piping volumes available for steam flashing were calculated from plant drawings and assumed in the analysis.

Volume from SG nozzle to FCV (faulted loop) - 238 ft^3
Volume from SG nozzle to FCV (intact loops) - 75 ft^3/loop

Auxiliary Feedwater System

Generally, within the first minute following a steamline break, the auxiliary feedwater system will be initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The auxiliary feedwater flow control valves are set to supply a fixed flow to each steam generator, regardless of the backpressure in the steam generator. The maximum AFW flowrate has been determined to be 254 gpm/FCV (1 FCV per AFW train, 2 AFW trains per SG; therefore, the total AFW flowrate is 508 gpm/SG) for the first 120 seconds, decreasing to 140 gpm/FCV (total AFW flowrate is 280 gpm/SG) for the remainder of the event. A higher AFW flowrate to the faulted loop steam generator is conservative for the steamline break event; consequently, 254 gpm/FCV for 120 seconds decreasing to 140 gpm/FCV was assumed for the faulted loop steam generator AFW flowrate. Conversely, a lower AFW flowrate is conservative for the intact loop steam generators; thus, a constant 140 gpm/FCV was assumed for each intact loop for the entire transient.

Steam Generator Fluid Mass

Maximum initial steam generator masses in the faulted loop steam generator were used in both of the analyzed cases. The use of high initial steam generator masses maximizes the steam generator inventory available for release to containment. The initial masses were calculated as the mass corresponding to the programmed level +6% narrow range span. Minimum initial steam generator masses in the intact loops steam generators were used in both of the analyzed cases. The use of reduced initial steam generator masses minimizes the availability of the heat sink afforded by the steam generators on the intact loops. The initial masses were calculated as the mass corresponding to the programmed level -6% narrow range span. All steam generator fluid masses are calculated corresponding to 0% tube plugging which is conservative with respect to the RCS cooldown through the faulted loop steam generator resulting from the steamline break. The water mass defined by the unisolable portion of the steam generator blowdown recovery system is accounted for as part of an overall mass uncertainty applied to the steam generator initial conditions. This mass uncertainty is applied to both the faulted and intact steam generators and is in addition to the programmed 6% narrow range span level uncertainty previously mentioned.

Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, those steam generators in the intact steam loops become sources of energy which can be transferred to the steam generator with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact steam generators resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. The effects of reverse steam generator heat transfer are included in the results.

Break Flow Model

Piping discharge resistances were not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an f (ℓ / D) = 0 was used].

Core Decay Heat

Core decay heat generation assumed is based on the 1979 ANS Decay Heat + 2σ model (Reference 2).

Steamline Volume Blowdown

The contribution to the mass and energy releases from the secondary plant steam piping was included in the mass and energy release calculations. The flowrate was determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. For the limiting steamline DER case analyzed for the power uprating, the unisolable steamline mass is included in the mass exiting the break from the time of steamline isolation until the unisolable mass is completely released to containment.

Main Steamline Isolation

The postulated single failure for these two cases is the failure to close the MSCV in the faulted loop. In this instance, MSIV closure in the intact loops is required to terminate the blowdown. A delay time of 7 seconds was assumed (2-second signal processing plus 5-second valve closure) with full steam flow assumed through the valve during the valve stroke. The assumption of full steam flow from the intact steam generators for this time conservatively accounts for the effects of the unisolable steamline volume which would be released following closure of the MSIVs.

Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick metal masses and heat transfer coefficients.

Rod Control

The rod control system was assumed to be in manual operation for the steamline break analyses.

Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, feedwater isolation, emergency fan coolers, and containment spray. The first protection system signal actuated was High Containment Pressure (2-of-3 channels) which initiated safety injection; the safety injection signal produced a reactor trip signal. Feedwater system and steam generator blowdown recovery system isolation also occurred as a result of the safety injection signal. Finally, steamline isolation occurred via a High Steam Flow in 2-of-3 steamlines (1-of-2 channels per steamline) coincident with a Low T-avg SI signal in 2-of-3 loops.

Safety Injection System

Minimum safety injection system (SIS) flowrates corresponding to the failure of one SIS train (2-of-4 pumps) were assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow was assumed to be 23 seconds for this analysis.

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions, including HZP stuck-rod moderator density coefficients, were used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns critical, thus maximizing heat transfer to the secondary side of the steam generators.

3.5.1.3 Description of Analysis

The break flows and enthalpies of the steam release through the steamline break is analyzed with the LOFTRAN (Reference 3) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer.

The Turkey Point NSSS is analyzed using LOFTRAN to determine the transient steam mass and energy releases inside containment following a steamline break event. The tables of mass and energy releases are used as input conditions to the analysis of the containment response as discussed in Section 3.5.4.

The single most-limiting case with respect to peak containment pressure, based on the results in Reference 1 was analyzed: a 1.4 ft^2 DER at hot-zero-power (HZP) conditions.

The DER steamline break event was modeled taking credit only for MSIV closure on the intact loops for steamline isolation.

3.5.1.4 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. Additional clarification of the ANS classification of this event is presented in Section 3.2.16 of this report, which discusses the core response to a steamline break event. The acceptance criteria associated with the steamline break event resulting in a mass and energy release inside containment is based on providing sufficient conservatism in the analysis to assure that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model including entrainment, main and auxiliary feedwater flow, steamline and feedwater isolation, blowdown recovery system isolation, and single failure such that the containment peak pressure is maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in Reference 1 and Section 3.5.1.2 of this report. The tables of mass and energy release for the limiting steamline break case noted in the previous section are used as input to a containment response calculation to confirm the design pressure limit of the Turkey Point containment structure.

3.5.1.5 Results

Using Reference 1 as a basis, including parameter changes associated with the power uprating, the mass and energy release rates were developed to determine the containment pressure response for the limiting steamline break case noted in Section 3.5.1.3. The mass and energy releases from the 1.4 ft^2 DER at HZP conditions resulted in the highest containment pressure. The steam mass and energy releases discussed in this section provide the basis for the containment response described in Section 3.5.4 of this report. Table 3.5.4-6 provides the sequence of events for the limiting steamline break inside containment.

3.5.1.6 Conclusions

The mass and energy releases from the steamline break case resulting in the limiting containment pressure response identified in Reference 1 has been analyzed at the uprated power conditions. The assumptions delineated in Section 3.5.1.2 have been included in the steamline break analysis such that the applicable acceptance criteria are met. The steam mass and energy releases discussed in this section provide the basis for the containment response described in Section 3.5.4 of this report.

3.5.1.7 References

1. Gresham, J. A., Heberle, G. H., Wills, M. E. and Scobel, J. H., "Analysis of Containment Response Following a Main Steam Line Break for Turkey Point Units 3 and 4," WCAP-12262 (non-Proprietary), August 1989

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (non-Proprietary)," April 1984

Table 3.5.1-1

Nominal Plant Parameters and
Initial Condition Assumptions *
(MSLB M&E Releases)

NOMINAL CONDITIONS	
NSSS Power, MWt	2311.4
Core Power, MWt	2300
Reactor Coolant Pump Heat, MWt	11.4
Reactor Coolant Flow (total), gpm	255,000
Pressurizer Pressure, psia	2250
Core Bypass, %	6.0
Reactor Coolant Temperatures, °F	
Core Outlet	611.3
Vessel Outlet	607.8
Core Average	580.5
Vessel Average	577.2
Vessel/Core Inlet	546.6
Steam Generator	
Steam Temperature, °F	522.8
Steam Pressure, psia	832
Steam Flow (total), 10^6 lbm/hr	10.17
Feedwater Temperature, °F	443
Zero-Load Temperature, °F	547

INITIAL CONDITIONS POWER LEVEL (%)

PARAMETER	102	0
RCS Average Temperature (°F)	583.2*	547.0
RCS Flowrate (gpm)	255,000	255,000
RCS Pressure (psia)	2250	2250
Pressurizer Water Volume (ft^3)	688.6	321.9
Feedwater Enthalpy (Btu/lbm)	424.9	70.68
SG Water Level, faulted/intact (% span)	66/54	56/44

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T-avg window.

3.5.2 Steamline Break Radiological Consequences

3.5.2.1 Introduction of Causes and Accident Description

The complete severance of a main steamline outside containment is assumed to occur. The affected Steam Generator (SG) will rapidly depressurize and release radioiodines initially contained in the secondary coolant and primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to tube leakage is released to atmosphere through either the atmospheric dump valves (ADV) or the safety relief valves (SRV). This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from this release.

3.5.2.2 Input Parameters and Assumptions

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). These along with plant specific assumptions are summarized in Table 3.5.2-1.

3.5.2.3 Description of Analyses

The radiological consequences of a SLB are analyzed with both the pre-accident and accident initiated iodine spike models. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 60 μCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike the reactor trip associated with the steamline break (SLB) creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 1.0 μCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.6 hours.

3.5.2.4 Acceptance Criteria

The offsite dose limits for a SLB with a pre-accident iodine spike are the guideline values of 10 CFR 100. These guideline values are 300 rem thyroid and 25 rem γ-body. For a SLB with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10 CFR 100 guideline values, or 30 rem thyroid and 2.5 rem γ-body.

3.5.2.5 Results

The calculated thyroid and γ-body doses (rem) at the exclusion boundary and low population zone outer boundary are as follows:

	EB (0-2 Hr)	LPZ (0-24 Hr)
Thyroid: Accident Initiated Spike	4.2 E-1	1.1 E-1
Thyroid: Pre-Accident Spike	5.2 E-1	1.1 E-1
γ-Body	1.9 E-4	4.6 E-5

3.5.2.6 Conclusions

The offsite thyroid and γ-body doses due to the SLB are within the acceptance criteria in Section 3.5.2.4.

3.5.2.7 References

1. NUREG-0800, Standard Review Plan 15.1.5, Appendix, A, "Radiological Consequences of Main Steam Line Failures Outside of a Containment," Rev. 2, July 1981.

Table 3.5.2-1

Assumptions Used for SLB Dose Analysis

Power	2346 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 µCi/gm of DE I-131
Accident Initiated Spike	1.0 µCi/gm of DE I-131
Reactor Coolant Iodine Activity Increase Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.6 hours after SLB
Secondary Coolant Activity Prior to Accident	0.10 µCi/gm of DE I-131
SG Tube Leak Rate During Accident	500 gpd per SG
Iodine Partition Factors	
Faulted SG	1.0 (SG assumed to steam dry)
Intact SGs	0.01
Duration of Activity Release from Secondary System	24 hr
Offsite Power	Lost
Steam Release from SGs	
Faulted SG	84,128 lb (0-2 hr)
Intact SGs	269,700 lb (0-2 hr) 369,300 lb (2-8 hr) 984,700 lb (8-24 hr)

3.5.3 LOCA M&E Releases

3.5.3.1 Introduction

The purpose of this analysis was to calculate the long-term Loss-of-Coolant Accident (LOCA) mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot leg (DEHL) rupture break cases with the uprated conditions for the Turkey Point Units 3 and 4 Thermal Uprating Program.

The uncontrolled release of pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in an increase in the containment pressure and temperature. The mass and energy release rates described in this section form the basis of further computations to evaluate the structural integrity of the containment following a postulated accident (see Section 3.5.4).

3.5.3.2 Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Some of the most-critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed below. Tables 3.5.3-1 and 3.5.3-2 present key data assumed in the analysis.

For the long-term mass and energy release calculations, operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The modeled core power was 2346 MWt, adjusted for calorimetric error (+2 percent of power). The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The initial reactor coolant system (RCS) pressure in this analysis is based on a nominal value of 2250 psia plus an allowance which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the core stored energy. The margin in core stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy release calculation considered configurations/ failures to conservatively bound respective alignments. A spectrum of cases included:

(a) a Diesel Failure (1 HHSI, 1 LHSI, & 1 CSS Pump);
(b) a Containment Spray Pump Failure (2 HHSI, 2 LHSI, & 1 CSS Pump); and
(c) a No Failure Case (2 HHSI, 2 LHSI, & 2 CSS Pumps).

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the reactor coolant system (100% full-power conditions)

2. An allowance in temperature for instrument error and dead band (+7.4°F)

3. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. 102% of core rated power, 2346 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. A margin in core stored energy (+15 percent included to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+70 psi)

10. A maximum containment backpressure equal to design pressure

11. Allowance for RCS flow uncertainty (-3.5%)

12. Steam generator tube plugging leveling (0% uniform)

- Maximizes reactor coolant volume and fluid release
- Maximizes heat transfer area across the SG tubes
- Reduces coolant loop resistance, which reduces the Δp upstream of the break and increases break flow

Thus, based on the previously discussed conditions and assumptions, a bounding analysis of Turkey Point Units 3 and 4 is made for the release of mass and energy from the RCS in the event of a LOCA at 2346 MWt.

3.5.3.3 Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the NRC. It has also been utilized and approved on the plant-specific dockets for other Westinghouse PWRs such as Catawba Units 1 and 2, Beaver Valley Unit 2, McGuire Units 1 and 2, Millstone Unit 3, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, Surry Units 1 and 2, and Indian Point Unit 2.

This report section presents the long-term LOCA mass and energy releases that were generated in support of the Turkey Point Units 3 and 4 thermal uprating program. These mass and energy releases are then subsequently used in the containment integrity analysis presented in Section 3.5.4.

3.5.3.3.1 LOCA M&E Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases.

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill - the period of time when the lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two-phase.

3.5.3.3.2 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, and FROTH. These codes were used to calculate the long-term LOCA mass and energy releases for Turkey Point Units 3 and 4.

SATAN calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the Emergency Core Cooling refills the reactor vessel and provides cooling to the core. The most-important feature is the steam/water mixing model (see Section 3.5.3.5.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

3.5.3.3.3 Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture for M&E release purposes:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.48 ft^2), and the double-ended (DEHL) rupture (9.19 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators and vents directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, which results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this uprating.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment.

3.5.3.3.4 Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period which is limited by the DEHL break.

Three cases have been analyzed for the effects of a single failure. The first case postulated the single failure is the loss of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The second case is the assumed failure of a containment spray pump. As compared to the first case, the SI flow would be greater and the time of RWST depletion would be earlier. For the third case, no failure is postulated to occur that would impact the amount of ECCS flow. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

3.5.3.4 Acceptance Criteria for Analyses

A large break loss-of-coolant accident is classified as an ANS Condition IV event, an infrequent fault. The relevant requirements are as follows.

- 10 CFR 50, Appendix A
- 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed.

1. Sources of Energy
2. Break Size and Location
3. Calculation of Each Phase of the Accident

3.5.3.5 M&E Release Data

3.5.3.5.1 Blowdown Mass and Energy Release Data

A version of the SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 3.5.3-3 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Table 3.5.3-6 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

3.5.3.5.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model, in recent analyses, e.g., D.C. Cook docket (Reference 2). Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 1). This assumption is justified and supported by test data, and is summarized as follows.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most-important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most-applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3-scale tests (Reference 3), which are the largest scale data available and thus most-clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3-scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump is vented into containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References 1 and 3.

Tables 3.5.3-7 presents the calculated mass and energy release for the reflood phase of the pump suction double-ended rupture with a single limiting failure of a diesel generator. This failure case was the most-limiting for the LOCA containment integrity analysis (see Section 3.5.4) for the post-blowdown phase. Other failure scenarios were analyzed, but since the diesel failure is the most-limiting it will be presented. The other scenarios that were considered were a spray pump failure case and a no safeguards failure case.

The transients of the principal parameter during reflood are given in Table 3.5.3-8 for the DEPS diesel-failure case.

3.5.3.5.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 4) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. The methodology for the use of this model is described in Reference 1. The mass and energy release rates are calculated by FROTH until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Table 3.5.3-9 presents the two-phase post-reflood (FROTH) mass and energy release data for the DEPS diesel-failure case.

3.5.3.5.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS Standard 5.1 (Reference 5) for the determination of decay heat. This standard was used in the mass and energy release model.

Significant assumptions in the generation of the decay heat curve for use in design basis containment integrity LOCA analyses include:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11, of Reference 5 up to 10,000 seconds, and Table 10, of Reference 5 beyond 10,000 seconds.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70% of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model, use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a loss-of-coolant accident.

3.5.3.5.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is Tsat at the containment design pressure. After the FROTH calculations, steam generator secondary energy is removed based on first and second stage rates. The first stage rate is applied until the steam generator reaches Tsat at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of Tsat at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F.

3.5.3.5.6 Sources of M&E

The sources of mass considered in the LOCA mass and energy release analysis are given in Table 3.5.3-10. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Table 3.5.3-11. The energy sources include:

- Reactor Coolant System Water
- Accumulator Water
- Pumped Injection Water
- Decay Heat
- Core Stored Energy
- Reactor Coolant System Metal - Primary Metal (includes SG tubes)
- Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)
- Steam Generator Secondary Energy (includes fluid mass and steam mass)
- Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

Energy Reference Points

Available Energy: 212°F; 14.7 psia
Total Energy Content: 32°F; 14.7 psia

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint
7. Time of full depressurization (3600 seconds)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

3.5.3.6 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the acceptance criteria presented in Section 3.5.3.4 have been satisfied. Any other conclusions cannot be drawn from the generation of mass and energy releases directly since the releases are inputs to the containment integrity analyses. The containment response must be performed. See Section 3.5.4 for the LOCA containment integrity conclusions.

In contrast to the revised long-term LOCA M&E analyses for the thermal uprate program, the original design basis short-term LOCA mass and energy releases resulting from double-ended ruptures of the primary loop piping for the subcompartment analyses will remain bounding. This is due to the application of the Leak-Before-Break (LBB) Technology to the short-term LOCA M&E releases (Reference 6). Under LBB, the most-limiting break would be a double-ended rupture of one of the largest RCS loop branch lines (i.e., pressurizer surge line, accumulator/SI line, or RHR suction line).

3.5.3.7 References

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version", WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-proprietary)

2. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D.C. Cook Nuclear Plant Unit 1", June 9, 1989

3. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report June 1975

4. "Westinghouse Mass and Energy Release Data For Containment Design", WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Non-proprietary)

5. ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors", August 1979

6. Letter, G. E. Edison (NRC) to W. F. Conway (FPL), "NRC Generic Letter 84-04, Asymmetric Loads for Turkey Point Units 3 and 4", dated November 28, 1988.

Table 3.5.3-1

System Parameters Initial Conditions

PARAMETERS	VALUE
Core Thermal Power (MWt)	2346
Reactor Coolant System Total Flowrate (lbm/sec)	25,813.75
Vessel Outlet Temperature* (°F)	615.2
Core Inlet Temperature* (°F)	554.0
Vessel Average Temperature* (°F)	584.6
Initial Steam Generator Steam Pressure (psia)	832
Steam Generator Design	Model 44F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	103,501.2
Assumed Maximum Containment Backpressure (psia)	69.7
Accumulator	
Water Volume (ft^3)	920
N_2 Cover Gas Pressure (psia)	615
Temperature (°F)	130
Safety Injection Delay (sec)	35.0

* (analysis value includes an additional +7.4°F allowance for instrument error and deadband)

Table 3.5.3-2

Safety Injection Flow
Diesel Failure (Single Train)

INJECTION MODE (REFLOOD PHASE)

RCS Pressure (psig)	Total Flow (gpm)
0	3581.0
20	3318.0
40	3028.0
60	2705.0
80	2324.0
100	1772.0
120	562.0
140	557.0
160	551.0
180	546.0
200	540.0
300	511.0

INJECTION MODE (POST-REFLOOD PHASE)

RCS Pressure (psig)	Total Flow (gpm)
40	584.0

COLD LEG RECIRCULATION MODE

RCS Pressure (psig)	Total Flow (gpm)
0	2455.0

Table 3.5.3-3

Double-Ended Hot Leg Break
Blowdown Mass and Energy Releases

TIME (SECONDS)	BREAK PATH NO.1 FLOW* (LBM/SEC)	BREAK PATH NO.1 FLOW* THOUSAND (BTU/SEC)	BREAK PATH NO.2 FLOW** (LBM/SEC)	BREAK PATH NO.2 FLOW** THOUSAND (BTU/SEC)
.0000	.0	.0	.0	.0
.0502	52052.2	33058.2	27440.1	17291.7
.100	43931.8	27888.9	26452.2	16683.2
.150	35897.9	22981.1	24471.6	15407.5
.200	33326.1	21354.7	22866.7	14346.6
.251	33160.2	21218.4	21435.1	13371.4
.350	32570.3	20826.9	19771.1	12155.6
.451	31951.0	20439.4	18862.1	11414.2
.651	31684.6	20310.2	17657.4	10398.5
.801	30915.5	19905.6	17137.6	9933.6
1.00	30269.1	19678.5	16589.6	9456.4
1.10	29886.8	19540.7	16459.6	9316.9
1.30	28980.0	19164.3	16433.3	9188.6
1.50	27877.9	18666.2	16584.7	9178.9
1.70	26631.5	18065.9	16804.4	9225.9
2.00	24669.1	17049.7	17091.4	9307.7
2.50	21669.7	15305.2	17288.0	9354.7
3.00	19519.6	13836.8	17132.6	9254.5
3.50	18277.5	12801.6	16707.8	9031.4
4.00	18070.1	12415.4	16017.6	8682.6
4.50	18724.0	12411.8	14976.4	8157.9
5.00	19164.9	12391.8	13787.8	7561.6
5.50	19629.4	12455.8	12448.7	6872.4
6.00	15408.3	10487.2	11153.4	6194.5
6.50	15291.3	10332.4	10052.0	5613.3
7.00	14964.2	10046.5	9145.5	5132.4
7.50	14560.5	9662.1	8373.0	4722.1
8.00	14559.9	9506.0	7684.5	4358.0
8.50	14274.3	9216.9	7061.7	4031.3
9.00	13796.2	8844.6	6486.8	3733.1
9.50	13107.5	8386.8	5951.6	3459.7
10.0	12278.3	7880.0	5457.8	3212.0
10.5	11394.0	7366.6	5005.0	2989.3
11.5	9639.3	6403.6	4216.3	2611.3
12.0	8625.7	5886.7	3817.1	2422.7
13.0	6475.6	4922.5	2860.2	2002.0
13.5	5475.6	4495.4	2455.6	1809.2
14.0	4478.2	4080.0	2185.7	1656.7
14.5	3403.7	3450.0	2013.2	1544.0
15.0	2756.9	2981.2	1881.0	1447.5
15.5	2343.8	2615.5	1717.5	1367.6

Table 3.5.3-3 (cont.)

Double-Ended Hot Leg Break
Blowdown Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
		THOUSAND		THOUSAND
(SECONDS)	(LBM/SEC)	(BTU/SEC)	(LBM/SEC)	(BTU/SEC)
16.5	1705.4	2030.6	1409.9	1224.6
17.0	1456.8	1771.1	1284.2	1163.4
17.5	1076.7	1329.4	924.0	1092.3
18.0	993.6	1238.0	602.9	743.5
19.0	530.3	670.7	280.3	348.7
19.5	402.8	514.4	185.2	231.7
20.0	298.7	382.6	.0	.0
20.5	141.1	182.2	.0	.0
21.5	.0	.0	.0	.0

* Mass and Energy exiting from the reactor vessel side of the break

** Mass and Energy exiting from the S/G side of the break

Table 3.5.3-4

Double-Ended Hot Leg Mass Balance

	Time (Seconds)	.00	21.50	21.50
		Mass (Thousand lbm)		
Initial	In RCS and ACC	579.16	579.16	579.16
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
*** Total Available ***		579.16	579.16	579.16
Distribution	Reactor Coolant	403.94	50.05	93.69
	Accumulator	175.22	138.53	94.90
	Total Contents	579.16	188.58	188.58
Effluent	Break Flow	.00	390.56	390.56
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	390.56	390.56
*** Total Accountable ***		579.16	579.15	579.15

Table 3.5.3-5

Double-Ended Hot Leg Energy Balance

	Time (Seconds)	.00	21.50	21.50
		Energy (Million BTU)		
Initial Energy	In RCS, ACC, S/G	623.75	623.75	623.75
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	4.75	4.75
	Heat from Secondary	.00	-6.15	-6.15
	Total Added	.00	-1.40	-1.40
*** Total Available ***		623.75	622.35	622.35
Distribution	Reactor Coolant	237.49	13.09	17.43
	Accumulator	17.43	13.78	9.44
	Core Stored	23.36	11.01	11.01
	Primary Metal	118.73	111.46	111.46
	Secondary Metal	58.66	57.22	57.22
	Steam Generator (S/G)	168.07	162.68	162.68
	Total Contents	623.75	369.25	369.25
Effluent	Break flow	.00	253.09	253.09
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	253.09	253.09
*** Total Accountable ***		623.75	622.33	622.33

Table 3.5.3-6

Double-Ended Pump Suction Break
Blowdown Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
.0000	.0	.0	.0	.0
.0501	40934.2	22404.7	28380.4	15458.8
.100	40700.7	22324.0	21635.0	11808.3
.201	41067.2	22685.4	23122.8	12635.5
.301	41492.3	23129.7	24162.3	13211.5
.400	41955.2	23638.6	24282.2	13283.4
.500	42113.5	23999.6	23792.6	13020.9
.601	41711.5	24037.8	23164.5	12682.8
.701	40664.3	23672.0	22675.9	12421.6
.900	38327.9	22702.5	22172.8	12156.4
1.10	36612.3	22054.7	21699.8	11902.2
1.30	34733.0	21285.3	21198.8	11629.2
1.40	33920.5	20944.9	20986.1	11512.9
1.80	31411.9	20017.1	20217.2	11089.5
2.00	29608.8	19271.9	19522.5	10705.4
2.50	20674.6	14138.4	17630.2	9660.9
3.00	15463.2	10687.9	15998.0	8765.6
3.50	12005.3	8469.4	14856.0	8144.9
4.00	10540.3	7553.9	13742.1	7539.0
4.50	9597.1	6963.7	13632.1	7489.7
5.00	9075.7	6638.5	13489.2	7411.9
5.50	8756.9	6481.3	13343.6	7336.9
6.00	8375.5	6316.8	13102.9	7207.7
6.50	8050.8	6145.7	12836.6	7061.0
7.00	7616.5	6460.3	12539.9	6895.5
7.50	6973.8	5903.2	12126.7	6665.1
8.00	7093.6	5690.8	11756.4	6459.0
8.50	7105.0	5535.3	11390.8	6254.7
9.00	6896.8	5428.5	11005.6	6041.0
9.50	6453.3	5244.3	10589.0	5811.2
10.0	6068.8	4998.9	10162.7	5576.7
11.0	5543.0	4523.6	9373.6	5144.5
12.0	4984.3	3991.5	8572.2	4706.4
13.0	4505.5	3481.2	7592.7	4159.8
13.5	4308.2	3286.2	7254.1	3868.3
14.0	4130.7	3143.7	7069.4	3634.3
15.5	3483.0	2879.5	6172.5	2960.7

Table 3.5.3-6 (cont.)

Double-Ended Pump Suction Break
Blowdown Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
16.0	3244.1	2847.0	5802.9	2742.3
16.5	2955.0	2840.7	5382.6	2521.1
17.0	2435.4	2707.5	4617.4	2081.5
17.5	1964.7	2397.6	3983.0	1687.8
18.0	1598.3	1975.9	3410.5	1362.5
18.5	1319.7	1640.2	3020.0	1145.1
19.0	1093.7	1365.0	2709.2	982.7
19.5	870.4	1089.5	2797.1	954.2
20.0	682.7	856.5	3050.1	977.2
20.5	525.7	660.8	2420.2	754.5
21.5	233.1	294.0	724.0	215.2
22.0	100.8	127.6	.0	.0
22.5	.0	.0	.0	.0

* Mass and Energy exiting from the S/G side of the break

** Mass and Energy exiting from the pump side of the break

Table 3.5.3-7

Double-Ended Pump Suction Break with Diesel Failure
Reflood Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
22.5	.0	.0	.0	.0
24.0	.2	.2	.0	.0
24.3	5.4	6.4	.0	.0
24.6	20.8	24.5	.0	.0
25.4	47.0	55.5	.0	.0
26.6	72.6	85.7	.0	.0
27.6	89.8	106.0	.0	.0
30.6	129.0	152.4	.0	.0
31.6	139.7	165.0	.0	.0
32.6	151.0	178.4	1160.4	214.9
33.6	153.9	181.8	1858.3	347.7
34.6	153.5	181.3	1869.6	352.6
35.6	153.8	181.8	2212.4	381.9
37.6	152.4	180.1	2136.5	372.8
39.6	151.1	178.5	2062.6	363.8
41.6	149.9	177.1	1991.4	354.9
42.6	149.3	176.4	1956.9	350.6
44.6	148.2	175.0	1890.0	342.2
46.6	147.1	173.8	1825.9	334.1
48.6	146.1	172.6	1764.4	326.2
50.6	145.1	171.4	1705.2	318.5
52.6	144.2	170.3	1648.4	311.1
53.6	143.7	169.8	1620.7	307.5
55.6	142.9	168.8	1566.9	300.4
57.6	142.0	167.8	1515.0	293.5
59.6	141.3	166.9	1464.7	286.7
61.6	140.5	166.0	1416.1	280.1
65.6	139.0	164.2	1323.1	267.3
69.6	137.6	162.6	1235.2	255.0
73.6	136.3	161.0	1151.8	243.0
77.6	135.0	159.5	1072.2	231.3
78.6	134.2	158.6	781.4	187.6
80.6	135.1	159.6	754.0	184.4
81.7	135.5	160.1	739.3	182.7
85.6	136.6	161.4	689.2	176.9
89.6	137.2	162.1	640.6	171.3
91.6	136.0	160.7	247.5	150.2
93.6	134.0	158.3	245.2	147.4
101.6	126.4	149.3	236.2	136.9
102.1	125.9	148.7	235.7	136.2

Table 3.5.3-7 (cont.)

Double-Ended Pump Suction Break with Diesel Failure
Reflood Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
109.6	119.6	141.3	228.2	127.5
115.6	115.2	136.1	223.0	121.3
123.6	110.0	129.9	216.9	114.1
125.6	108.9	128.6	215.5	112.5
133.6	104.7	123.6	210.5	106.5
141.6	101.2	119.5	206.3	101.5
163.6	94.6	111.7	198.2	92.0
189.6	91.0	107.5	193.5	86.5
201.6	90.4	106.7	192.5	85.2
210.8	90.5	106.8	194.6	85.7

TABLE 3.5.3-8
DOUBLE-ENDED PUMP SUCTION BREAK WITH DIESEL FAILURE
PRINCIPLE PARAMETERS DURING REFLOOD

TIME SECONDS	FLOODING TEMP DEGREE F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (POUNDS MASS PER SECOND)	INJECTION ACCUMULATOR (POUNDS MASS PER SECOND)	INJECTION SPILL (POUNDS MASS PER SECOND)	ENTHALPY BTU/LBM
22.5	156.0	.000	.000	.00	.00	.333	.0	.0	.0	.00
23.3	155.5	16.138	.000	.52	.73	.000	2895.7	2895.7	.0	99.50
23.8	155.2	8.217	.000	1.08	.73	.000	2857.6	2857.6	.0	99.50
24.2	155.4	2.602	.035	1.23	1.31	.197	2827.9	2827.9	.0	99.50
24.5	155.6	3.115	.073	1.29	1.82	.303	2806.0	2806.0	.0	99.50
25.7	156.3	2.309	.285	1.50	3.98	.396	2713.2	2713.2	.0	99.50
26.6	156.8	2.227	.380	1.61	5.51	.409	2656.7	2656.7	.0	99.50
30.7	159.7	2.497	.588	2.00	12.64	.427	2412.1	2412.1	.0	99.50
32.6	161.4	2.650	.629	2.16	15.36	.432	2306.1	2306.1	.0	99.50
35.6	164.1	2.545	.659	2.39	15.57	.437	2547.8	2163.7	.0	95.51
37.2	165.5	2.494	.668	2.51	15.57	.437	2476.8	2092.7	.0	95.40
45.0	172.8	2.351	.690	3.01	15.57	.435	2174.7	1790.5	.0	94.82
53.5	181.0	2.264	.699	3.50	15.57	.433	1905.0	1520.7	.0	94.16
62.5	189.8	2.197	.704	4.00	15.57	.432	1663.7	1279.4	.0	93.38
72.6	199.6	2.136	.708	4.54	15.57	.431	1429.2	1044.8	.0	92.38
78.6	205.4	2.100	.709	4.85	15.57	.430	1023.3	638.8	.0	89.55
80.6	207.4	2.101	.710	4.95	15.57	.432	996.4	612.0	.0	89.29
81.7	208.5	2.101	.711	5.00	15.57	.433	981.9	597.6	.0	89.14
89.6	215.8	2.092	.714	5.40	15.57	.438	883.1	499.1	.0	87.99
91.6	217.5	2.079	.714	5.50	15.43	.437	384.2	.0	.0	73.03

TABLE 3.5.3-8 (cont.)
DOUBLE-ENDED PUMP SUCTION BREAK WITH DIESEL FAILURE
PRINCIPLE PARAMETERS DURING REFLOOD

TIME SECONDS	FLOODING TEMP DEGREE F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (POUNDS MASS PER SECOND)	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY BTU/LBM
93.6	219.2	2.050	.714	5.60	15.27	.436	384.2	.0	.0	73.03
102.1	225.9	1.934	.714	6.00	14.66	.435	384.3	.0	.0	73.03
113.6	233.8	1.801	.713	6.52	14.08	.433	384.3	.0	.0	73.03
125.3	240.7	1.691	.713	7.00	13.73	.431	384.4	.0	.0	73.03
139.6	247.9	1.587	.713	7.56	13.56	.430	384.4	.0	.0	73.03
151.4	253.1	1.522	.714	8.00	13.58	.429	384.4	.0	.0	73.03
165.6	258.6	1.466	.715	8.50	13.74	.428	384.4	.0	.0	73.03
180.2	263.6	1.427	.718	9.00	14.03	.428	384.5	.0	.0	73.03
195.6	268.4	1.400	.721	9.51	14.41	.428	384.5	.0	.0	73.03
210.8	272.5	1.388	.725	10.00	14.83	.429	384.5	.0	.0	73.03

Table 3.5.3-9

Double-Ended Pump Suction Break with Diesel Failure
Post-Reflood Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
210.9	100.7	127.5	283.9	95.7
225.9	99.9	126.5	284.7	95.4
230.9	100.6	127.4	283.9	95.0
260.9	98.9	125.3	285.6	94.4
265.9	99.6	126.2	284.9	94.0
290.9	98.2	124.4	286.3	93.5
295.9	98.9	125.3	285.6	93.2
325.9	97.2	123.2	287.3	92.5
330.9	97.9	124.0	286.6	92.2
355.9	96.5	122.2	288.0	91.7
360.9	97.1	123.0	287.4	91.3
385.9	95.7	121.2	288.8	90.8
390.9	96.3	122.0	288.2	90.4
420.9	95.0	120.3	289.6	91.9
425.9	95.7	121.2	288.8	91.5
455.9	94.5	119.6	290.1	90.7
460.9	95.2	120.5	289.4	90.3
490.9	93.9	118.9	290.6	89.4
495.9	94.6	119.8	289.9	89.0
525.9	93.3	118.2	291.2	88.2
530.9	94.0	119.0	290.6	87.8
555.9	92.9	117.6	291.7	89.1
560.9	93.5	118.4	291.0	88.7
585.9	92.3	117.0	292.2	87.9
590.9	93.0	117.7	291.6	87.5
615.9	91.9	116.4	292.6	86.7
645.9	92.3	116.9	292.2	87.2
670.9	91.2	115.5	293.3	86.3
695.9	91.7	116.1	292.9	86.8
715.9	90.7	114.9	293.8	86.0
740.9	91.0	115.3	293.5	86.5
810.9	89.5	113.3	295.1	84.5
825.9	90.0	114.0	294.5	85.1
850.9	89.2	112.9	295.3	83.6
865.9	89.5	113.4	295.0	84.2
915.9	88.4	112.0	296.1	83.8
925.9	88.8	112.5	295.7	82.9
1055.9	87.1	110.3	297.5	81.8
1060.9	51.0	64.6	333.6	92.2
1172.8	51.0	64.6	333.6	92.2
1172.9	59.5	72.7	325.0	89.0

Table 3.5.3-9 (cont.)

Double-Ended Pump Suction Break with Diesel Failure
Post-Reflood Mass and Energy Releases

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
1289.1	59.5	74.2	325.0	88.8
1289.2	57.5	66.2	327.0	29.3
1680.0	54.0	62.2	330.5	30.0
1680.1	54.0	62.2	26.5	7.8
3600.0	45.2	52.0	35.4	9.4
3600.1	32.0	36.8	48.6	3.6
3780.0	31.3	36.0	52.4	3.8
3780.1	34.3	39.5	49.4	8.3
10000.0	23.2	26.7	60.5	10.1
64800.0	14.1	16.2	69.6	11.6
64800.1	15.5	17.8	68.2	11.5
100000.0	13.6	15.7	70.1	11.8
1000000.0	5.8	6.7	77.9	13.1

TABLE 3.5.3-10
DOUBLE-ENDED PUMP SUCTION BREAK WITH DIESEL FAILURE
MASS BALANCE

	TIME SECONDS	.00	22.50	22.50	210.83	1172.93	1289.05	3600.00
		MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	579.16	579.16	579.16	579.16	579.16	579.16	579.16
ADDED MASS	PUMPED INJECTION	.00	.00	.00	67.57	437.50	482.15	1370.77
	TOTAL ADDED	.00	.00	.00	67.57	437.50	482.15	1370.77
*** TOTAL AVAILABLE ***		579.16	579.16	579.16	646.73	1016.66	1061.31	1949.93
DISTRIBUTION	REACTOR COOLANT	403.94	26.46	70.06	111.82	111.82	111.82	111.82
	ACCUMULATOR	175.22	145.27	101.66	.00	.00	.00	.00
	TOTAL CONTENTS	579.16	171.73	171.73	111.82	111.82	111.82	111.82
EFFLUENT	BREAK FLOW	.00	407.43	407.43	534.90	904.83	949.48	1838.10
	ECCS SPILL	.00	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	407.43	407.43	534.90	904.83	949.48	1838.10
*** TOTAL ACCOUNTABLE ***		579.16	579.15	579.15	646.72	1016.65	1061.30	1949.92

TABLE 3.5.3-11 DOUBLE-ENDED PUMP SUCTION BREAK WITH DIESEL FAILURE ENERGY BALANCE								
	TIME SECONDS	.00	22.50	22.50	210.83	1172.93	1289.05	3600.00
		ENERGY (MILLION BTU)						
INITIAL ENERGY	IN RCS, ACC, S GEN	624.22	624.22	624.22	624.22	624.22	624.22	624.22
ADDED ENERGY	PUMPED INJECTION	.00	.00	.00	4.93	31.95	35.21	100.10
	DECAY HEAT	.00	4.60	4.60	19.55	72.63	78.09	168.40
	HEAT FROM SECONDARY	.00	-5.17	-5.17	-5.17	-3.23	-3.22	-3.22
	TOTAL ADDED	.00	-.57	-.57	19.32	101.35	110.07	265.28
*** TOTAL AVAILABLE ***		624.22	623.66	623.66	643.54	725.57	734.29	889.50
DISTRIBUTION	REACTOR COOLANT	237.49	7.14	11.48	29.50	29.50	29.50	29.50
	ACCUMULATOR	17.43	14.45	10.12	.00	.00	.00	.00
	CORE STORED	23.83	14.14	14.14	4.03	3.87	3.82	2.68
	PRIMARY METAL	118.73	112.88	112.88	97.99	58.99	56.38	40.49
	SECONDARY METAL	58.66	58.24	58.24	54.45	32.97	31.04	22.54
	STEAM GENERATOR	168.07	166.30	166.30	153.12	89.28	84.17	61.01
	TOTAL CONTENTS	624.22	373.15	373.15	339.08	214.61	204.90	156.21
EFFLUENT	BREAK FLOW	.00	250.03	250.03	303.87	510.38	514.54	721.54
	ECCS SPILL	.00	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	250.03	250.03	303.87	510.38	514.54	721.54
*** TOTAL ACCOUNTABLE ***		624.22	623.18	623.18	642.95	724.98	719.44	877.75

3.5.4 Containment Response

3.5.4.1 Identification of Causes and Accident Description

The Turkey Point containment system is designed such that for all high-energy line break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure should remain below the design pressure with adequate margin. This section details the containment response subsequent to a hypothetical main steamline break (MSLB) (Section 3.5.1) or a loss-of-coolant accident (LOCA) (Section 3.5.3).

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a high-energy line break inside containment. The impact of MSLB or LOCA mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the uprated 2300 MWt core power conditions.

In addition to the containment peak pressure and temperature response, the thermal performance of the CCW System is also analyzed for a postulated RCS primary or secondary side rupture.

3.5.4.2 Input Parameters and Assumptions

An analysis of containment response to the rupture of the RCS or main steamline must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis.

Also, values for the initial temperature of the component cooling water (CCW) and temperature of the intake cooling water (ICW) and refueling water storage tank (RWST) solution are assumed, along with the initial water inventory of the RWST. All of these values are chosen conservatively for maximizing containment pressure, as shown in Table 3.5.4-1.

The following are the major assumptions made in the analysis.

(a) The mass and energy released to the containment are described in Sections 3.5.1 for MSLB and 3.5.3 for LOCA.

(b) Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

(c) Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

(d) For the steamline break analysis and the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

3.5.4.2.1 Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. For each node, a conservation of energy equation, expressed in finite-difference form, accounts for transient conduction into and out of the node and temperature rise of the node. Table 3.5.4-2 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 3.5.4-3.

The heat transfer coefficient to the containment structure is calculated based primarily on the work of Tagami (Reference 1). From this work, it was determined that the value of the heat transfer coefficient increases parabolically to a peak value. The value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air-weight ratio.

Tagami presents a plot of the maximum value of heat transfer coefficient, h, as a function of "coolant energy transfer speed," defined as follows:

$$h = \frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) (time interval to peak pressure)}}$$

From this, the maximum h of steel is calculated:

$$h_{max} = 75 \left(\frac{E}{t_p\ V} \right)^{0.60} \qquad (3.5.4\text{-}1)$$

where:

h_{max} = maximum value of h (Btu/hr ft^2-°F).

t_p = time from start-of-accident to end-of-blowdown for LOCA and steam line isolation for secondary breaks (sec).

V = containment free volume (ft^3).

E = coolant energy discharge (Btu).

The parabolic increase to the peak value is given by:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{0.5}, \; 0 \le t \le t_p \qquad (3.5.4\text{-}2)$$

where:

h_s = heat transfer coefficient for steel (Btu/hr-ft^2-°F).

t = time from start-of-accident (sec).

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel.

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})e^{-0.05(t-t_p)} \quad t > t_p \qquad (3.5.4\text{-}3)$$

where:

h_{stag} = $2 + 50X, 0 \le X \le 1.4$.

h_{stag} = h for stagnant conditions (Btu/hr-ft^2-°F).

X = steam-to-air weight ratio in containment.

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system or the main steam system, the containment safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low long-term pressure. Also, although the containment structure is not a very effective heat sink during the initial reactor coolant system blowdown, it still contributes significantly as a form of heat removal throughout the rest of the transient.

3.5.4.2.2. Active Heat Removal

During the injection phase of post-accident operation, the emergency core cooling systems deliver water from the refueling water storage tank and accumulators into the reactor vessel. Since this water enters the vessel at refueling water storage tank and accumulators ambient temperatures, which is less than the temperature of the water in the vessel, it can absorb heat from the core until saturation temperature is reached. During the recirculation phase of operation, water is taken from the containment sump and cooled in the residual heat removal heat exchanger. The cooled water is then pumped back to the reactor vessel to absorb more decay heat. The heat is removed from the residual heat exchanger by the CCW System. The RHR System and CCW System performance parameters are explained in Section 5.5.2.

Another containment heat removal system is the containment spray. Containment spray is used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the containment spray pumps also draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase of post-accident operation, water can be drawn from the residual heat removal heat exchanger outlet and sprayed into the containment atmosphere via the containment spray system. The spray flow rate modeled is shown in Table 3.5.4-4.

When a spray droplet enters the hot, saturated, steam-air containment environment following a loss-of-coolant accident, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling droplet are as follows.

$$\frac{d}{dt}(Mu) = mh_g + q \tag{3.5.4-4}$$

$$\frac{d}{dt}(M) = m \tag{3.5.4-5}$$

where

$$q = h_c A * (T_s - T),$$
$$m = k_g A * (P_s - P_v).$$

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, <u>Nu</u>, and the Nusselt number for mass transfer, <u>Nu'</u>.

Both <u>Nu</u> and <u>Nu'</u> may be calculated from the equations of Ranz and Marshall (Reference 2).

$$\underline{Nu} = 2 + 0.6\ (\underline{Re})^{1/2}(\underline{Pr})^{1/3} \quad (3.5.4\text{-}6)$$

$$\underline{Nu'} = 2 + 0.6\ (\underline{Re})^{1/2}(\underline{Sc})^{1/3} \quad (3.5.4\text{-}7)$$

Thus, Equations 3.5.4-4 and 3.5.4-5 can be integrated numerically to find the internal energy and mass of the droplet as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99 percent of the bulk containment ambient temperature in less than 2 seconds.

Droplets of this size will reach equilibrium temperature with the steam-air containment atmosphere after falling through less than half the available spray fall height.

Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly (Reference 3) show that droplets of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment.

These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere. Nomenclature used in this section is as follows.

<u>Nomenclature</u>

A	= area
h_c	= coefficient of heat transfer
k_g	= coefficient of mass transfer
h_g	= steam enthalpy
M	= droplet mass
m	= diffusion rate
<u>Nu</u>	= Nusselt number for heat transfer
<u>Nu'</u>	= Nusselt number for mass transfer
P_s	= steam partial pressure
P_v	= droplet vapor pressure
<u>Pr</u>	= Prandtl number
q	= heat flow rate
<u>Re</u>	= Reynolds number
<u>Sc</u>	= Schmidt number
T	= droplet temperature
T_s	= steam temperature
t	= time
u	= internal energy

The emergency containment coolers (ECCs) are a final means of heat removal. The ECCs consist of the fan and the banks of cooling coils. The fans draw the dense post-accident atmosphere through banks of finned cooling coils and mix the cooled steam/air mixture with the rest of the containment atmosphere. The coils are kept at a low temperature by a constant flow of component cooling water (CCW). Since this system does not use water from the RWST, the mode of operation remains the same both before and after the spray system and emergency core cooling system change to the recirculation mode. However, CCW is also used to cool the RHR heat exchanger(s) during recirculation. This will adversely affect ECC performance due to increased CCW temperatures and lower CCW flowrates to the ECCs. See Table 3.5.4-5 for ECC heat removal capability for the design basis containment integrity analyses. The ECC heat removal rates used for the CCW thermal performance analyses are explained in Section 5.5.2.

With these assumptions, the heat removal capability of the passive and active containment heat removal systems are sufficient to absorb the energy releases and still keep the maximum calculated pressure below the design pressure for the LOCA and MSLB containment integrity transients. The assumptions made for the CCW thermal performance analyses are more than adequate to demonstrate the heat removal capability of the CCW System.

3.5.4.3 Description of Analysis

Calculation of containment pressure and temperature, as well as the CCW System response is accomplished by use of the computer code COCO (Reference 4). For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with the appropriate boundary conditions.

3.5.4.3.1 MSLB Containment Integrity

The MSLB mass and energy releases that were performed for the 1.4 ft^2 DER at Hot Zero Power (HZP) as discussed in Section 3.5.1 were used to analyze the containment response. The failure of a MSCV was the limiting single failure for MSLB containment integrity. Since the failure was postulated to occur in the secondary steam system safety equipment, all of the containment heat removal equipment was assumed to be operational. This case was analyzed to the time of steam generator dryout. The sequence of events for this case is shown in Table 3.5.4-6.

3.5.4.3.2 LOCA Containment Integrity

A series of cases was performed for the LOCA containment integrity. Section 3.5.3 documented the M&E releases for the most-limiting single failure of a diesel generator for a DEPS break and the releases from the blowdown of a DEHL break. Each of these cases was performed at an initial containment pressure of +0.3 psig and +3.0 psig. These two pressures represent the nominal assumed and maximum operating pressures in the containment.

Two additional DEPS cases with a diesel failure were performed. These cases were performed with only 1 ECC actuating from the auto-start signal, a second ECC manually actuated at 24 hours after accident initiation, and continuous operation of the recirculation sprays upon actuation during the cold leg recirculation switchover sequence. This differs from the other DEPS cases such that each of those cases assumed that the recirculation sprays would be terminated no later than 18 hours after accident initiation.

The COCO calculations for all of the base DEPS cases were performed for 1 million seconds (approximately 11.6 days) and the additional cases were performed for greater than 31 days. The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Tables 3.5.4-7 through 3.5.4-9.

3.5.4.3.3 CCW Thermal Performance

A series of cases were performed that maximized the heat input to the CCW System and/or minimized the heat removed from the CCW System. This is a different approach than the containment integrity cases which minimize the heat input to the CCW System in order to maximize the containment pressure and temperature conditions. The intent of this portion of the analysis was to determine the impact of the thermal uprating on the inlet and outlet temperatures from the following components:

- CCW Heat Exchangers (CCW as well as ICW)
- ECCs (Emergency Containment Coolers)
- RHR Heat Exchangers (RHR as well as CCW)

As part of this analysis, the CCW, ICW and RHR flowrates and heat exchanger overall heat transfer coefficients based on fouling were modified throughout the series of runs to maximize the temperatures at the entrance or exit of a particular component. The ECC heat removal rates were also modified based on higher than ECC design CCW flowrates which maximized the heat input to the CCW System. For a description of the CCW, ICW and RHR input assumptions, see Section 5.5.2.

The series of CCW thermal performance cases was based on the same failure scenarios for the MSLB and LOCA mass and energy releases from Sections 3.5.1 and 3.5.3. The mass and energy releases for the MSLB cases were based on the MSCV failure. Mass and energy releases from the diesel failure, the spray pump failure and the "no failure" were used for the LOCA cases. As previously noted, the "no-failure" LOCA releases were based upon all of the ECCS pumps operating. Therefore, these releases could be used for cases that modeled a failure of an ICW pump (an ICW pump failure has no impact on the calculation of M&E releases).

The COCO models for the containment heat sinks and the containment spray system remained the same as for the containment integrity analyses. The performance of the ECCs was maximized with modified conservative assumptions (see Section 5.5.2) so that the ECCs would transfer a maximum amount of energy into the CCW System.

Since the mass and energy releases are calculated to maximize the containment pressure and temperature conditions for the design basis containment integrity analyses, these same releases provide a conservative steam temperature profile for use with the modified ECC performance. This combination of energy input to the containment and energy removal via the ECCs provided a maximum of energy transfer into the CCW System.

The amount of energy transferred out of the CCW System was minimized by conservative assumptions for the amount of CCW heat exchanger fouling and the ICW System flow rates (see Section 5.5.2).

The temperature of most interest was the peak CCW temperature at the outlet of the CCW heat exchanger (referred to as CCW supply temperature). Although, the entrance and exit conditions of the other CCW System and RHR System components, and the ECCs were also determined. For cases with 2 ECCs operating, the CCW supply temperature peaked within 10 minutes after switchover to cold leg recirculation. All cases resulted in CCW supply temperatures that were within acceptable limits. Section 5.5.2 of this report contains the overall conclusions of this analysis for all components considered.

3.5.4.4 Acceptance Criteria

The containment response for design-basis containment integrity is based on an ANS Condition IV event, an infrequent fault. The acceptance criteria for the containment response are:

- the peak calculated containment pressure should not exceed the containment design pressure of 55 psig;

- the calculated pressure at 24 hours should be 50% of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

3.5.4.5 Results

The results of the transient analysis of the containment at an initial pressure of +0.3 psig for the LOCA cases are shown in Figures 3.5.4-1 through 3.5.4-6. Figures 3.5.4-1 and 3.5.4-2 show the response to the DEPS case with 2 ECCs assumed to be operating initially. The containment response to the DEHL blowdown is presented in Figures 3.5.4-3 and 3.5.4-4. The results of the long term DEPS transient with only 1 ECC operating initially and a second ECC manually actuated at 24 hours are presented in Figures 3.5.4-5 and Figure 3.5.4-6. The containment pressure transient for the 1.4 ft^2 DER MSLB at 0% power with a MSCV failure is shown in Figure 3.5.4-7. All of these cases show that the containment pressure will remain below design pressure of 55 psig. In addition, all of the cases performed at the maximum initial containment pressure of +3.0 psig were also below the design pressure. After the peak pressure is attained, the operation of the safeguards system reduced the containment pressure. For the LOCA, at 24 hours following the accident, the containment pressure has been reduced to a value well below 50 percent of the peak calculated value. The containment integrity results are shown in Table 3.5.4-10 for LOCA and the MSLB ruptures.

The CCW thermal analysis considered several failure scenarios. Cases that modeled a single failure of a diesel generator, a containment spray pump, and an ICW pump were considered. In addition, several non-diesel scenarios were performed where all 3 ECCs would be actuated and/or RHR pumps were assumed to be in a runout condition. In this configuration, the CCW supply temperature was predicted to exceed the acceptable system temperatures. This prompted the need to limit the number of ECCs that would auto-start to two and the flow from the RHR pumps in the "piggy-back" mode. The results of these modifications are acceptable. When the same logic is used to limit the number of ECCs that auto-start to two for the MSLB transients, then the COCO-predicted CCWS temperatures show that large break LOCAs are more limiting than MSLB transients.

3.5.4.6 Conclusions

The containment integrity analyses have been performed for the thermal uprate program at Turkey Point Units 3 and 4. The analyses included both long-term MSLB and LOCA transients. As described in the results Section 3.5.4.5, all cases resulted in a peak containment pressure that was less than 55 psig. In addition, all long-term cases were well below 50% of the peak value within 24 hours. Based on these results, all applicable acceptance criteria from Section 3.5.4.4 have been met and Turkey Point Units 3 and 4 are safe to operate at 2300 MWt (core).

The CCW thermal performance analyses have also been performed for the thermal uprate program. This analysis also considered the LOCA and MSLB transients. As described in Section 3.5.4.5 and Section 5.5.2, all cases resulted in entrance and exit temperatures that were less than the design values. Based on these results for the CCW System analysis, all applicable criteria for the components have been met and Turkey Point Units 3 and 4 are safe to operate at 2300 MWt (core).

3.5.4.7 References

1. Takashi Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965", No. 1

2. Ranz, E. W. and Marshall, W. R. Jr., "Evaporation for Drops", Chemical Engineering Progress, 48, pp. 141-146, March 1952

3. Parsly, L. F., "Spray Tests at the Nuclear Safety Pilot Plant", Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1970, ORNL-4647, 1971, p. 82

4. "Containment Pressure Analysis Code (COCO)", WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), June 1974

Table 3.5.4-1

Containment Analysis Parameters

ICW temperature (°F)[Containment Integrity]	100
ICW temperature (°F)[CCW Thermal Performance]	95
Refueling water temperature (°F)	105
RWST minimum water deliverable volume (gal)	2.399×10^5
Initial containment temperature (°F)	130
Initial containment pressure (psia)	15.0
Initial relative humidity (%)	20
Net free volume (ft^3)	1.55×10^6
Emergency Containment Coolers	
Total	3
Analysis maximum	2
Analysis minimum	1
Setpoint (psig)	6.0
Delay time (sec)	
Without Offsite Power	50.0
With Offsite Power	35
Containment Spray Pumps	
Total	2
Analysis maximum	2
Analysis minimum	1
Setpoint (psig)	25.0
Delay time (sec)	
Without Offsite Power	60.0
With Offsite Power	45.0

Table 3.5.4-2

Containment Heat Sink Data

Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
1	360.9	Paint	0.000833
		Carbon Steel	0.617473
2	2725.6	Paint	0.000833
		Carbon Steel	0.232245
3	6368.1	Paint	0.000833
		Carbon Steel	0.109355
4	5426.0	Paint	0.000833
		Carbon Steel	0.066368
5	17366.0	Paint	0.000833
		Carbon Steel	0.038986
6	137461.3	Paint	0.000833
		Carbon Steel	0.021498
7	84988.4	Paint	0.000833
		Carbon Steel	0.011212
8	105344.0	Paint	0.000833
		Carbon Steel	0.005121
9	89906.9	Paint	0.000833
		Carbon Steel	0.001918
10	1378.0	Stainless Steel	0.08398
11	2335.8	Stainless Steel	0.043972
12	2684.9	Stainless Steel	0.015155
13	27329.0	Stainless Steel	0.002537
14	1207.0	Stainless Steel	0.0091
15	2150.0	Aluminum	0.020833

Table 3.5.4-2 (cont.)

Containment Heat Sink Data

Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
16	106200.1	Aluminum	0.000603
17	50132.0	Paint	0.00325
		Concrete	1.5
18	67240.0	Paint	0.000833
		Carbon Steel Liner	0.020833
		Concrete	1.5
19	775.0	Stainless Steel Liner	0.01
		Concrete	1.5
20	5825.0	Stainless Steel Liner	0.005417
		Concrete	1.5

Table 3.5.4-3

Thermal Properties of Containment Heat Sinks

Material	Thermal Conductivity (Btu/hr-°F-ft)	Volumetric Heat Capacity (Btu/ft^3-°F)
Paint	0.138	11.105
Carbon Steel	28.88	54.66
Stainless Steel	14.48	57.37
Aluminum	91.25	38.59
Concrete	1.048	26.27

Table 3.5.4-4

Containment Spray Pump Flow

Containment Pressure (psig)	1 Pump (gpm)	2 Pumps (gpm)
0.0	1548.0	3009.0
10.0	1509.0	2947.0
20.0	1469.0	2870.0
30.0	1429.0	2789.0
40.0	1386.0	2704.0
50.0	1340.0	2611.0

Table 3.5.4-5

Emergency Containment Cooler Performance
Containment Integrity Analyses
(Btu/sec/ECC)
(Based on 2000 gpm CCW Flow/ECC and 25,000 CFM Steam-Air Flow)

Containment Temperature (°F)

CCW Temp. (°F)	120.	140.	160.	180.	200.	220.	240.	260.	280.	300.
95.	319.7	898.	1726.	2852.	4504.	6652.	9599.	13505.	18320.	25209.
110.	222.4	806.	1635.	2780.	4406.	6550.	9485.	13294.	18164.	24973.
120.	0.0	589.	1421.	2585.	4181.	6311.	9168.	12921.	17900.	24450.
130.	0.0	325.	1162.	2302.	3917.	6030.	8860.	12577.	17253.	23705.
135.	0.0	170.	1012.	2171.	3767.	5871.	8704.	12368.	17036.	23402.
140.	0.0	0.0	848.	2016.	3603.	5702.	8518.	12196.	16797.	23107.
145.	0.0	0.0	664.	1840.	3422.	5516.	8251.	11865.	16541.	22740.
150.	0.0	0.0	464.	1649.	3230.	5310.	7954.	11618.	16082.	22357.
170.	0.0	0.0	0.0	636.4	2188.4	4227.2	6762.	10291.	14652.	20622.
210.	0.0	0.0	0.0	0.0	0.0	1022.3	3373.6	6597.6	10588.	16012.

Table 3.5.4-6

1.4 ft^2 MSLB Hot Zero Power with MSCV Failure
Sequence of Events

Time (sec)	Event Description
0.0	Main Steamline Break Occurs
1.4	HI-1 Containment Pressure Setpoint Reached
3.4	Rod Motion Occurs (HI-1 actuates SI which actuates Reactor Trip)
9.9	High Steam Flow Coincident with Low T_{avg} SI Signal (539°F)
14.4	Safety Injection Initiated (actuated on HI-1) Feedwater Isolation (actuated on HI-1)
14.5	HI-2 Containment Pressure Setpoint Reached
16.9	Steamline Isolation Occurs via a High Steam Flow Coincident with Low T_{avg} SI Signal
36.1	Emergency Containment Coolers (2) Actuate
76.1	Containment Sprays (2 trains) Actuate
238.3	Peak Containment Pressure (48.1 psig) and Temperature (269.4°F) Occur
606.0	Mass and Energy Releases Terminate (SG Dryout)

Table 3.5.4-7

Double-Ended Pump Suction Break @ +0.3 psig with Diesel Failure
Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
0.8	Containment HI-1 Pressure Setpoint Reached
4.0	Low Pressurizer Pressure SI Setpoint = 1745.0 psia Reached
5.0	Containment HI-2 Pressure Setpoint Reached
12.7	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injecting Water
19.7	Peak Pressure and Temperature Occur
22.5	End of Blowdown Phase
50.8	Emergency Containment Coolers (2) Actuate
65.0	Containment Spray Suction from RWST Begins (1 train)
77.8	Broken Loop Accumulator Water Injection Ends
89.9	Intact Loop Accumulator Water Injection Ends
210.8	End of Reflood for MIN SI Case
1680.0	RWST Low Level Reached - Recirc Sequence Begins
3780.0	RWST Low-Low Level Reached - Cold Leg Recirc Begins Containment Spray (RWST) Ends
3780.1	Containment Spray (SUMP) Begins
64,800.	Switchover to Hot Leg Recirculation Begins Containment Spray (SUMP) Ends
1.0E+06	Transient Modeling Terminated

Table 3.5.4-8

Double-Ended Pump Suction Break @ +0.3 psig with Diesel Failure (Only 1 ECC)

SEQUENCE OF EVENTS

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
0.8	Containment HI-1 Pressure Setpoint Reached
4.0	Low Pressurizer Pressure SI Setpoint = 1745.0 psia Reached
5.0	Containment HI-2 Pressure Setpoint Reached
12.7	Broken Loop Accumulator Begins Injecting Water
13.0	Intact Loop Accumulator Begins Injecting Water
19.7	Blowdown Peak Pressure and Temperature Occur
22.5	End of Blowdown Phase
50.8	Emergency Containment Coolers (1) Actuate
65.0	Containment Spray Suction from RWST Begins (1 train)
77.8	Broken Loop Accumulator Water Injection Ends
89.9	Intact Loop Accumulator Water Injection Ends
210.8	End of Reflood for MIN SI Case
1059.5	Overall Peak Pressure and Temperature Occur
1680.0	RWST Low Level Reached - Recirc Sequence Begins
3780.0	RWST Low-Low Level Reached - Cold Leg Recirc Begins Containment Spray (RWST) Ends
3780.1	Containment Spray (SUMP) Begins
86,400.	Second ECC Manually Actuated
1.0E+06	Transient Modeling Terminated

Table 3.5.4-9

Double-Ended Hot Leg Break
Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.3	Low Pressurizer Pressure SI Setpoint = 1745.0 psia reached
10.9	Broken Loop Accumulator Begins Injecting Water
11.1	Intact Loop Accumulator Begins Injecting Water
18.7	Peak Pressure and Temperature Occur
21.5	End of Blowdown Phase
50.0	Transient Modeling Terminated

Table 3.5.4-10

Containment Integrity Results

LOCA
(Loss of Offsite Power Assumed)

FAILURE SCENARIO	PINIT (psig)	PEAK PRESS (psig)	TIME OF PEAK PRESS (sec)	PEAK TEMP (°F)	TIME OF PEAK TEMP (sec)	PRESS @ 24 HRS (psig)
DEPS w/Diesel, 2 ECCs & Recirc Spray Off @ 18 hrs	0.3	45.8	19.7	270.8	19.7	11.5
DEPS w/Diesel, 1 ECC, 2nd ECC @ 24 hrs & Cont'd Recirc Spray	0.3	46.2	1059.5	271.1	1059.5	7.6
DEHL	0.3	48.1	18.7	273.9	18.7	-

MSLB
(Offsite Power Available)

FAILURE SCENARIO	PINIT (psig)	PEAK PRESS (psig)	TIME OF PEAK PRESS (sec)	PEAK TEMP (°F)	TIME OF PEAK TEMP (sec)
1.4 ft^2 DER HZP	3.0	48.1	238.3	269.4	238.3

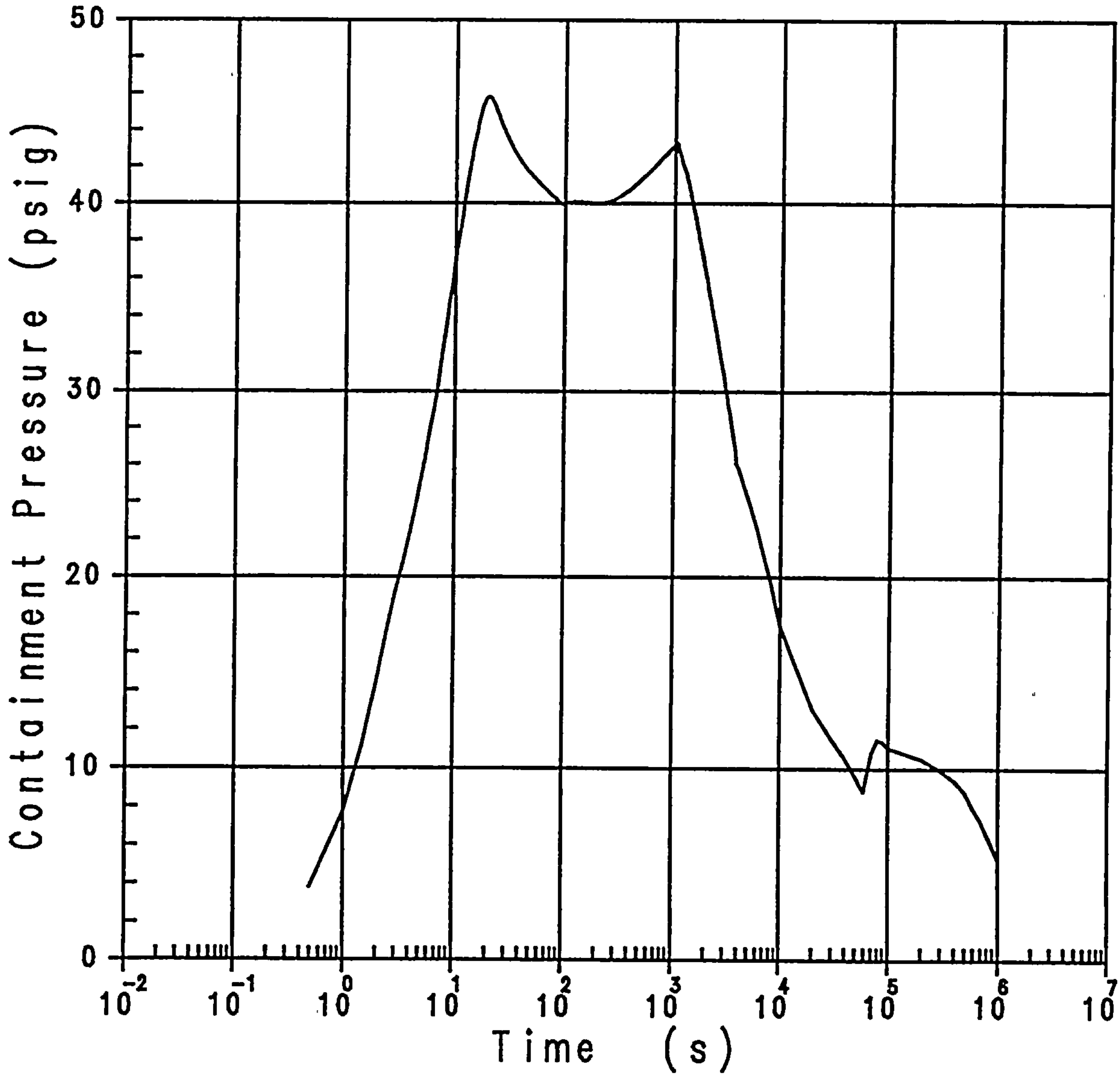

Figure 3.5.4-1: DEPS: Diesel Failure Case with 1 CSS and 2 ECCs at PINIT = 0.3 psig Containment Pressure

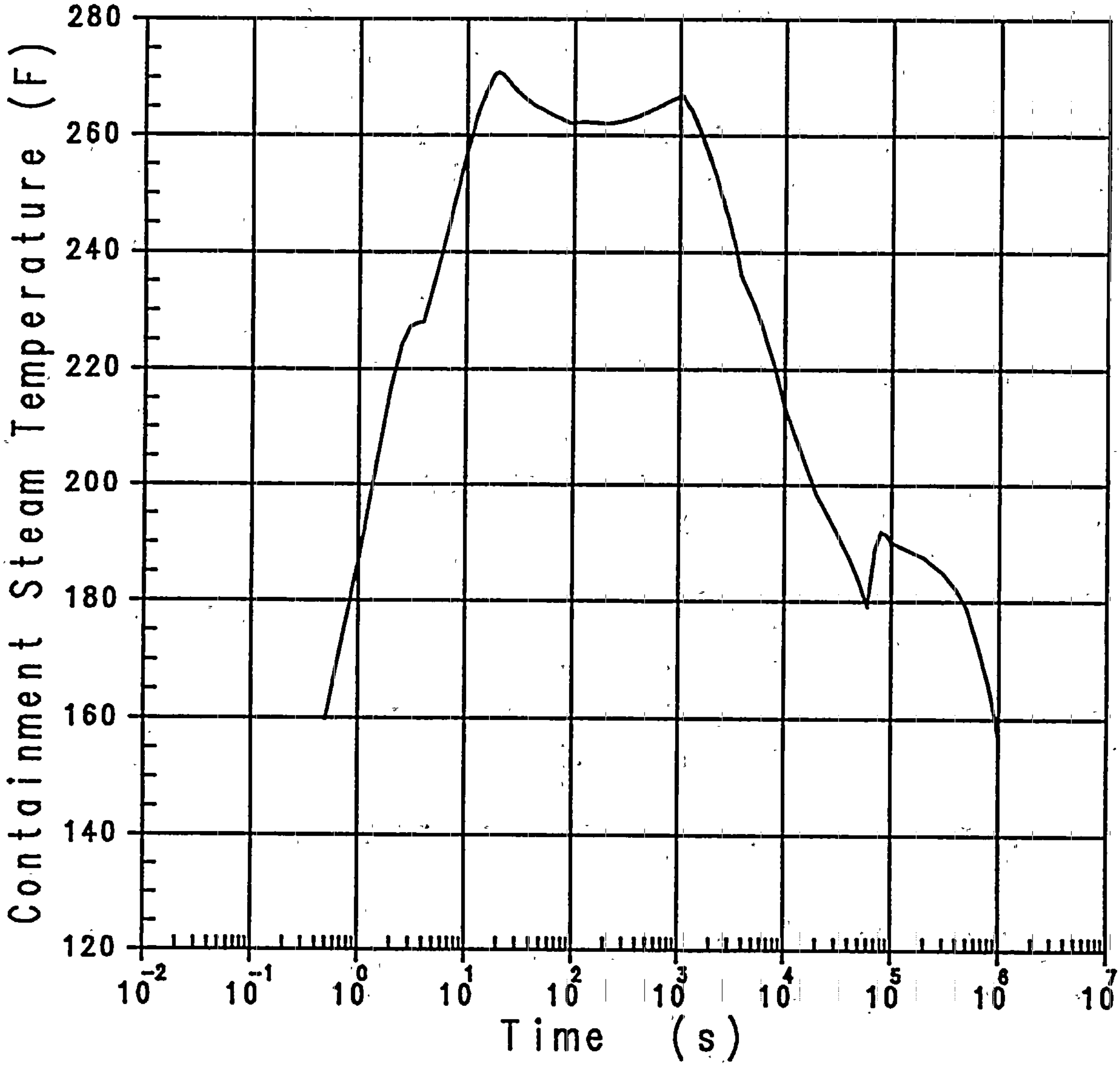

Figure 3.5.4-2: DEPS: Diesel Failure Case with 1CSS and 2ECC at PINIT = 0.3 psig Containment Steam Temperature

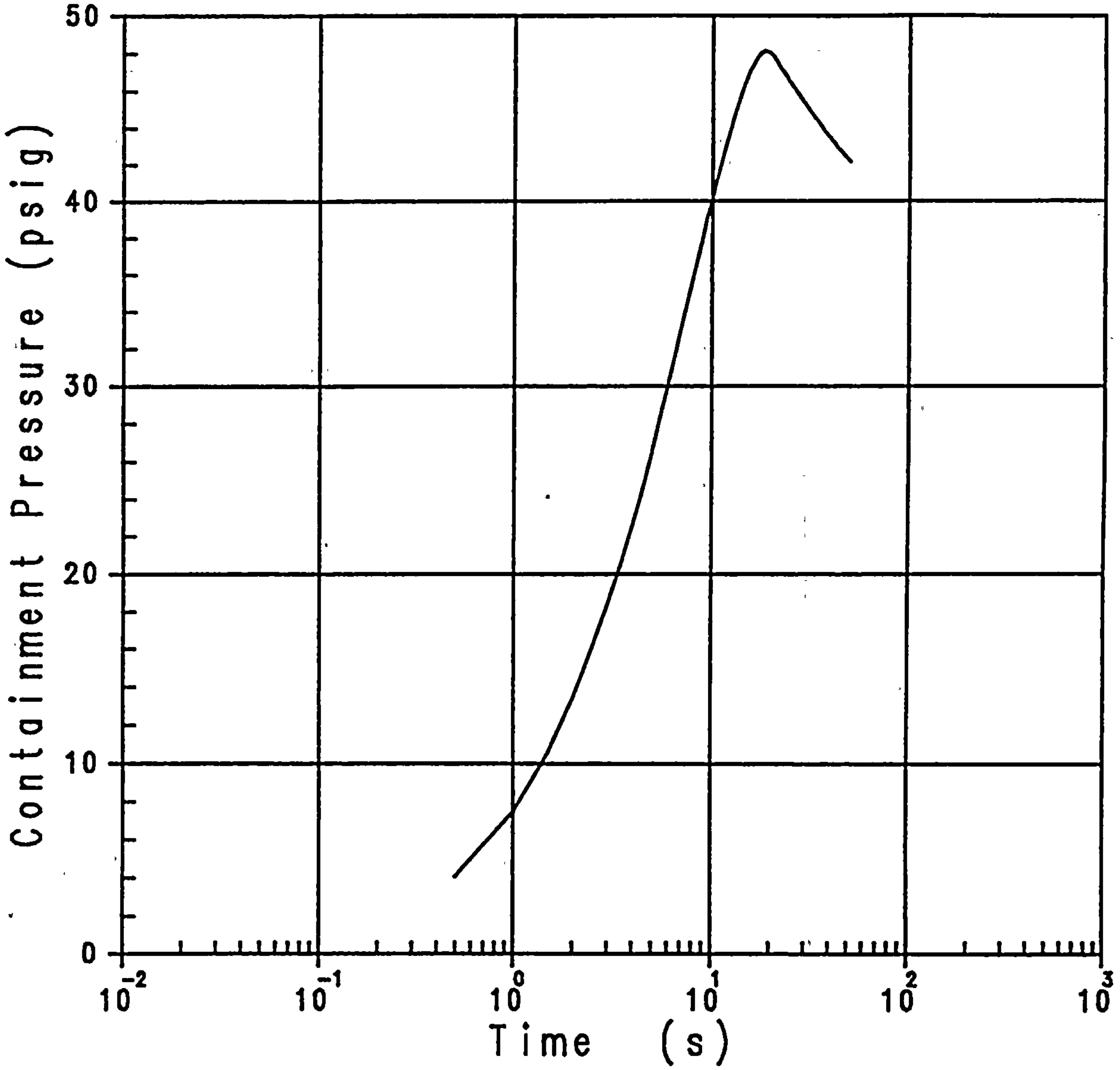

Figure 3.5.4-3: DEHL: Case with PINIT = 0.3 psig
Containment Pressure

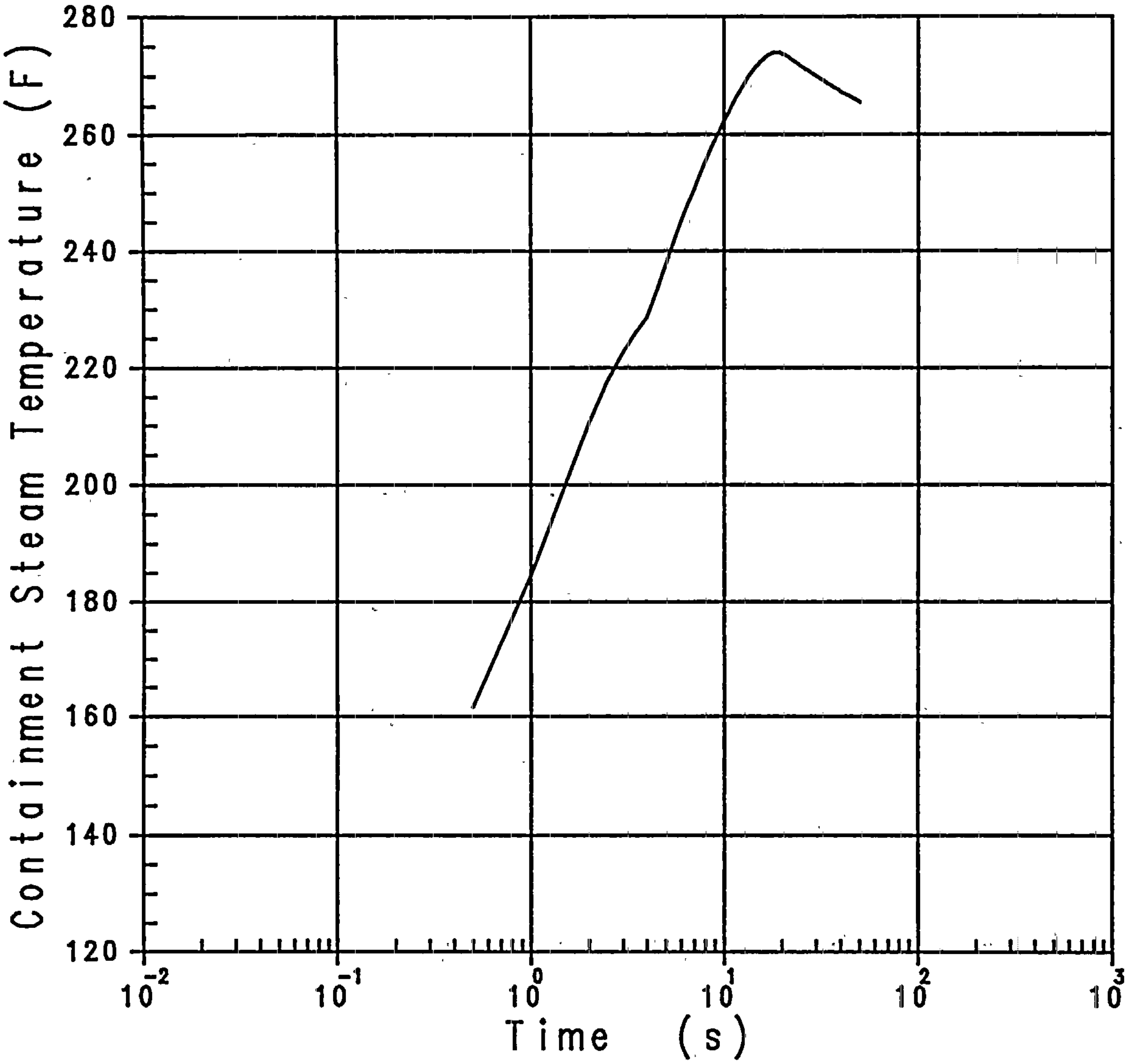

Figure 3.5.4-4: DEHL: Case with PINIT = 0.3 psig
Containment Temperature

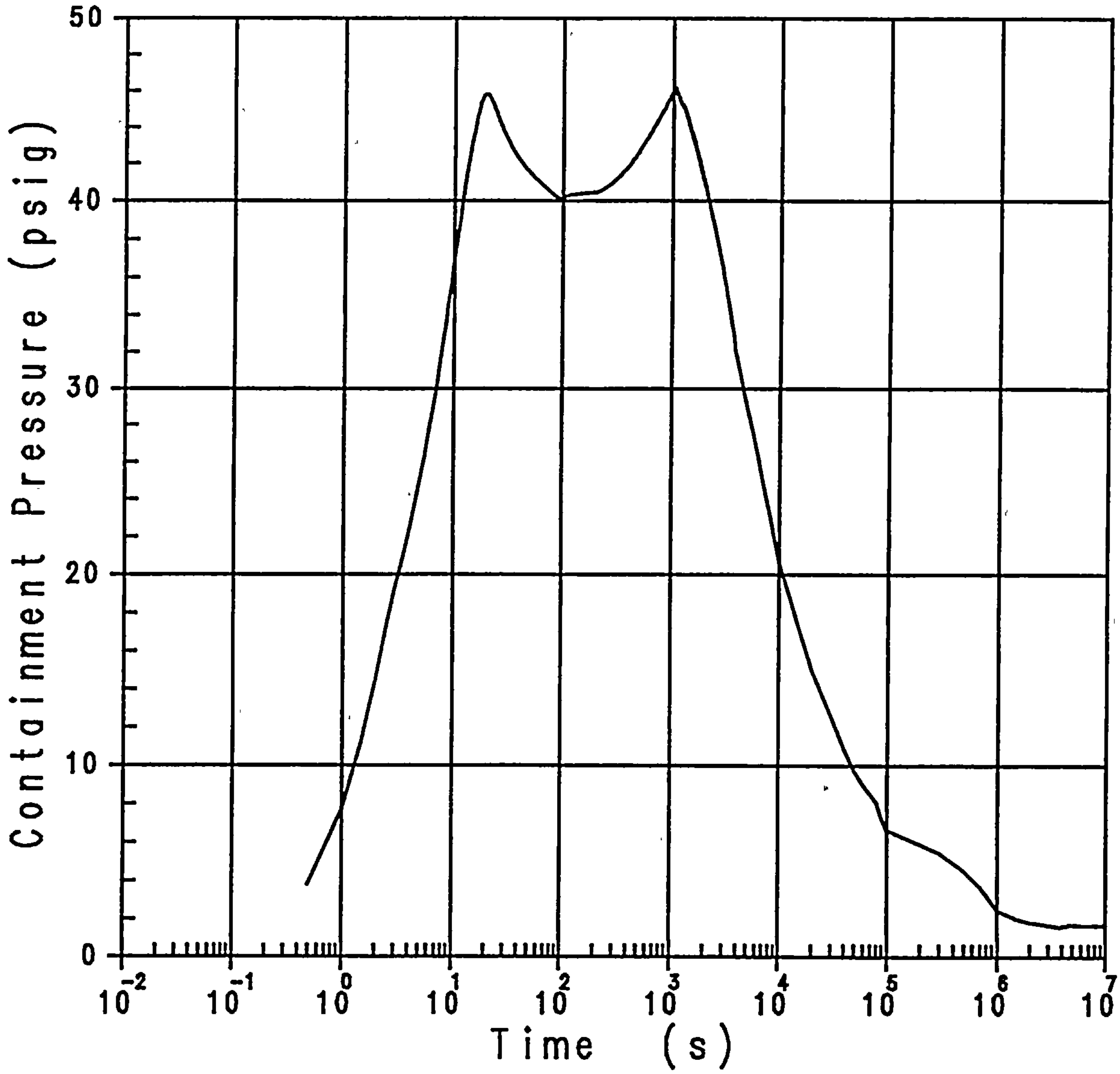

Figure 3.5.4-5: DEPS: Diesel Failure with 1 CSS and 1 ECC at PINIT = 0.3 psig Containment Pressure

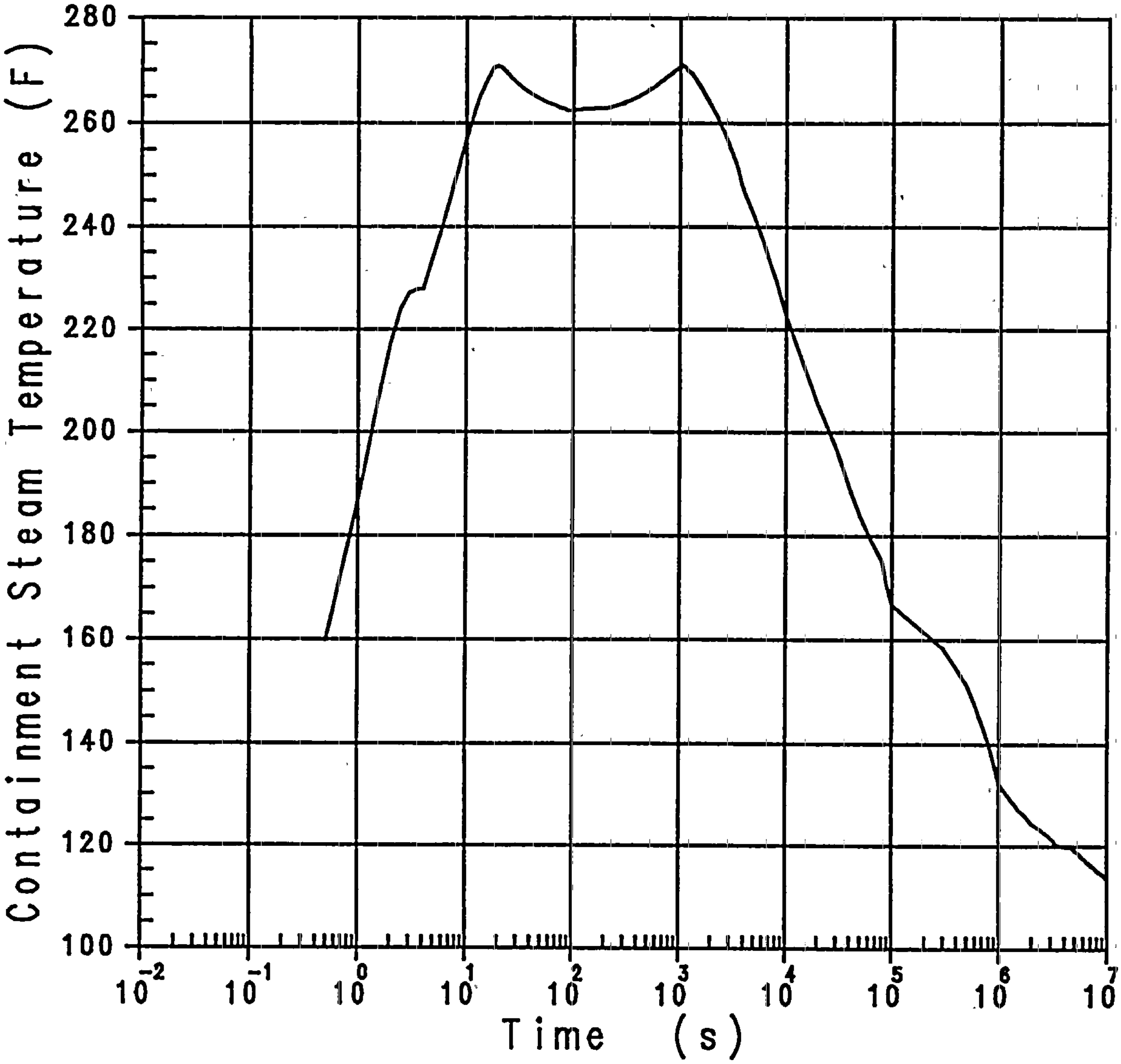

Figure 3.5.4-6: DEPS: Diesel Failure with 1 CSS and 1 ECC at PINIT = 0.3 psig Containment Steam Temperature

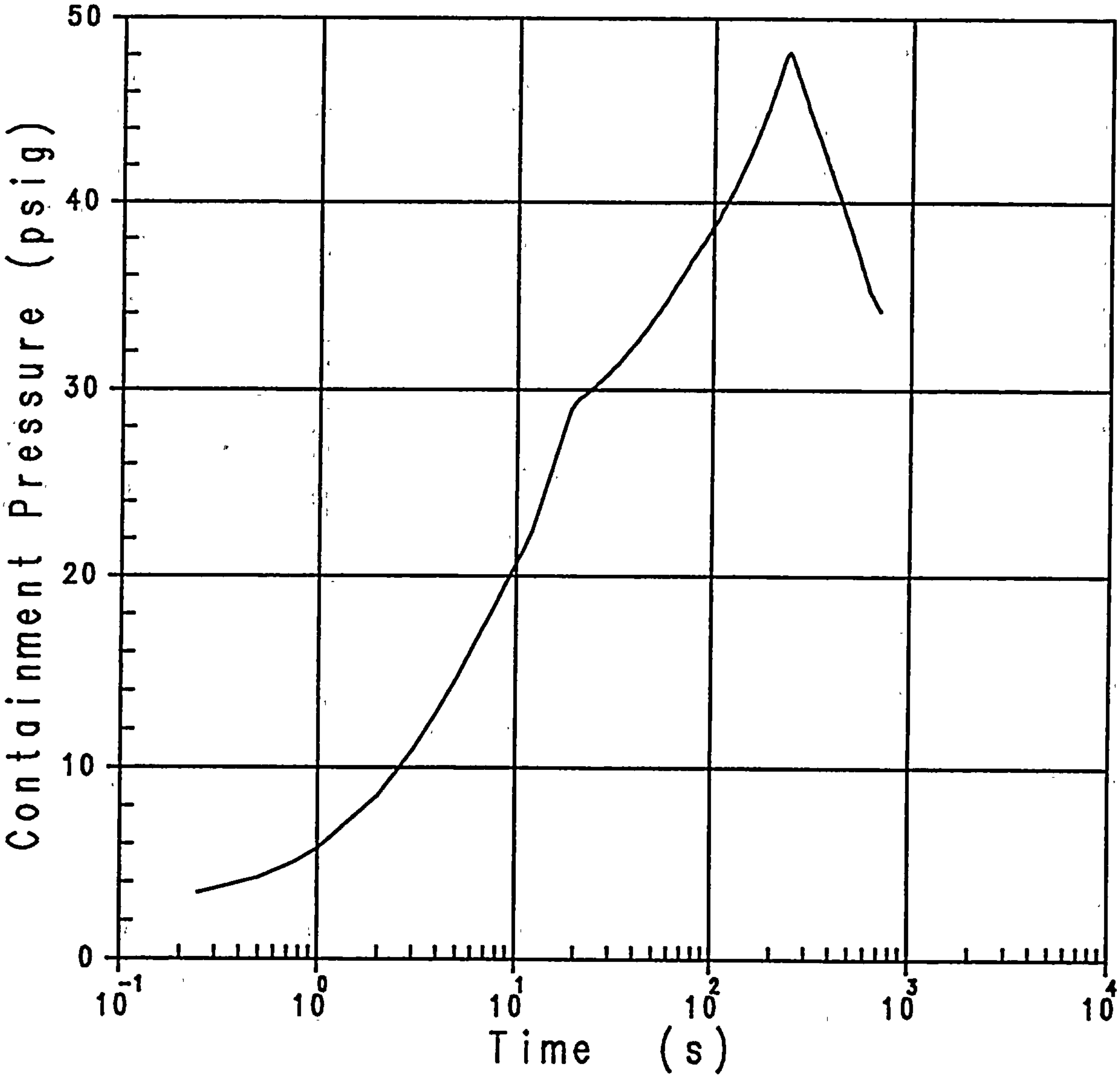

Figure 3.5.4-7: 1.4 ft² HZP Steamline Break, MSCV Failure, 2 ECCs and CSSs Containment Pressure

3.6 ADDITIONAL DESIGN BASIS AND PROGRAMMATIC EVALUATIONS

3.6.1 Equipment Qualification Events

The revised containment accident analysis temperature and pressure are within the existing EQ profiles, except for the long term temperature at 31 days. The EQ profile is based on the containment temperature returning to 120°F after 31 days following a LOCA. The uprate Containment Integrity Analysis (Refer to Section 3.5) results in an increase of 2.4°F at 31 days. This is within the normal range for containment temperature (104°F - 130°F). Therefore, the accident duration of 31 days is still acceptable and uprate will not have an adverse impact on the EQ program.

3.6.2 Hydrogen Generation Rates

An analysis of containment post-LOCA hydrogen generation rates was performed for the Turkey Point uprating program. The hydrogen generation analysis was based on an uprated total core thermal power of 2346 MWt (102% of 2300 MWt core power). The Westinghouse analysis demonstrates that with no recombiner in service, the hydrogen concentration in containment will not exceed four volume percent for 17 days following a LOCA. Placing a hydrogen recombiner in service prior to the 18th day following a LOCA will maintain containment hydrogen levels below the lower flammability limit of four percent.

3.6.3 Plant Programs

Evaluations of the following generic issues/programs were performed to determine the impact of thermal power uprate to a core power of 2300 MWt.

- Appendix R
- Station Blackout
- Erosion/Corrosion
- Check Valve Program
- NRC Generic Letter 89-10 — "Safety-Related Motor-Operated Valve Testing and Surveillance"
- NRC Generic Letter 89-13 — "Service Water System Problems Affecting Safety-Related Equipment"
- NRC Generic Letter 88-20 — "Individual Plant Examination (IPE)"
- ATWS

The evaluation of plant compliance with Appendix R consists of determining the impact of uprate on the equipment and systems required to provide the safe shutdown functions. In addition, the existing Appendix R analysis is reviewed to identify any issues that would be impacted by plant uprate. Changes in system/component design and operating conditions are reviewed to determine if there is adverse impact on post-fire safe shutdown.

The evaluation of the Erosion/Corrosion (E/C) and the Check Valve programs consists of determining the revised program parameters as a result of the uprate. The revised parameters are identified and addressed in the applicable BOP Engineering Report sections and these sections are reviewed to determine the impacts on the programs are adequately addressed.

The evaluation of Generic Letters 89-10 and 89-13 consists of identifying the applicable system/components and the design basis parameters used in the program inspections. The impact of the uprating on the design basis parameters are reviewed to determine if the parameters used in the inspections are changed.

The evaluation of the Equipment Qualification (EQ) consists of reviewing the revised containment accident analysis temperature and pressure profiles against the EQ program pressure/temperature profiles to determine that the existing EQ profiles are bounding. Where the EQ profiles are not bounded the impact of the conditions outside the bounding conditions are reviewed to demonstrate the new condition will not impact equipment qualification based on the existing EQ profiles. Radiological EQ review is addressed in Section 3.6.1.

The Uprating Program does not have an adverse impact on the Turkey Point Units 3 and 4 generic issues and programs as discussed in the followings paragraphs.

Appendix R:

The evaluations of the systems impacted by the uprate did not identify changes to design or operating condition that will adversely impact the ability to provide post-fire safe shutdown in accordance with Appendix R. The most noticeable change was in the inventory of the Condensate Storage Tank (CST) and Demineralized Water Storage Tank (DWST) minimum required volumes. The required volumes were increased resulting in an increased minimum Technical Specification volume. The revised minimum volume will provide additional available inventory to satisfy the design basis requirement for post-fire safe shutdown and does not adversely impact post-fire safe shutdown.

Station Blackout (SBO):

The evaluations of the systems impacted by the uprate did not identify changes to design or operating conditions that will adversely impact the ability to provide safe shutdown for SBO. The most noticeable change was in the inventory of the Condensate Storage Tank (CST) minimum required volume. The required volume was increased resulting in an increased minimum Technical Specification volume. The revised minimum volume ensures the CST design basis has sufficient inventory to maintain the plant at hot standby for 15 hours followed by a four-hour cooldown to RHR cut-in. This provides adequate inventory available for safe shutdown during an SBO event.

Erosion/Corrosion (E/C):

The impact of increased operating velocities in the secondary system susceptible to E/C are evaluated as part of the system/component evaluations. The increase in velocity will have an impact on the E/C rates, however the impact is not expected to increase the rates beyond design limits and the existing program will continue to ensure the effects of wall thinning are monitored and evaluated.

Check Valve Program:

The evaluations of the systems impacted by the uprate did not identify changes to design or operating conditions that will adversely impact the Check Valve program. The velocities will increase in the secondary plant system but will not have an adverse impact on the operation of the check valves in these systems.

Generic Letter (GL) 89-10: "Safety-Related Motor-Operated Valve Testing and Surveillance"

The impact of increased operating parameters on the design basis differential pressures used in the GL 89-10 Program were evaluated. The design basis differential pressures were conservatively based on pump shutoff head, relief and safety valve setpoints (plus accumulation), and interlock setpoints which are not changed as a result of the uprate. Therefore, the uprate does not impact the Generic Letter 89-10 Program.

Generic Letter 89-13: "Service Water System Problems Affecting Safety-Related Equipment"

The impact of revised heat exchanger parameters used in the CCW thermal analysis were evaluated for their impact on Generic Letter 89-13. The CCW analysis assumed higher tube fouling factors in order to reduce the frequency of maintenance of the CCW heat exchangers. The revised fouling and associated CCW and ICW flow rates are to be included in the Generic Letter 89-13 program for monitoring the system and heat exchanger performance.

Generic Letter 88-20: "Individual Plant Examination (IPE)"

A review of plant uprating was performed for its impact on the Individual Plant Examination (IPE) performed for Turkey Point in response to Generic Letter 88-20. The impact of uprating, changes to plant procedures that would be required, and plant modifications associated with uprating were considered. Because the uprating is limited to 4.5% and has very minimal impact on plant configuration, no change to core damage frequency (CDF) was calculated.

Anticipated Transients Without Scram - ATWS

The Final ATWS Rule, 10 CFR 50.62, as applicable to Westinghouse designed PWRs, requires the installation of a system diverse from the reactor protection system that helps mitigate the adverse consequences of an ATWS event by initiating a turbine trip and actuating the auxiliary feedwater system. To comply with this rule, AMSAC (ATWS Mitigation System Actuation Circuitry) systems have been installed and are operational in both Turkey Point Units 3 and 4. Supporting the basis of the Final ATWS Rule are ATWS analyses performed by Westinghouse (Reference 1). In these analyses, a 3-Loop PWR with an NSSS power of 2785 MWt was considered. This power level is significantly higher and bounds the proposed Turkey Point uprated power condition of 2308 MWt. Hence, the proposed Turkey Point uprated power condition remains within the bounds of the basis of the ATWS analysis.

Reference

1. Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC), "ATWS Submittal," NS-TMA-2182, December 30, 1979, .

3.7 CONCLUSIONS OF ACCIDENT ANALYSES

All of the UFSAR Chapter 14 accident analyses applicable to Turkey Point Units 3 and 4 were re-analyzed or evaluated to support plant operation at the uprated conditions. All acceptance criteria continue to be met.

3.8 SUMMARY OF UFSAR ASSESSMENT

Paragraph (e) of 10 CFR 50.71 provides the requirement to periodically update the contents of the UFSAR originally submitted as part of application for the operating license. This is to maintain information in the FSAR as the latest material developed. The information in the update is to include the effects of changes made to the facility or procedures as described in the FSAR. In compliance with this regulation, revised sections of the Turkey Point UFSAR have been generated as appropriate which reflect the analyses and evaluations that take into account operation at the uprated conditions. These revisions will be incorporated into the Turkey Point Units 3 and 4 UFSAR on a schedule consistent with the FSAR update program already established.

CHAPTER 4

NSSS AND TURBINE GENERATOR (TG) COMPONENTS REVIEW

4.0 NSSS AND TURBINE GENERATOR (TG) COMPONENTS REVIEW

4.1 INTRODUCTION

The following information addresses the evaluation of the NSSS and TG components to support operation of Turkey Point Units 3 and 4 at the uprated power, within the bounds of the parameters defined in Table 2.1-1. The components evaluated for the uprating are as follows:

- Reactor Vessel
- Reactor Internals
- Reactor Coolant Pumps
- Control Rod Drive Mechanisms
- Reactor Coolant Piping and Supports
- Pressurizer
- Steam Generators
- Fuel
- Auxiliary Systems Components
- Turbine Generator Components

The primary components of the NSSS were designed and fabricated to the then applicable codes (B31.1 or ASME III, as listed in Table 4.1-9 of the Turkey Point 3 and 4 UFSAR, Rev. 12). Like most PWR plants as originally licensed, Turkey Point's NSSS and TG components and systems were generally designed for the capacity to operate at the "stretch" rating of 2308 MWt NSSS. However, to support this program, it was necessary to perform specific evaluations or analyses (e.g., stress analyses) at the uprated conditions in order to clearly utilize the existing plant margin for the uprated power and associated parameters.

The analyses and evaluations performed for the NSSS and TG components to support the uprating considered the original codes and standards, where those were applied. The ASME Boiler and Pressure Vessel Code, which was the design code for the majority of the RCS components, provides criteria and requirements for the evaluation of stress levels in pressure boundary components for design, normal operating, and accident conditions. The margin of safety provided by use of the design pressure as a basis for pressure limits is provided by the inherent safety factors in the criteria and requirements of the ASME Code.

The nature of the analyses and evaluations performed for the NSSS and TG components is found in detail in the sections below. However, in general, the efforts focused on structural evaluation, based on revised design performance capability parameters (from Section 2.0 of this report) and on revised NSSS design transients.

In addition, Appendix A to 10 CFR 50, "Fracture Prevention of Reactor Coolant Pressure Boundary (RCPB)", requires in part that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and accident conditions, (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. PWRs evaluate reactor vessel embrittlement in accordance with the criteria in RG 1.99 Revision 2, and 10 CFR 50.61, the Pressurized Thermal Shock (PTS) rule. The PTS rule requires that the PTS submittal be updated whenever there are changes in core loadings, surveillance measurements, or other information that indicates a significant change in projected values. A re-evaluation of the susceptibility of the reactor vessel to PTS was performed, due to the effects on neutron fluences and transient loadings. These effects result partly from the revised vessel average temperature range, but primarily from the higher power level assumed in the evaluations. The results of this evaluation are presented in Section 4.3.

4.2 NSSS DESIGN TRANSIENTS

The NSSS design transients were reviewed and revised as necessary to incorporate the uprating parameters, as reflected in Table 2.1-1. These were provided to the component designers for their use in structural evaluations and/or analyses to support the uprating. The component analysts used the most limiting NSSS design transient(s) for each component.

4.3 REACTOR VESSEL

4.3.1 Reactor Vessel Integrity

4.3.1.1 Introduction

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or pressure/temperature transients. The changes in neutron fluence resulting from the proposed Turkey Point Units 3 and 4 Uprating Program have been evaluated to determine the impact on reactor vessel integrity. This assessment included a review of the current integrated material surveillance capsule withdrawal schedule, applicability of the plant heatup and cooldown pressure-temperature limit curves currently contained in the Technical Specifications, and a revision to the RT_{PTS} values used in the submittal to the NRC for meeting the requirements of 10 CFR 50.61, known as the Pressurized Thermal Shock (PTS) Rule. The most critical area, in terms of reactor vessel integrity, is the beltline region of the reactor vessel.

4.3.1.2 Input Parameters and Assumptions

Material data was obtained for the Turkey Point reactor vessels from FPL's latest PTS submittal. Fluence projections on the vessel were calculated for the uprated power level for input to the reactor vessel integrity calculations. These fluence values were used to calculate the end-of-life transition temperature shift (EOL ΔRT_{NDT}) for development of the integrated surveillance capsule withdrawal schedule, adjusted reference temperature (ART) values for determining the applicability of the heatup and cooldown curves, and RT_{PTS} values.

4.3.1.3 Descriptions of Analyses/Evaluations

The reactor vessel integrity evaluation for the Turkey Point uprating included the following objectives:

1. Review the integrated reactor vessel surveillance capsule schedule to determine if changes are required as a result of changes in vessel fluence due to the uprating.

2. Calculate adjusted reference temperature (ART) values, following the methods of Regulatory Guide 1.99, Revision 2, for all beltline material based upon fluence values projected for the uprated condition to determine the applicability of the heatup and cooldown curves presently contained in the Turkey Point Technical Specifications.

3. Calculate RT_{PTS} values per the PTS Rule for all beltline material in the Turkey Point reactor vessels based upon fluence values projected for the uprated condition at the time of uprating and EOL.

4.3.1.4 Acceptance Criteria for Analyses/Evaluations

With respect to the analysis objectives stated in 4.3.1.3, the following are the criteria for each area:

1. Surveillance Capsule Removal Schedule: The proposed integrated surveillance capsule removal schedule developed for Turkey Point following the uprating shall meet the intent of ASTM E185-82.

2. Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves: The applicability date to which the heatup and cooldown curves presently contained in the Turkey Point Technical Specifications shall be determined.

3. Pressurized Thermal Shock (PTS): The uprated RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the PTS Rule.

4.3.1.5 Results

An evaluation of the impact of uprating on reactor vessel integrity was performed for the neutron fluence changes and other relevant system parameters associated with the uprating.

A review of the applicability of the current Technical Specification heatup and cooldown curves was completed and ART values were calculated for all beltline material using the material properties and uprated fluence projections. It was determined that the Turkey Point Unit 3 heatup and cooldown curves will be applicable to 19.0 EFPY after the uprating is implemented. The applicability date of the Turkey Point Unit 4 curves will be 19.7 EFPY after the uprating is implemented.

Calculations were performed for the uprating using the latest procedures specified by the NRC in the PTS Rule. All RT_{PTS} values remain below the NRC screening criteria values using the projected fluence values through 28.9 EFPY for Turkey Point Unit 3. For Turkey Point Unit 4, all RT_{PTS} values remain below the NRC screening criteria using the projected fluence values through 28.7 EFPY. These values represent end of operating license for Turkey Point Units 3 and 4.

4.3.1.6 Conclusions

It is concluded that the uprating program for Turkey Point Units 3 and 4 will not have significant impact on the reactor vessel integrity.

4.3.1.7 References

1. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", May 15, 1991.

2. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)"

3. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", January 1, 1990 Edition.

4. Regulatory Guide 1.99, Revision 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials"

4.3.2 Structural Evaluation

4.3.2.1 Introduction

Evaluations were performed for the various regions of the Turkey Point Units 3 and 4 reactor vessels to determine the stress and fatigue usage effects of NSSS operation at the revised operating conditions of the uprating program throughout the current plant operating licenses.

4.3.2.2 Input Parameters and Description of Evaluation Performed

The evaluations assessed the effects of the revised design transients and operating parameters on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress report and addenda. The evaluations consider a worst case set of operating parameters and design transients from among the high temperature uprating conditions, the low temperature uprating conditions and the original design basis.

In addition, reactor vessel operation from plant startup until implementation of the uprating and any future operation in accordance with the original design basis is still fully covered by the stress and fatigue analyses in the reactor vessel stress report. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated for the uprating program. In other cases the original design basis stress analysis remains conservative so that no new calculations were necessary, and the maximum ranges of stress intensity and fatigue usage factors reported in the stress report and the addenda continue to govern.

In addition to the revised operating parameters and design transients for the uprating program, a new set of LOCA loads at the reactor vessel/reactor internals interfaces was identified. The revised interface loads were evaluated by comparing them with the corresponding Faulted Condition reactor vessel/reactor internals interface loadings which were justified for application to the Turkey Point Units 3 and 4 reactor vessels.

4.3.2.3 Acceptance Criteria and Results of Evaluations

The uprating does not affect the maximum ranges of stress intensity reported in the Turkey Point Units 3 and 4 reactor vessel stress report. The evaluations show that for all of the limiting locations, the existing design stress analyses remain conservative when the revised operating parameters and design transients are incorporated. The maximum cumulative fatigue usage factors at all of the limiting locations increase somewhat, except those in the CRDM housing, vessel shell, core support pads, vent nozzle and bottom mounted instrument tubes which remain unchanged. However, the increases that occur are generally minimal, and all of the cumulative fatigue usage factors remain under the 1.0 limit with significant margin.

The evaluation of the Turkey Point Units 3 and 4 reactor vessels show they are acceptable for plant operation in accordance with the uprating program. Therefore, the reactor vessel uprating evaluation, in conjunction with the reactor vessel stress report, addresses reactor operation within the expanded operating temperature ranges as indicated above. Such operation is shown to be acceptable in accordance with the 1965 Edition of Section III of the ASME Boiler and Pressure Code with Addenda through the Summer 1966 for the remainder of the plant licenses.

4.3.2.4 Conclusions

Based on the analysis results discussed in the preceding section, the reactor vessel uprating evaluation demonstrates that the uprating does not affect any of the maximum ranges of stress intensity reported in the reactor vessel stress reports for Turkey Point Units 3 and 4. In addition, the maximum cumulative fatigue usage factors are affected minimally by the revised uprating conditions and continue to remain significantly below the acceptance criterion of 1.00.

4.3.2.5 References

1. Westinghouse Equipment Specification G-676244, Rev. 0 and Addendum Equipment Specification, "Three Loop - 155-1/2 Inch I.D. Reactor Vessel," dated 1/28/66.

4.4 REACTOR INTERNALS

Since the operating conditions for the Turkey Point Units 3 and 4 Uprating Program differ from the original design operating conditions, the reactor pressure vessel system and the reactor internal components were thoroughly addressed in order to assure compatibility and structural integrity of the components. In addition, thermal/hydraulic analyses are required to verify that existing core bypass flow limits are not exceeded and to develop pressure drops and upper head temperatures for input to Appendix K (ECCS), non-LOCA accident analyses, and NSSS performance evaluations. The subject areas most likely to be affected by changes in system operating conditions are:

1) Reactor internals system thermal/hydraulic performance,
2) Rod control cluster assembly (RCCA) scram performance, and
3) Reactor internals system structural response and integrity.

The effects on the pressure vessel/reactor internals system at Turkey Point Units 3 and 4 due to the Uprating Program are addressed below.

4.4.1 Thermal/Hydraulic System Evaluations

4.4.1.1 System Pressure Losses

An evaluation has been performed which determined the pressure distributions and flow characteristics within the reactor vessel, reactor internals, and reactor core for the uprating program conditions as specified in References 1 and 2. The total coolant pressure drops across the reactor internals increased by 8%. This data was utilized in the structural evaluation of the reactor internal components and as input into several analyses (i.e. LOCA).

4.4.1.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Analyses were performed to estimate core bypass flow values to either ensure that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow. The present Turkey Point design core bypass flow limit is 6.0% of the total reactor vessel flow. The increase in design core bypass flow from 4.5% to 6.0% is due primarily to the thimble plug elimination which was implemented in 1988. The total core bypass flow values were determined to be 5.19% and 5.54% for Turkey Point Units 3 and 4, respectively. Therefore, the design core bypass flow value of 6.0% of the total vessel flow can be maintained for the uprating.

4.4.1.3 Hydraulic Lift Forces

An evaluation was performed to determine hydraulic lift forces on the various reactor internals components to ensure that the reactor internals assembly would remain seated and stable for all conditions. The results of the calculations show that, with the uprated RCS conditions, there is a sufficient net clamping force between the reactor vessel head flange and upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals to ensure that the Turkey Point reactor internals assembly will remain seated and stable.

4.4.2 RCCA Scram Performance Evaluation

The rod drop time-to-dashpot entry (from gripper release of the drive rod) must be determined to be less than 2.4 seconds so that existing accident analyses remain valid. Evaluations were performed that determined that the RCCA drop time for the uprated conditions are bounded by the current limit of 2.4 seconds. In addition, the current normalized RCCA position versus time curve also remains bounding.

4.4.3 Flow Induced Vibration/Structural Integrity

The primary cause of lower internals' excitations is the flow turbulence generated by the expansion and turning of the flow at the transition from the inlet nozzle to the barrel-vessel annulus and the wall turbulence generated in the downcomer. Evaluations were performed which determined that there is a negligible impact on the core barrel response due to the RCS changes due to the uprating program.

The significant flow-induced forces on the upper internals are due to random turbulences generated by the cross flows which converge on the outlet nozzles. Evaluations were performed which determined that there is approximately 1.9% increase in the flow-induced vibration loads on the guide tubes and support columns due to the RCS changes due to the uprating program. Previous flow induced vibration analysis on the guide tube and the upper support column show that there exist sufficient margins to accommodate this increase in the flow induced vibration loads.

Stresses and fatigue usage factors for the limiting internal components of the upper and lower internals were evaluated for the changes in RCS conditions due to the uprating program and are within acceptable limits.

In summary, the reactor internals components at Turkey Point Units 3 and 4 remain in compliance with the current design requirements for operation at the uprated power conditions.

4.5 REACTOR COOLANT PUMPS

4.5.1 Introduction

The Turkey Point Units 3 and 4 have Model 93 reactor coolant pumps (RCPs) which were built prior to ASME Code Stamping requirements. The uprating parameters were evaluated for their effect on the RCP structural integrity and the RCP motor performance.

4.5.2 Reactor Coolant Pump Evaluation

For the uprating program, the RCPs were evaluated for any temperature increases or pressure increases that exceeded the original Equipment Specification (E-Spec.). The "50% step load decrease" transient was found to increase the ΔP above 2250 psia from the E-Spec. value of 120 psi to 128.7 psi (max). The resultant pressure is less than the design pressure and so the increase is considered insignificant.

The "Loss Of Flow" transient for the uprated condition produces a temperature and a pressure increase as compared to the original E-Spec. These changes are considered minor and less than original design values and other evaluated transients.

The "Feedwater Cycling" transient was not listed in the original E-Spec. No pressure increase for the RCPs is postulated and only a small temperature increase is postulated. The temperature cycling does not meet the ASME definition for a significant temperature difference fluctuation. Thus, the Feedwater Cycling transient has no effect on the fatigue integrity of the RCP.

4.5.3 RCP Motor Evaluation

The motor is required to drive the pump continuously under hot loop conditions without exceeding a specified stator winding temperature rise that is consistent with National Electrical Manufacturers Association (NEMA) Class B requirements. Motor testing has shown that the actual temperature rise at rated hot loop load (6000 HP) is well within the specification. Therefore, adequate margin exists for continuous operation with any load less than 6000 HP. For the uprated conditions, the worst case hot loop load is 5635 HP which is therefore acceptable.

The motor is required to drive the pump for up to 50 hours (continuous) under cold loop conditions without exceeding a specified stator winding temperature rise that is consistent with NEMA Class F requirements. Motor testing has shown that the actual temperature rise at the rated cold loop load (7500 HP) is well within the specification for the RCP. Therefore, adequate margin exists for continuous operation with any load less than 7500 HP. For the uprated conditions, the worse case cold loop load is 7155 HP which is therefore acceptable.

The motor must be capable of accelerating its worse case load without damage when 80% rated voltage at the rated frequency is applied. The limiting component for this type of starting duty is the rotor cage winding. For the uprated conditions, the calculated temperature rise for the critical motor components show that the allowable temperature limits are not exceeded.

Performance of the thrust bearings in the motor could be adversely affected by excessive or inadequate loading. The axial down thrust is increased for the uprated condition which results in a 1.2% reduction in bearing loading. This change has been reviewed and determined to be insignificant.

4.5.4 Conclusions

The uprating parameters are acceptable to the Model 93 RCPs including the spare with respect to structural integrity and motor performance.

4.5.5 References

1. Westinghouse Equipment Specification 676335, Rev. 1, "Florida Power and Light - Controlled Leakage Pump," WPAD, 10-9-67.

2. WAED Equipment Specification E-565604, Rev. C.

4.6 CONTROL ROD DRIVE MECHANISMS

4.6.1 Introduction

This section addresses the acceptability of the Model L106B Control Rod Drive Mechanisms (CRDMs), both full length and part length, for the uprating parameters. The part length (P/L) CRDMs are not used in the operation of the plant, however the P/L housings are primary pressure boundary components and so were evaluated with respect to the uprated conditions.

4.6.2 Input Parameters and Description of Evaluations Performed

The CRDMs were evaluated using the uprating design performance capability parameters. The applicable ASME Section III Code stress analyses were reviewed for the full length (F/L) and part length (P/L) CRDMs. The higher temperatures of the uprating are still bounded by the stress analysis.

The original equipment stress report evaluates the part length CRDMs for the general loadings provided by Westinghouse. In the report, it is stated that the thrust bearing retainer assembly, located in the lower portion of the CRDM adapter, is designed to act as a thermal barrier between the reactor vessel and the CRDM proper. Therefore, the uprating transients will not affect the part-length mechanism at elevations above the thrust bearing retainer assembly.

The geometry of the P/L CRDM lower joint is nearly identical to the geometry of the F/L CRDM lower joint. The canopy length on the F/L CRDM latch housing however, is much shorter than the canopy on the P/L CRDM adapter. Hence, the F/L CRDM canopy will be more rigid than the P/L CRDM canopy. Since a major portion of thermal induced stress in the canopy is caused by differential expansion between the two connecting components the thermal induced stresses in the canopy will be smaller for the P/L CRDM lower joint. Therefore, it may be concluded that the stress analysis of the full-length CRDM lower joint may also be used as a basis to justify the part-length CRDM lower joint.

The transients for the Turkey Point Uprating were compared to the Turkey Point Equipment Specification values of Reference 1. The Uprating Transients are bounded by the original transients except for a) the large step load decrease which now has a higher maximum pressure of 2379 psia, and b) feedwater cycling.

4.6.3 Acceptance Criteria and Results of Evaluations

For the two cases not bounded by the original analysis, the fatigue waiver criteria of the ASME Code, NB-3222-4(d) will be used. From the Code NB-3222-4(d) fatigue waiver, a significant pressure fluctuation is one which exceeds a pressure difference of 1282 psi. A significant temperature difference fluctuation is a change of 51.6°F.

The changes in these two transients are not significant changes; the pressure change is only 23 psi for the new large step load decrease and the feedwater cycling has a temperature swing of only 32°F.

4.6.4 Conclusions

The transient pressure/temperature changes associated with the uprating conditions do not qualify as significant fluctuations to be included in a code fatigue waiver and hence any fatigue usage increase is insignificant.

Thus, it is concluded that the Turkey Point Uprating/SGTP transients are acceptable for the F/L and P/L CRDM's.

4.6.5 Reference

1. CRDM Equipment Specification 676426, Rev. 1, WAPD, 11-3-67, and Interim Change No. 1, dated 12-10-76.

4.7 REACTOR COOLANT PIPING AND SUPPORTS

4.7.1 Introduction

Evaluations were performed on the potential impact the uprating program could have on the following components: Reactor Coolant Loop (RCL) piping, primary equipment nozzles, primary equipment supports, Reactor Vessel Head Vent System (RVHVS) piping, and the Pressurizer surge line.

4.7.2 Input Assumptions

The evaluation utilized the same analyses and methods and criteria used in the existing design basis for Turkey Point continue to be used.

The uprated parameters that define the various temperature conditions associated with the potential full power operating conditions of the plant were defined in Section 2. All of the thermal expansion, seismic, and LOCA analyses performed on the piping systems are performed at full power conditions. The system thermal design transients are used only in the pressurizer surge line thermal stratification analysis (in which a formal fatigue analysis is performed). The primary loop piping was designed and analyzed to the ASA B31.1 Power Piping Code which did not require a formal fatigue analysis.

The loop LOCA analysis considers forces associated with defined postulated breaks and reactor vessel dynamic LOCA displacements associated with defined postulated break cases. The design basis for the Turkey Point RCL piping LOCA analysis has changed as the result of the licensing of loop Leak-Before-Break (LBB) methodology, which eliminates the consideration of dynamic effect due to large break LOCA. Postulated guillotine breaks in the primary loop piping have been replaced with postulated guillotine breaks at the loop branch connections for the largest class 1 auxiliary lines (pressurizer surge line on the hot leg and accumulator line on the cold leg).

Because the seismic response spectra have been upgraded since the existing design basis loop analysis (NRC Bulletin 79-07 vintage evaluation) therefore, new seismic analyses were run incorporating the more recent spectra.

Two earlier programs were used as sources of information and models for this uprating work. The reactor coolant loop model used in the structural analysis for uprating was taken from the work performed to respond to the NRC Bulletin 79-07. The primary equipment support stiffnesses used in the analysis were upgraded from the original values to those used in the A-2 program which investigated the asymmetric LOCA loads on operating plants.

4.7.3 Description of Evaluations

Computer structural analyses were performed on the RCL piping system model for the loading conditions of deadweight, thermal, and seismic. The thermal expansion analysis was run to give the range of loadings associated with the temperature conditions defined.

The seismic analysis was run to include the newer seismic response spectra provided by FPL. The model used for the thermal analysis was also used to run the deadweight analysis to have a consistent set of results. The seismic model merely modified the supports on the deadweight model to account for lateral loadings. All three analysis types used the primary equipment support stiffnesses updated for the A-2 asymmetric LOCA loads evaluation.

The deadweight, thermal (low temperature and high temperature cases), and seismic analysis results for this RCL model were used as input to the specific evaluations for the loop piping, the primary equipment nozzles, the primary equipment supports, and the loop LBB.

As discussed above, a LOCA loop analysis was not necessary because the increase in margins after implementing loop LBB was more than enough to balance off any potential increases in LOCA loadings associated with the uprated conditions. Any existing design basis LOCA loadings continue to envelope the proposed uprated condition LOCA loadings.

The evaluation for the primary equipment nozzles involved a comparison of the newly generated loads for the deadweight, thermal, and seismic loading conditions with the allowable nozzle loadings for that equipment.

The primary equipment supports were not a Westinghouse design and the design basis calculations were not available. The analysis/evaluation for the supports consisted of comparing the loads on the various support components to the capacities for those same components. The basis for many of these support calculations goes back to the A-2 asymmetric LOCA loads evaluation.

The RVHVS piping was evaluated by comparing the new temperatures and pressures associated with the uprating program with those used in the existing head vent analysis. These new temperatures and pressures associated with uprating are enveloped by the parameters used in the piping analysis.

The evaluation performed on the pressurizer surge line stratification analysis included a review of the fatigue analysis and the stratification loadings that were transmitted to the pressurizer nozzle from the surge line piping. The changes and the percent increases for the uprated thermal design transients were tabulated and the impact on the fatigue usage factor was calculated. The new uprated conditions were reviewed to determine if the old envelope loads on the nozzle changed significantly. Temperature differences between the hot leg and pressurizer were used to calculate stratified moments in the surge line piping.

4.7.4 Acceptance Criteria and Results

The acceptance criteria for the loop piping stress evaluation is contained in the B31.1 Power Piping Code. The applicable load combinations of deadweight, pressure, seismic and thermal loads were checked against the appropriate allowable for the loop piping material. The pipe stress conditions were met.

The primary equipment nozzle loads were compared to the equipment specification allowables for the specific loading conditions analyzed. All of the nozzle loads met the allowables.

The primary equipment support loads were compared to the various support capacities. All support components assessed met the allowables.

Since the parameters of interest (temperatures and pressures) in the RVHVS piping analysis enveloped the uprating parameters, there was no impact on this piping analysis due to the uprating program.

The results of the evaluation for the pressurizer surge line stratification showed that the uprating conditions changed the fatigue usage factor at the location of highest usage factor from 0.942 to 0.944. The allowable usage factor is 1.0 and the change calculated was not significant. The calculated change in loadings on the pressurizer nozzle due to stratification for the uprated conditions was less than 4%. The change in nozzle loadings was considered insignificant because the original loadings on the pressurizer nozzle were conservative envelopes that lumped various transients under a small number of bounding thermal cases.

4.7.5 Conclusions

The parameters associated with the uprating program for Turkey Point have been evaluated for impact on the RCL piping, the primary equipment nozzles, the primary equipment supports, the RVHVS piping, and the pressurizer surge line. The evaluation indicates that all components met appropriate allowables. The evaluation for the stated components concluded that the plant uprating program had no adverse effect on the ability of these components to operate until the scheduled end of plant operation.

4.7.6 Reference

1. Turkey Point Units 3 and 4, "Approval of Leak-Before-Break (LBB) Methodology for Reactor Coolant System Piping", June 23, 1995.

4.8 PRESSURIZER

4.8.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and pressure and to keep the reactor coolant system (RCS) at the desired pressure.

The components in the lower end of the pressurizer (surge nozzle, lower head/heater well and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (spray nozzle, safety and relief nozzle, upper head/upper shell, manway and instrument nozzle) are affected by pressure, sprays through the spray nozzle, and steam temperature differences.

The pressurizer temperature is kept at the water saturation temperature (T-sat) corresponding to the desired pressure. The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (T-hot) and cold leg (T-cold) temperatures are low. This maximizes the ΔT experienced by the pressurizer because of the comparison to T-sat. Due to flow in and out of the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T-sat) and the RCS hot leg (T-hot). If the RCS pressure is high, with a correspondingly high T-sat, and T-hot is low, then the surge nozzle will experience the maximum thermal stress. Likewise the spray nozzle and upper shell temperatures alternate between steam at T-sat and spray, which for many transients is at T-cold. Thus, if RCS pressure is high, with a correspondingly high T-sat and T-cold is low, then the spray nozzle and upper shell will experience the maximum thermal stresses.

4.8.2 Input of Assumptions and Description of Evaluation

For the uprating, the transient conditions differ from the conditions to which the Turkey Point Units 3 and 4 pressurizers were originally designed and analyzed. To conservatively maximize thermal stresses the lowest T_{hot} and the lowest T_{cold} conditions were evaluated, regardless of which parameter set they came from.

The analysis was performed by modifying the original Turkey Point Units 3 and 4 pressurizer stress report, which was performed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition, Summer of 1965 Addendum. Analytical models of various sections of the pressurizer were subjected to pressure loads, external loads (such as piping loads), and thermal transients.

The maximum pressure and maximum external loads on the pressurizer are not affected by the thermal uprating conditions. Thus, the primary stresses calculated for the original analysis are still valid. The conditions that affect maximum primary plus secondary stresses do not change as a result of the thermal uprating, except for the surge nozzle. For all the components, the fatigue analysis is affected.

The original Turkey Point Units 3 and 4 pressurizer analysis was previously modified to account for normal transients and the surge nozzle analysis was previously updated for the thermal stratification pipe loads in response to Generic Letter 88-11 (Reference 3). The analysis update for the uprating considered all the previously reported changes to the original analysis.

4.8.3 Acceptance Criteria and Results

The evaluation showed that the pressurizer components will continue to meet the ASME Code stress and fatigue requirements for the uprated conditions. The new total fatigue usage factor for each component was determined to be less than 1.0 per the ASME Code.

4.8.4 Conclusions

The results of the pressurizer analysis show that the Turkey Point Plant Units 3 and 4 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III for the plant operation in accordance with the uprating program.

4.8.5 References

1. Equipment Specification 676359, Revision 1, "Reactor Coolant System, Florida Power and Light – Turkey Point Unit No. 3, 1300 cu. ft. Pressurizer," Westinghouse Electric Corporation, Atomic Power Division, Pittsburgh, Pennsylvania, March 1969.

2. Equipment Specification 676458, Revision 3, "Reactor Coolant System, Florida Power and Light – Turkey Point Unit No. 4, Pressurizer," Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, Pennsylvania, January 1975.

3. NRC Generic Letter 88-11, "Pressurizer Surge Line Stratification" dated 12/20/88.

4.9 STEAM GENERATORS

4.9.1 Thermal/Hydraulic Evaluation

4.9.1.1 Introduction

The thermal hydraulic evaluation of the steam generators at uprated conditions has been assessed and determined to be acceptable.

4.9.1.2 Input Parameters and Assumptions

Applicable design parameters for operation at uprated conditions were used for the thermal/hydraulic evaluation. The operating steam generator water level was assumed to be at 60% of narrow range span. The as-built steam generator configuration was used for calculation of thermal/hydraulic operating characteristics.

The design fouling factor was originally assumed conservatively high to provide a margin factor for steam pressure performance. In the absence of significant field performance experience, a large value was used to assure that design steam pressure was met. As in the case of Turkey Point, this value was often carried over to the replacement units. Increasing field experience showed that the large values of design fouling factor were very conservative. The uprated value defines a more realistic design operating point, permits lower design operating temperatures and still provides adequate margin so that the generator is assured of meeting the design steam pressure.

4.9.1.3 Evaluations Performed

The steam generator thermal/hydraulic evaluation of the Turkey Point Units 3 and 4 steam generators included several facets. Operating characteristics of the steam generators at all the uprated conditions were calculated. Attention was focused on secondary side parameters. Parameter values calculated for uprated conditions are compared to the values at the current design conditions. Where appropriate, the parameter values are compared to other existing field experience. In addition, the question of voiding below the water level and its effect on level setpoint is addressed.

4.9.1.4 Thermal/Hydraulic Operating Characteristics

Several secondary side operating characteristics were used to assess the acceptability of steam generator operation at uprated conditions.

The circulation ratio (CR) is a measure of liquid flow in the bundle in relation to the steam flow. It is primarily a function of power. The CR decreases for the uprated condition. Since the steam flow increases with power, the bundle liquid flow decreases at the same condition. The bundle liquid flow minimizes the accumulation of contaminants on the tubesheet and in the bundle. The uprating has no material effect on this function.

The total bundle flow rate remains essentially unchanged with uprating. The increase in steam flow and concurrent increase in void fraction result in an increased potential for vibration in the U-bend region. This circumstance, however, does not contribute to any significant decrease in long term bundle integrity for the Model 44F steam generators.

The hydrodynamic stability of a steam generator is characterized by the damping factor. For uprated conditions, damping factors are seen to remain negative at about the same level as current design. All the uprated conditions, therefore, continue to be hydrodynamically stable.

The reduced steam pressure brings about an increased void fraction in the tube bundle. This causes a small reduction in steam generator mass that is not considered significant.

The maximum calculated heat flux at uprated conditions is well within nucleate boiling limits and is lower than values for steam generators currently operating in the field.

The increase in average heat flux will cause some increased potential for corrosion and long term fouling though it is not the dominant factor. Operating temperatures and plant chemistry coupled with plant materials are more significant factors. Operating history to date, more than changes which will result from uprating, is the best indication of whether the Turkey Point units are susceptible to significant corrosion or performance loss due to fouling.

The maximum increase, 3 psi, in total secondary side pressure drop for the steam generator is very small in relation to the total feed system pressure drop. This should have no significant effect on the feed system operation.

In summary, the thermal/hydraulic operating characteristics of the Turkey Point Units 3 and 4 steam generators are within acceptable ranges for all anticipated uprated conditions.

4.9.1.5 Acceptance Criteria and Results

The thermal/hydraulic characteristics of the steam generators were evaluated with respect to plant safety as to the operational stability and secondary side measurements that are used for trip functions. Steam generator stability involves the behavior of the unit in response to perturbations to the operating parameters. The measurement of secondary side level is performed by the narrow range taps.

4.9.1.6 Conclusions

The thermal/hydraulic operating characteristics are within acceptable ranges for all anticipated uprating conditions. This evaluation has shown that the steam generator uprated thermal/hydraulic conditions are within an acceptable range and are similar to the current conditions. The current high level setpoint of the secondary side will perform as intended.

4.9.2 Structural Evaluation

4.9.2.1 Introduction

A structural evaluation of the steam generators was performed at the uprated conditions. The structural integrity of the steam generators at the increased thermal rating has been assessed and determined to be acceptable.

4.9.2.2 Input Parameters and Assumptions

The parameters for steam generator structural evaluation covered six uprated condition cases. Cases were analyzed for a steam generator without tube plugging, and for 20% tube plugging. Variations in the primary and secondary temperatures under high and low temperature uprating conditions at full normal power operations are within ±1% of the reference conditions. Variations in the secondary side pressure are about ±6% and those for the primary-to-secondary pressure differential are within about ±3%. The multiplying factors to be used for adjusting pressure induced stresses under steady-state conditions to obtain stresses for the uprating conditions are: 1.01 for primary side pressure; 1.06 for secondary side pressure; 1.03 for primary to secondary pressure differential.

4.9.2.3 Evaluation Criteria

The design transient applicable for the uprated conditions are in general more severe than the previous ones. Comparison with the original transients indicates that the primary side temperature variations are somewhat greater for the uprated cases. Thermal gradients across the thickness of steam generator components do not change drastically. Secondary side transients basically remain unchanged. Primary to secondary pressure differential changes were evaluated and the stress range multiplied by the appropriate factor for the transients affected.

The critical steam generator components evaluated structurally were the tubesheet, tubesheet junctions, tube to tubesheet weld, tubes, secondary shell, minor shell penetrations and the feedwater nozzle. The divider plate is not a critical pressure boundary component, but it was also evaluated for a higher pressure drop across the plate at a plugging level of 20%.

4.9.2.4 Conclusions

Results of analysis performed above for the Turkey Point Units 3 and 4 Model 44F steam generator components show that structural integrity of the components would be maintained for operation at the uprated power level with a maximum plugging level of 20% in the steam generator.

4.10 FUEL

4.10.1 Fuel Assembly Structural Evaluation

4.10.1.1 Introduction

The current fuel design in place at Turkey Point Units 3 and 4, which is 15x15 Debris Resistant Fuel Assembly (DRFA), was evaluated at the uprated power conditions to ensure that it still meets the applicable design criteria. The fuel assembly design was evaluated to show that it was structurally adequate to support the increased power level.

4.10.1.2 Description of Evaluations/Acceptance Criteria

The maximum spacer grid loads and assembly deflections for LOCA conditions were determined for the uprated power. The grid loads and assembly deflections were compared to those from the original Turkey Point analysis of the DRFA. The maximum grid loads obtained from seismic and LOCA loading analyses were also combined using the square root sum of the squares (SRSS) method.

The design lift forces for the uprating were compared to the generic 15x15 Optimized Fuel Assembly (OFA) design in order to verify the fuel assembly holddown spring capability under the uprating conditions.

4.10.1.3 Results

The results indicate that both spacer grid load and assembly deflection are lower than those from the original analysis of the DRFA in the Turkey Points units. Thus, the most recent LOCA analyses results remain applicable for the DRFA in both Turkey Point units.

Results of the seismic and LOCA peak grid loads and the combined grid load, show the load is significantly less than the grid strength. Based on these results, the 15x15 DRFA designs are structurally acceptable for both Turkey Point units.

It was also determined that the design lift forces for Turkey Point 3 and 4 under uprated conditions are bounded by the generic 15x15 OFA design. The fuel assembly holddown spring capability is therefore verified.

4.10.1.4 Conclusions

The Turkey Point 3 and 4 Debris Resistant Fuel Assembly design was determined to be structurally acceptable for the uprated conditions. The fuel assembly holddown springs were also found to be acceptable.

4.10.2 Fuel Rod Design Analysis

4.10.2.1 Introduction

A fuel rod design analysis was performed to determine the impacts of the uprating on fuel rod design. This section summarizes the fuel rod design analyses performed to determine if the design criteria impacted by the uprating are met.

4.10.2.2 Evaluations and Results

Fuel rod evaluations were performed to determine the impact of the uprated core power on fuel performance. Evaluations of the fuel rod design criteria impacted by the uprating, including rod internal pressure, cladding stress and strain, cladding fatigue and cladding corrosion were performed at the uprated conditions. These fuel rod design evaluations, performed with the NRC-approved models, have shown that fuel rod design criteria can be satisfied at the uprated core conditions.

4.10.2.3 Conclusions

It has been demonstrated that the fuel rod design criteria will be satisfied at the uprated core conditions. Furthermore fuel performance evaluations are completed for each fuel region and cycle to demonstrate that all fuel rod design criteria will be satisfied under the planned operating conditions as part of the reload safety evaluation process performed during each reload cycle.

4.11 NSSS AUXILIARY SYSTEM COMPONENTS

4.11.1 Introduction

The NSSS auxiliary equipment of Turkey Point Units 3 and 4, such as valves, pumps, tanks and heat exchangers, have been evaluated for the uprated conditions.

4.11.2 Input Parameters and Assumptions and Description of Evaluations Performed

The impact of the uprating on the maximum system operating temperatures and pressures were evaluated. The increased decay heat and post-accident conditions were considered. The maximum temperature of the component cooling water supplied to auxiliary equipment was also evaluated.

An evaluation of the maximum operating temperatures and pressures was performed on the following equipment:

- Residual heat removal system (RHRS), component cooling water system (CCWS), containment spray system (CSS), and Spent Fuel Pool (SFP) valves

- CSS, CCWS, SFP, SFP skimmer, charging, RHRS, and HHSI pumps

- RHRS, CCWS, nonregenerative, sample, excess letdown, seal water, and SFP heat exchangers

- Boron injection and CCWS surge tanks

- Waste gas compressors

- Radiation Monitors R-17A & B (Component Cooling Water).

The impact of changes to thermal transients was evaluated on the following equipment:

- RHRS, CCWS, CSS and SFP valves
- CSS, CCWS, SFP, SFP skimmer, charging (PD), RHRS, and HHSI pumps
- RHRS, CCWS, and SFP heat exchangers
- Boron injection and CCWS surge tanks.

The impact of increased cooling water temperatures was evaluated for the following equipment:

- RHRS, CSS, HHSI, and charging pumps.

4.11.3 Acceptance Criteria

The evaluation of the NSSS Auxiliary equipment for the uprated condition depends on the comparison of the following values to the original design conditions:

- Maximum operating temperatures and pressures,
- Revised thermal transients
- Increased cooling water temperatures.

If the uprated parameters were bounded by the original design values, then the auxiliary equipment remain qualified for the uprating program. If the revised parameters were not bounded, then the affected equipment needed to be requalified.

4.11.4 Results

All maximum operating temperatures and pressures are bounded by the existing systems design bases. Therefore, the auxiliary equipment is qualified for the maximum operating temperatures and pressures resulting from the Uprating Program. Also, the auxiliary equipment thermal transients resulting from the uprating parameters are bounded by the original design parameters. Therefore, the auxiliary equipment remains qualified for thermal transients for the Uprating Program. The evaluations performed demonstrate that the Turkey Point RHRS, CSS, HHSI, and PD Charging pumps will operate as designed for the CCW conditions at the uprated parameters.

4.11.5 Conclusions

The NSSS Auxiliary equipment at Turkey Point Units 3 and 4 have been evaluated for the uprated conditions. Upon considerations of peak system temperatures and pressures, thermal transients and increased CCW temperatures, it was determined that the evaluated equipment will function as intended for the uprated conditions.

4.12 TURBINE GENERATOR (TG) COMPONENTS

The critical components of the high and low pressure turbines were evaluated to establish that structural integrity and functional adequacy can be maintained at the 2308 MWt (NSSS) uprated conditions. This review included the stationary parts of the high pressure and low pressure cylinders, blade rings, nozzle blocks, high pressure blading, low pressure blading, piping, Moisture Separator-Reheater's (MSR), extraction piping, and valves. In addition, the rotating blading and rotors of both the high and low pressure turbines were evaluated. The turbine auxiliary interface was also evaluated, including the main steam inlet.

The basis for this evaluation was a review of the expected design conditions at the uprated power level. These conditions were compared to the applicable design criteria to determine the acceptability of operation at the higher power level. A review was performed for both the thermodynamic operation of the equipment and the mechanical function. In cases where design margin was minimal, plant operating data was also considered to evaluate whether the component could be approved for uprating.

A review of all the turbine components and turbine auxiliaries meet the design criteria for the 2308 MWt uprating. Based on the evaluation, it can be concluded that operation at the uprated power level for the TG components is acceptable.

4.13 CONCLUSIONS

NSSS components were re-evaluated and results compared to the allowable stress fatigue limits defined by the ASME Code Editions to which the components were originally designed and evaluated. The revised conditions and transient loadings resulted in stresses and fatigue usage factors below the Code allowable limits. The conservative assumptions of the original stress report remain bounding for the revised conditions reflected in Table 2.1-1, and therefore the original conclusions remain unchanged. Therefore, it has been determined that the NSSS components will not be adversely affected by the uprating for an NSSS power level of up to 2308 MWt.

CHAPTER 5

NSSS AND TURBINE GENERATOR (TG) SYSTEMS REVIEW

5.0 NSSS AND TURBINE GENERATOR (TG) SYSTEMS REVIEW

5.1 INTRODUCTION

The impact of the uprating and associated conditions (as described in Table 2.1-1) was evaluated for the NSSS Fluid Systems, Control Systems, Reactor Protection Systems, NSSS/BOP Interface Systems, and Turbine Generator Systems. The purpose of these evaluations was to confirm that the NSSS and TG systems continue to perform their design functions acceptably at the uprated conditions.

5.2 NSSS FLUID SYSTEMS

5.2.1 Reactor Coolant System

5.2.1.1 Introduction

The Reactor Coolant System (RCS) consists of three heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP), which circulates the water through the loops and reactor vessel, and a steam generator (SG), where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer which controls the RCS pressure through electrical heaters, water sprays, power operated relief valves (PORVs) and spring loaded safety valves. The steam discharged from the PORVs and safety valves flows through interconnecting piping to the pressurizer relief tank (PRT).

The key RCS functions are as follows:

- To transfer heat generated in the reactor core to the MSS via the SGs,
- To transfer decay and sensible heat to the Residual Heat Removal System (RHRS) when the core is subcritical and RCS temperatures are approximately 350°F and lower,
- The RCS fluid acts as a moderator of neutrons,
- The RCS fluid is a solvent and carrier of boric acid which is used as a neutron poison,
- The RCS is a barrier against fission product release,
- The RCS provides means for pressure control via use of pressurizer heaters, spray flow, PORV's and safety valves.

The calculated uprated RCS design operating conditions include increases in core power, and the allowable operating range for average RCS temperature (Tavg). The potential impact of the uprated conditions on the previous RCS functions are described below:

- The core power increase will affect the total amount of heat transferred to the MSS.

- During the second phase of plant cooldown, the RHRS will be required to remove larger amounts of decay heat from the RCS as the reactor core is operating at a higher power level. However, at plant shutdown conditions, the RCS conditions are not affected.

- The thermal uprating project can change the transient response of the RCS during normal and postulated design basis events. The acceptability of the RCS with respect to control and protection functions has been demonstrated in this report.

- With higher core power levels, the decay heat levels that must be cooled by the Spent Fuel Pool Cooling System (SFPCS) are increased. Section 5.5.5 addresses the SFPCS capabilities and associated changes to operating temperatures at uprated conditions.

- With higher core power levels, the amount of boric acid required to achieve desired shutdown margins can increase. Section 5.2.2 of this report addresses boration capabilities at uprated conditions.

- With higher core power and increased SG tube plugging, RCS available volume and RCS loop flows can decrease, which can reduce pressurizer spray flow capability since loop velocity head is used for driving head. In addition, a range of steady state full power RCS operating temperatures is established. This range, in turn, can cause changes in nominal pressurizer level which can change the steam release potential to the PRT.

5.2.1.2 Input Parameters and Assumptions

The evaluation of the RCS at the uprated condition required that the following changes be considered:

- Higher SG tube plugging levels reduces the available RCS liquid volume. To provide design input to the calculation of revised RCS source terms, a minimum RCS liquid mass at full power operating conditions was calculated.

- Higher SG tube plugging may reduce available loop flows. For RCS loops used for pressurizer spray flow, lower RCS flows reduces the available driving head for spray. To support RCS transient response and plant safety analyses, a range of pressurizer spray flow under full spray operation was calculated.

- The range of RCS operating temperatures provided in Section 2.0 of this report were used as a basis to evaluate RCS design temperatures.

- Operation at a lower RCS Tavg condition increases the available pressurizer steam space volume that may have to be condensed in the PRT under limiting RCS transient conditions (e.g., loss of load event).

- In the cases where a setpoint may be potentially affected, the FPL I&C Matrix Instrument List was reviewed to verify it's adequacy relative to the current process control setpoint value.

5.2.1.3 Description of Analyses/Evaluations Performed

To determine the RCS minimum hot full power liquid mass, the allowable SG tube plugging was considered as well as the limiting masses of other components and other calculation parameters were used to provide a conservative RCS mass condition. To determine pressurizer spray flow capability, a detailed flow calculation was performed which define the expected minimum, nominal and maximum pressurizer spray flow as a function of assumed RCS loop flow. Expected variations in component hydraulic data were considered to provide a range of expected flows.

Assessing system operation at the higher range of RCS Tavg condition, the maximum expected RCS Thot temperature was compared to RCS design temperatures. In the assessment of system operation at the lower RCS Tavg condition, the available steam space volume in the pressurizer was compared to that assumed in the PRT design basis calculation to assess available margin.

5.2.1.4 Acceptance Criteria for Analyses/Evaluations

In the calculation of a revised minimum RCS hot full power liquid mass, no specific criteria had to be met. The calculation biased inputs to establish a conservative (minimum) value.

In the assessment of system operation at the range of RCS Tavg conditions, the maximum expected RCS Thot temperature must be less than or equal to the applicable RCS design temperature to ensure pressure boundary integrity.

The acceptance of the PRT relief capability is not based on a safety function but on a desirable criterion of precluding contamination of containment following a maximum expected pressurizer discharge.

5.2.1.5 Results

Pressurizer spray flow capability was calculated considering a range of component hydraulic conditions at the revised RCS Thermal Design Flow (TDF) of 85,000 gpm per loop. The minimum calculated total spray flow continues to meet the acceptance criteria.

With respect to maximum expected RCS Hot Leg (Thot) temperature, the uprated condition temperature is well within the RCS loop design temperature of 650°F. Note, the pressurizer and the surge line has a higher design temperature of 680°F.

With respect to the PRT, the revised range of RCS Tavg has the potential to change the nominal full load pressurizer steam volume at uprated conditions. In general, the reference nominal pressurizer level is coordinated with RCS Tavg such that an increase in Tavg raises the nominal pressurizer reference level condition. With respect to the PRT discharge analysis, a lower RCS Tavg condition is potentially more limiting since pressurizer level is lower (steam volume is higher).

Although the revised nominal steam volume at uprated power can be somewhat greater than the PRT original sizing basis value, the inherent availability of 10 percent steam volume conservatism in the sizing calculation would more than compensate for the possible uprating increase.

5.2.1.6 Conclusions

The acceptability of the revised RCS operating conditions at uprated power has been evaluated. The overall conclusion is that the RCS can continue to perform its design basis functions without any anticipated plant changes.

5.2.2 Chemical and Volume Control System

5.2.2.1 Introduction

The Chemical and Volume Control System (CVCS) is designed as an interface to the Reactor Coolant System (RCS). Its primary design function is to maintain the required water inventory, soluble boron concentration and water chemistry of the RCS. Other CVCS functions include filling and draining the RCS, reducing the quantity of fission and corrosion product impurities in the RCS, and supplying seal injection flow to the reactor coolant pumps (RCPs). In addition the CVCS meets the requirement in 10 CFR 50 Appendix A which states that there be two independent means of reactivity control, one of which is not the control rods. CVCS reactivity control is performed with the injection of boric acid solution, which is a neutron absorber, into the RCS.

During normal plant operation, the CVCS provides the charging and letdown to the RCS. Charging is generally performed with one of three positive displacement pumps. In addition to providing charging flow and pressurizer auxiliary spray, the charging pumps also provide seal injection flow to the RCPs.

5.2.2.2 Input Parameters and Assumptions

Provided below is a list of key input parameters used on the assessment performed on this system:

- Of the specified changes in RCS operating conditions addressed by this project, the most significant change due to uprating is the increase in the reactor core power level. In general, the higher reactor core power level may require the CVCS to borate the RCS to a higher concentration at a faster rate. The adequacy of the boron concentrations of the BAST and RWST will need to be assessed.

- Since the CVCS interfaces with the RCS, specifically the RCS cold and intermediate legs, a change in RCS design temperature may also have an impact on the CVCS functions.

5.2.2.3 Description of Analyses/Evaluations Performed

The present CVCS boration capability was evaluated at the uprated conditions. Specifically, the minimum amounts of boric acid (boric acid concentrations) in the BAST and RWST presently required in the Turkey Point Units 3 and 4 Technical Specification were reviewed to assure they are sufficient in meeting the Turkey Point Units 3 and 4 Technical Specification (See Section 5.2.2.4).

In the assessment of system operation at the higher range of RCS Tavg condition, the maximum expected RCS temperature was compared to the CVCS design temperatures. Specifically, the RCS cold leg and intermediate leg temperatures at the uprated conditions were evaluated since letdown occurs at the Loop B cold leg and alternate letdown occurs at the Loop A intermediate leg.

5.2.2.4 Acceptance Criteria for Analyses/Evaluations

The following CVCS boration requirements are specified in the Turkey Point Units 3 and 4 Technical Specifications:

- The amount of boric acid in the BAST and RWST, individually, is sufficient to borate the RCS to cold shutdown (Mode 5) conditions from Mode 1 through 4. This includes the amount of boric acid needed to achieve the required shutdown margin in Mode 5.

- In Modes 5 and 6, the amount of boric acid in the BAST and RWST, individually, is sufficient to account for the effects of RCS shrinkage and the moderator temperature coefficient.

- The CVCS boration rate is sufficient to keep up with the rate at which Xenon burns out after the peak.

Besides these requirements from the Turkey Point Technical Specifications, the CVCS performance at the uprated conditions were compared with the design basis. Presently, with one charging pump and either boric acid transfer pump in operation, enough boric acid can be injected into the RCS (during a feed and bleed process) to take the reactor to hot shutdown within forty minutes. In an additional forty minutes, enough boric acid is injected into the RCS to compensate for Xenon decay.

In addition to the CVCS boration requirements, the change in RCS operating conditions need to be assessed. The maximum expected RCS cold leg temperature must be less than or equal to the applicable CVCS design temperature.

5.2.2.5 Results

The maximum expected RCS cold leg temperature at uprated conditions is well within the CVCS mechanical design temperature of 650°F. The CVCS operating design temperature is limited by the capability of the regenerative heat exchanger. The maximum cold leg (and intermediate leg) temperature at the uprated conditions is also below this temperature. Therefore, the RCS effluents at the uprate conditions are within the design conditions of the CVCS. The acceptance criteria of Section 5.2.2.4 are satisfied.

5.2.2.6 Conclusions

The evaluation of the CVCS at the uprated conditions has been performed. The CVCS can continue to perform its design basis functions at the uprated condition of the plant.

5.2.3 Safety Injection System/Containment Spray System

5.2.3.1 Introduction

The Safety Injection System (SIS) and the Containment Spray System (CSS) are Engineered Safeguards Systems. They mitigate the effects of postulated design basis events by providing core and containment cooling.

The passive portion of the SIS consists of the three accumulator vessels which are connected to each of the RCS cold leg pipes.

The active portion of the SIS is comprised of a high pressure and a low pressure injection subsystem. Both subsystems utilize centrifugal pumps which are normally in a stand-by mode and automatically start following generation of a Safety Injection (SI) signal.

The CSS also employs centrifugal pumps which are normally in a stand-by mode and automatically start following generation of a High-High containment pressure condition. These pumps are initially aligned to take suction from the RWST and deliver borated spray water in the upper portion of the containment volume.

As the design basis event proceeds, the RWST water inventory decreases as water is transferred to the RCS and/or containment building. Upon depletion of a majority of the RWST inventory on the affected unit, the operating SIS and CSS pumps are required to be realigned to support the cold leg recirculation mode of operation.

At approximately 12 hours from event initiation, the SIS subsystem is realigned a second time to support the hot leg injection mode of operation. This time has been reduced from the current 18 hours due to increases in core decay heat associated with the uprate.

5.2.3.2 Input Parameters and Assumptions

In general, the specified changes in RCS operating conditions addressed by the uprating have no direct effect on the overall performance capability of the SIS and/or CSS. These systems will continue to deliver a selected range of calculated flow performance as determined by interfacing system/structure operating conditions (RCS pressure, containment pressure, etc.). The acceptability of a given range of SIS and CSS performance is inherently justified by acceptable plant safety analyses results. For this project, numerous plant safety analyses were reanalyzed or evaluated.

For the High Head Safety Injection (HHSI) subsystem, a reduced minimum pump performance curve was used. The primary effect of this change on subsystem performance is a reduction in both "Cut-In" pressure and minimum flow delivery capability. Since the minimum allowable pump head decreased, revised flows were recalculated. The Technical Specification surveillance requested for the SI pumps (T.S. 3/4.5.2, 4.5.2c) is being revised in accordance with reduction of pump head by 100 feet.

5.2.3.3 Description of Analyses/Evaluations

As noted previously, the overall performance of the SIS and CSS are independent of the RCS operating conditions being evaluated as part of this project. The RCS operating conditions defined for this project do not affect system flow performance. As such, existing flow capabilities were generally used except as noted for the HHSI subsystem.

For the HHSI subsystem, revised minimum injection mode flows were specifically recalculated to consider a further degraded pump performance curve.

5.2.3.4 Acceptance Criteria for Analyses/Evaluation

The general acceptability of system performance are documented in the individual plant safety analyses that utilize such inputs.

5.2.3.5 Results

The performance of the SIS and CSS is independent of the thermal uprating analysis. The acceptability of the systems at uprated conditions is justified by acceptable safety analysis results.

5.2.3.6 Conclusions

As stated previously, the general acceptability of the existing and newly calculated SIS and CSS operating parameters defined for this project are documented in the various discussions of individual plant safety analyses results as summarized in Section 3.0 of this report.

5.2.4 Residual Heat Removal System

5.2.4.1 Introduction

The Residual Heat Removal System (RHRS) is a dual function system. During normal power operation, the system is in a stand-by mode to support its Engineered Safeguards function (i.e., safety injection). During the second phase of plant cooldown and the plant shutdown mode of operation, the RHRS is used to remove Reactor Coolant System (RCS) sensible and decay heat. This section discusses the RHRS normal functions (i.e., heat removal). The Engineered Safeguards functions of the RHRS are discussed in Sections 5.2.3 (SIS) and 5.5.2 (CCWS).

The maximum heat removal demand on the RHRS generally occurs during the plant cooldown mode of operation when RCS sensible heat (e.g., metal mass), core decay heat and heat input from a Reactor Coolant Pump (RCP) must all be removed to support RCS temperature cooldown. In addition, operating restrictions are imposed on the maximum allowable CCWS temperature during cooldown which can also restrict RHRS heat removal capability.

The overall RHRS heat removal capability can vary significantly depending on system equipment availability, cooling support system equipment availability, cooling support system flows, RHRS and CCWS heat exchanger thermal performance (e.g., fouling level) and ICW System inlet temperature.

The Turkey Point system design basis considered only the normal cooldown condition with all RHRS and associated cooling system equipment available. Once plant cooldown is achieved, only one train of RHRS equipment and associated cooling system support equipment is generally used to maintain RCS temperature.

5.2.4.2 Input Parameters and Assumptions

Of the changes in RCS operating conditions due to the uprating, only the increase in reactor core power level has a significant effect on RHRS thermal performance capability. Specifically, higher core power levels will increase RCS decay heat loads, which must be removed during plant cooldown and shutdown conditions. From a hydraulic (flow) perspective, the revised RCS operating conditions have no direct impact on the flow delivery capability of the RHRS. Likewise, existing instrumentation and controls are independent of uprated conditions and do not need to be evaluated.

For this project, RHRS thermal performance were calculated for the following cooldown scenarios:

- The ability of the RHRS to accept the RCS heat removal function during the second phase of plant cooldown (i.e., RHRS Cut-In).

- The ability of the RHRS to cool down the RCS with all equipment operating to a cold shutdown condition (200°F) and a refueling condition (140°F). Note: RHRS operation with all equipment available (including support systems) is referred to as a "normal" plant cooldown within the context of this section.

- The ability of the RHRS to cool down the RCS under a limiting Appendix R postulated fire incident.

- The ability of the RHRS to cool down the RCS under limiting equipment availability to a cold shutdown condition (200°F).

A set of thermal analysis operating conditions which would bound both current and expected thermal uprate plant conditions was developed.

5.2.4.3 Description of Analyses/Evaluations

The thermal (cooldown) performance of the RHRS was evaluated for the scenarios of RHR initiation, normal cooldown, Appendix R cooldown and abnormal cooldown. The evaluation of these scenarios considered normal equipment alignment and various cases of selected components unavailable. These cases assumed operation of RHRS, CCWS, ICW and RCS equipment in different configurations. Normally one RCP is in operation during RHRS cooldown to promote mixing within the RCS loops. Several cases were analyzed assuming no RCPs in operation.

5.2.4.4 Acceptance Criteria for Analyses/Evaluation

During a typical plant cooldown event, the RCS is cooled from its "no-load" condition to the RHRS cut-in condition of 350°F by providing secondary side water to the steam generators. For abnormal conditions where the Condensate Storage Tank (CST) will provide the water, the RHRS must be capable of accepting the RCS heat removal prior to depletion of the CST inventory.

The normal cooldown scenario assumes all cooling trains are available. The original RHRS equipment sizing criteria was selected to achieve a refueling condition in a certain time based on economic considerations. As such there is no a specific design basis acceptance criterion for the normal cooldown. However, the standard Technical Specifications typically specify a 36-hour time duration for achieving cold shutdown under certain conditions.

For an Appendix R event, the plant is required to be in cold shutdown within 72 hours of the event initiation considering the possibility of a concurrent loss-of-offsite power condition. Credit can be taken for operator action. The Appendix R cooldown assumes only a single active train of cooling equipment is available.

5.2.4.5 Results

Under various scenarios of different equipment either available or not, the RHRS was found capable of accepting the RCS heat removal function within the required time frame.

For normal cooldown under the most restrictive operating parameters and with all cooling equipment available, the RHRS is able to cool the RCS to Cold Shutdown conditions within the standard Technical Specification cooldown time of 36 hours.

For an Appendix R cooldown under the most restrictive operating parameters with one train of active cooling equipment available and loss of offsite power, it was determined that the RCS could be cooled down within the criteria.

5.2.4.6 Conclusions

The evaluation of the RHRS at the uprated conditions has been performed and it is concluded that the RHRS can perform its design basis functions.

5.2.5 NSSS Sampling System

5.2.5.1 Introduction

The NSSS Sampling System is designed to permit remote sampling of fluids from the Reactor Coolant System (RCS) by means of permanently installed lines. The system can obtain samples from the pressurizer liquid and steam space, the RCS A and B hot legs, the three accumulators, the residual heat removal loop, the letdown lines at the inlet and outlet of the demineralizers, the Volume Control Tank (VCT) gas space and the three steam generators (secondary side).

The sampling system integrity and performance is directly linked to the systems to which it is connected. Of the systems where samples are taken from, the RCS yields the highest pressure and temperature challenge to the sampling system. Therefore, the sampling system has been designed to bound the RCS maximum temperature and pressure.

At the uprated conditions, the changes in the RCS operating conditions may affect the performance of the sampling system. The samples, especially those taken directly from the RCS, are impacted by the changes in the RCS which include increases in core power and the allowable operating range for average RCS temperature (Tavg). The calculated RCS design operating conditions are based on the uprated conditions.

5.2.5.2 Input Parameters and Assumptions

The range of RCS operating conditions are the basis for determining limiting conditions on the sampling system. The limiting conditions are compared to the original system design parameters.

5.2.5.3 Description of Analyses/Evaluations Performed

In the assessment of system operation, the maximum sample temperature based on the maximum expected RCS temperature are compared to the sampling system mechanical design temperature.

5.2.5.4 Acceptance Criteria for Analyses/Evaluations

In the assessment of system operation, the maximum expected sample temperature and pressure must be less than or equal to the applicable sample system mechanical design temperature. This ensures the integrity of the system.

5.2.5.5 Results

Of the changes due to the thermal power uprating, the increase in range of RCS temperatures has the most direct impact on the sampling system. At the maximum uprated Thot temperature, samples taken from the RCS hot leg are well within the sampling system mechanical design temperature.

However, with respect to samples taken from the RCS, the pressurizer liquid and steam samples yield the most limiting conditions on the sampling system. The samples taken from the pressurizer are at a higher temperature since they are initially at saturated conditions. For the thermal power uprating the RCS operating pressure is not affected; therefore, the maximum expected temperature of the samples taken from the Pressurizer is also not affected (i.e., the saturation temperature at the RCS pressure is unchanged). The performance of the sample system piping and sample heat exchangers remain acceptable for the thermal power uprating.

5.2.5.6 Conclusions

The sampling system can continue to perform its design basis functions without any anticipated plant changes.

5.2.6 Head Vent/Pressurizer Vent

5.2.6.1 Introduction

The Reactor Coolant System (RCS) is provided with two primary vent paths for post-accident hydrogen venting and to support plant operations. One vent is provided near the top of the pressurizer and is tied-in to the common piping which connects the pressurizer head to the two power operated relief valves. The second vent is tied-in to a connection near the top of the Reactor Vessel (RV) head. For both vent paths, two power operated isolation valves are provided in parallel to ensure that a given path can be opened. These vents also utilize common discharge piping that allows flow to be directed to either the Pressurizer Relief Tank (PRT) or directly to containment (if desired). The two discharge paths are each provided with a power operated isolation valve to ensure positive isolation.

In general, the RCS vents are used to support normal plant operations (e.g., RCS draining and filling) and post-accident conditions (e.g., vent non-condensible gases that can interfere with core cooling). The safety related functions of the vent lines are 1) to maintain RCS pressure boundary integrity when the system is not in use, 2) to support venting operation when required during post-accident conditions and 3) to be capable of being isolated following venting operations.

The primary changes due to uprating include increases in core power, SG tube plugging level and the allowable operating range for average RCS temperature (Tavg). The potential impact on the RCS vent systems are described below:

- With changes in RCS operating conditions, the operating temperature of vented fluid can either increase or decrease.

- With increased SG tube plugging, RCS available volume can decrease.

5.2.6.2 Input Parameters and Assumptions

In evaluating the uprated condition, a 20% SG tube plugging level was considered because it reduced the available RCS volume. The range of RCS operating temperatures at the uprated condition was used to evaluate the adequacy of the vent system design temperature. Only the portion of vent piping that comprises the RCS pressure boundary is required to be evaluated to ensure pressure boundary integrity.

5.2.6.3 Description of Analyses/Evaluations

In general, an evaluation process was used to assess the overall acceptability of the vent systems at the Thermal Uprate revised operating conditions. To assess the impact of increased S/G tube plugging, the RCS volume basis used in sizing the vent systems was reviewed. To assess the impact of revised

RCS operating temperatures, the maximum expected RCS Thot temperature was compared to system design temperatures.

5.2.6.4 Acceptance Criteria

In the assessment of system vent sizing, the actual RCS volume should be less than or equal to the volume criteria to ensure that venting durations remain bounding. In the assessment of system operation at the higher range of RCS Tavg condition, the maximum expected RCS Thot temperature must be less than or equal to the applicable system design temperature to ensure pressure boundary integrity.

5.2.6.5 Results

The RV head vent flow rate capability is based on venting 1/2 of the RCS volume within a one-hour duration. Since the net effect of any S/G tube plugging is a reduction in RCS total volume, the existing system venting flow rate capacity is unaffected.

With respect to revised RCS operating temperatures, the uprated Thot temperature is increased but is lower than the design condition of the head vent and pressurizer vent piping that comprises the RCS pressure boundary. Since the revised Thot temperature is well within these design conditions, pressure boundary integrity is ensured.

5.2.6.6 Conclusions

Based on the evaluation outlined in this section, the pressurizer and RV head vent systems are not impacted by the changes in RCS operating conditions associated with the Thermal Uprate project. As such, these systems can continue to perform their design basis functions without requiring any plant changes.

5.3 CONTROL SYSTEMS

Control systems were evaluated in order to verify that adequate margin to reactor protection systems setpoints exists at the uprated conditions for the following design basis transients:

- 50% load rejection from full power
- 50% load rejection from 50% power
- 10% step load decrease
- 5% per minute unit loading/unloading

Results of these analyses indicate that adequate margin exists and that the plant is adequately stable at the uprated conditions. As such, no changes to control systems setpoints are recommended.

5.4 REACTOR PROTECTION SYSTEM/ENGINEERED SAFETY FEATURES ACTUATION SYSTEM SETPOINTS

The Technical Specification Reactor Protection System/Engineered Safety Features Actuation System setpoints, and the Core Operating Limits Report have been reviewed for plant operation at a core power level up to 2300 MWt for the RCS flow limit. As part of the review, Technical Specification changes were necessary to meet NRC approved Westinghouse setpoint and RTDP methodologies (References 1, 2, 3, and 4).

Tables 5.4-1 and 5.4.2 list both the current and proposed values for each function and parameter impacted. Incorporating these Technical Specification changes will ensure that the Turkey Point Units 3 & 4 will operate in a manner consistent with the UFSAR assumptions.

References:

1. WCAP-12632, "RTD Bypass Elimination Licensing Report for Turkey Point Units 3 & 4," June 1990.

2. WCAP-12745, Revision 1, "Westinghouse Setpoint Methodology for Protection Systems -- Turkey Point Units 3 & 4," December 1995.

3. WCAP-13719, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology -- Florida Power & Light Company Turkey Point Units 3 & 4," January 1995.

4. WCAP-13719, Revision 2, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology -- Florida Power & Light Company Turkey Point Units 3 & 4," June 1995.

Table 5.4-1
Summary Of The Reactor Protection System Setpoint Changes

**TURKEY POINT TECHNICAL SPECIFICATION TABLE 2.2-1
REACTOR TRIP SYSTEM INSTRUMENTATION
CURRENT AND PROPOSED TRIP SETPOINTS**

Overtemperature ΔT Reactor Trip

Functional Unit 5	*Trip Setpoint*		*Allowable Value*	
	Current Value[(a)]	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
K_1	1.25	1.24	0.73	0.84
K_2	0.016	0.017	N/A	N/A
K_3	0.0011	0.0010	N/A	N/A
T'	≤574.2°F	≤577.2°F	N/A	N/A
-ΔI Gain	1.5	0.0	N/A	N/A
+ΔI Gain	2.3	2.19	N/A	N/A
f(ΔI) Penalty Dead-band	-46, to +2	-50, to +2	N/A	N/A

Overpower ΔT Reactor Trip

Functional Unit 6	*Trip Setpoint*		*Allowable Value*	
	Current Value[(a)]	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
K_4	1.10	N/C	0.4	0.96
K_6	0.00232	0.0016	N/A	N/A
T''	≤574.2°F	≤577.2°F	N/A	N/A

[(a)]*The information provided in this column represents the parameters provided to the NRC via FPL Letter L-95-131, Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint.*

N/A - Not Applicable *N/C - No Change*

Table 5.4-1 (Continued)
Summary Of The Reactor Protection System Setpoint Changes

TURKEY POINT TECHNICAL SPECIFICATION TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION CURRENT AND PROPOSED TRIP SETPOINTS				
Reactor Coolant Flow-Low				
Functional Unit 10	*Trip Setpoint*		*Allowable Value*	
	Current Value[(a)]	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
Footnote	90% TDF(**)	90% TDF(***)	88.8%	N/C
Steam Generator Water Level Low-Low				
Functional Units 11 and 12	*Trip Setpoint*		*Allowable Value*	
	Current Value	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
Setpoint	≥10.0	N/C	≥8.9	≥8.15

[(a)]*The information provided in this column represents the parameters provided to the NRC via FPL Letter L-95-131, Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint.*

N/A - Not Applicable *N/C - No Change*

** Thermal Design Flow = 89,500 gpm

*** Thermal Design Flow = 85,000 gpm

Table 5.4-2
Summary Of The Engineered Safety
Features Actuation System Setpoint Changes

TURKEY POINT TECHNICAL SPECIFICATIONS TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS CURRENT AND PROPOSED TRIP SETPOINTS				
Steam Generator Water Level Low - Low				
Functional Unit 6.b	*Trip Setpoint*		*Allowable Value*	
	Current Value[(a)]	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
Setpoint	≥10.0%	N/C	≥8.9%	≥8.15%

Table 5.4-2 (Continued)
Summary Of The Engineered Safety
Features Actuation System Setpoint Changes

TURKEY POINT TECHNICAL SPECIFICATIONS TABLE 3.3-3 (Continued) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS CURRENT AND PROPOSED TRIP SETPOINTS				
High Steam Line Flow				
Functional Units 1.f and 4.d	*Trip Setpoint*		*Allowable Value*	
	Current Value[(a)]	*Proposed Value*	*Current Value*[(a)]	*Proposed Value*
Setpoint	≤40%*	≤40%**	≤42.6%#	≤44%##

* ≤ A function defined as follows: A ΔP corresponding to 40% Steam Flow at 0% load increasing linearly from 20% load to a value corresponding to <u>120%</u> Steam Flow at full load.

** ≤ A function defined as follows: A ΔP corresponding to 40% Steam Flow at 0% load increasing linearly from 20% load to a value corresponding to <u>114%</u> Steam Flow at full load.

\# ≤ A function defined as follows: A ΔP corresponding to <u>42.6%</u> Steam Flow at 0% load increasing linearly from 20% load to a value corresponding to <u>122.6%</u> Steam Flow at full load.

\#\# ≤ A function defined as follows: A ΔP corresponding to <u>44%</u> Steam Flow at 0% load increasing linearly from 20% load to a value corresponding to <u>116.5%</u> Steam Flow at full load.

[(a)]The information provided in this column represents the parameters provided to the NRC via FPL Letter L-95-131, Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint.

N/A - Not Applicable *N/C - No Change*

5.5 NSSS/BALANCE-OF-PLANT (BOP) INTERFACE SYSTEMS

5.5.1 Auxiliary Feedwater System/Condensate Storage Tank

5.5.1.1 Introduction

The Auxiliary Feedwater (AFW) System is evaluated to ensure that the current AFW flows and starting times are acceptable to support design basis plant transients at plant uprate conditions. In addition, the Condensate Storage Tanks (CSTs) are evaluated to ensure that their capacity is adequate to supply the AFW System at plant uprate conditions.

The AFW System is a Safety Related system, and is shared between Units 3 and 4. The AFW System supplies feedwater to the steam generators (SGs) during transients when normal feedwater sources are not available.

The following are the design basis transients that establish the minimum/maximum AFW System requirements:

- SBLOCA in combination with a LOOP (both units),
- LOOP to both units,
- LOMF (single unit), and
- MSLB.

The most limiting plant transients requiring minimum AFW flow are a LOOP event or a SBLOCA concurrent with a LOOP. The worst case transient for a single unit requiring minimum AFW flow is a LOMF. Maximum AFW flow to any one of the SGs is determined using the maximum flow assumed for a MSLB event.

In addition to flow requirements, the AFW System is required to begin delivering water to the SGs within a set delivery time at a pressure equal to or greater than the set pressure of the lowest set SG safety valve, plus 3% accumulation pressure.

A minimum CST volume for the AFW System is required for accident mitigation and subsequent cooldown of the plant to the RHR System initiation conditions.

5.5.1.2 Description of Analyses/Evaluation Performed

The evaluation consisted of comparing the AFW System minimum/maximum flow inputs used in the uprate core response and mass and energy release analyses of the LOMF, LOOP, SBLOCA (with LOOP), and MSLB events to the calculated expected flows to determine whether uprate affects the AFW System's capability in supplying feedwater to the SGs.

In addition, existing CST and AFW System component design parameters and Technical Specification requirements were reviewed to determine if the existing CST volume and AFW pressure and delivery are adequate at the uprate condition.

The evaluation identified that the flows utilized in the core response analyses for the LOMF, LOOP, and SBLOCA with LOOP events are less than or equal to the minimum flows calculated. It was also found that the flow utilized in the mass and energy analysis for a MSLB event is equal to the calculated maximum flow.

It was also determined that the AFW System components have sufficient margin to provide the required flow and pressure, and that the stroke time of the motor operated AFW steam supply isolation valves provided adequate time in delivering AFW flow to the SGs.

The CST minimum usable volume which is required to support the design basis that the plant be maintained at hot standby for 15 hours followed by a four-hour cooldown to RHR cut-in temperature (350°F) was also determined for the uprated power. This minimum usable volume is 199,100 gallons.

5.5.1.3 Conclusions

The existing AFW System and CST are capable of providing the required AFW flow and volume needed to support the design basis transients at plant thermal uprate conditions.

5.5.2 Component Cooling Water System

5.5.2.1 Introduction

The Component Cooling Water System (CCWS) is an intermediate closed-loop cooling system between NSSS equipment which potentially process radioactive fluids, and the plant ultimate heat sink. The primary CCWS function is heat removal which is accomplished by the continuous recirculation of flow within two main cooling headers. The CCWS is required to operate for all normal and abnormal plant operating conditions. Ultimate heat sink cooling flow is provided by the Intake Cooling Water (ICW) System which delivers flow to the tube side of the CCWS heat exchangers.

In addition to traditional system cooling requirements, the Turkey Point Units 3 and 4 CCWS also provides cooling to the containment building atmosphere. Separate sets of containment coolers (air-to-water design) are used to perform this function for normal plant operations and post-accident operating conditions. The post-accident function is provided by Emergency Containment Coolers (ECCs), of which three are provided. As part of uprate, design changes will be made to assure no more than 2 ECCs will automatically start in response to an accident.

CCWS heat removal capability will change depending on various operational factors. In general, system heat removal capability becomes more restrictive with the following operating condition changes:

- Higher CCWS cooling heat loads
- Higher CCWS heat exchanger tube plugging
- Higher CCWS heat exchanger fouling
- Lower ICW flow to the CCWS heat exchanger
- Higher ICW temperature to the CCWS heat exchanger

The evaluation of the CCWS for the Turkey Point uprated condition considered the operational modes of Power Operation (including startup), Residual Heat Removal (RHR) cooldown (including cold shutdown and refueling) and Post-Accident (injection and recirculation).

5.5.2.2 Input Parameters and Assumptions

In general, the thermal performance of the CCWS (in conjunction with the ICW System) was evaluated or analyzed under "worst-case" operating conditions to ensure conservative operating performance. Of the changes in RCS operating conditions due to uprating, only the increase in reactor core power level has a significant effect on CCWS thermal performance capability. Specifically, higher core power levels will increase RCS and Spent Fuel Pool (SFP) decay heat loads which must be removed during all operating configurations.

Provided below are the critical CCWS heat removal functions that were reviewed as part of this project:

- Accommodate expected Power Operation configuration heat loads while maintaining supply temperature to within the existing maximum normal limit.

- Support the RHR System relative to its RCS decay heat removal function. This capability is discussed in the RHR System section of this licensing report (See Section 5.2.4).

- Maintain operating temperatures during post-accident configurations within NSSS equipment cooling requirements. Peak system operating temperatures occur during post-accident operations due to elevated containment conditions and unrestricted heat rejection into the CCWS.

- The adequacy of the CCWS piping network at projected operating temperatures.

In general, input parameters were chosen to yield conservative analysis results based on allowable variations. For example, containment integrity analyses inputs were established based on minimum heat transfer conditions. System thermal analyses, however, maximized heat input into the CCWS in order to establish maximum operating temperatures.

5.5.2.3 Description of Analyses/Evaluation

For the Power Operation and RHR Cooldown configurations, thermal performance calculations were performed at steady-state plant operating conditions using standard water-to-water heat exchanger heat transfer equations and generalized heat transfer methodology. During postulated design basis events, the CCWS major heat loads (ECCs and the RHR heat exchangers) are variable in nature and vary significantly with containment operating conditions. Therefore Containment Integrity analysis methodology was used to conservatively calculate limiting CCWS and ICW System post-accident operating conditions. Input parameters to these analyses were modified to maximize CCWS and ICW Systems' operating temperatures.

As part of this project, maximum expected CCWS operating temperatures were conservatively defined/calculated for use in the evaluation of system piping stress analyses. For this work, maximum expected CCWS inlet temperatures and minimum expected component flows were generally used to calculate worst-case component outlet temperatures.

5.5.2.4 Acceptance Criteria

For the Power Operation configuration, the capability to maintain the CCWS supply temperature at or below the maximum allowable temperature. Credit can be taken for operator actions to reduce variable CCWS heat loads, if required.

For CCWS performance capability during post-accident operation, the following are the most critical CCWS operating temperatures:

- CCWS Heat Exchanger Shell Side Inlet (return) Temperature
- CCWS Heat Exchanger Shell Side Outlet (supply) Temperature
- ECC CCWS Outlet Temperature
- RHR Heat Exchanger CCWS Outlet Temperature

For the CCWS heat exchanger return temperature, verification that the CCWS outlet temperature remains at or below the point where two phase flow can occur and within CCWS pump Net Positive Suction Head (NPSH) limitations is required. This ensures that single phase (i.e., liquid) flow conditions continue to occur and that the CCWS pump would continue to operate.

For CCWS supply temperature, verification that it remains within analyzed limits is needed to ensure that equipment cooled by the system remains operable.

For the ECC and RHR heat exchangers, verification that the CCWS outlet temperature remains at or below the point where two phase flow can occur. With single phase (i.e., liquid) flow conditions, continuous heat removal would occur.

With respect to CCWS piping structural integrity, a set of maximum CCWS operating temperatures were defined for use in CCWS piping stress reanalyses. This criterion ensures overall system piping availability/operability under worst-case operating conditions.

5.5.2.5 Results

5.5.2.5.1 Power Operation

For Power Operation it was found that the maximum CCWS supply temperature could be maintained at or below its maximum limit with only two CCWS heat exchangers in service. Operator actions may be necessary to restrict system heat loads during limiting ICW System operating conditions (elevated ICW inlet temperature, elevated tube fouling, etc.). An example of such an operator action is the reduction of non-essential heat loads.

5.5.2.5.2 Post Accident

The Large Break LOCA and the MSLB inside containment accidents result in the highest heat input condition to the CCWS. The CCWS thermal response during both the injection and recirculation modes was considered. Calculations showed that when all three ECCs are allowed to operate, CCWS operating temperatures can be above its maximum allowable limits during injection and/or recirculation. When only one or two ECCs are assumed to start, CCWS acceptance criteria are met.

The piping stress analyses results showed that CCWS operating temperatures are within maximum allowable values.

5.5.2.6 Conclusions

Based on the CCWS thermal analyses and associated component evaluations performed at uprated condition, it is concluded that the CCWS is capable of performing its intended cooling function: For post accident conditions, this is based on allowing no more than two ECCs to automatically start on an "SI" signal.

5.5.3 Normal Containment Cooling System

5.5.3.1 Introduction

The Normal Containment Cooling System (NCCS) is not safety related and has no impact on the plant licensing basis.

During normal plant operation, the NCCS removes the heat lost from all equipment and piping in containment, and maintains containment bulk ambient temperature at or below a normal ambient temperature of 120°F. The NCCS also provides sufficient air mixing and circulation throughout all containment areas to permit maintenance and/or refueling operations after reactor shutdown.

The NCCS is comprised of, the Normal Containment Coolers (NCCS) and the Control Rod Drive Mechanism (CRDM) Coolers. The NCCS consists of four cooling units and associated ductwork. The CRDM, consisting of two cooling units and associated ductwork, supplements the NCCS, and can be used to remove heat from the reactor vessel head during natural circulation cooldown.

The required cooling coil cleanliness is maintained by regular cleaning, inspection and preventive maintenance practices.

5.5.3.2 Description of Analyses/Evaluation Performed

The NCCS evaluation consisted of comparing the total heat load in containment due to uprate with the total heat removal load provided by the NCCS and CRDM cooling units during normal operation and assuring that the NCCS can maintain the containment operating temperature at or below 120°F. The expected increase in the containment total heat load was calculated to be less than the heat removal capacity provided by the number of cooling units currently operating at Units 3 and 4. Due to the current margin in heat load removal capability, and the minimal expected increase in total heat load with uprate, operating temperatures inside containment are expected to increase no more than 2°F above current levels.

Regardless of the number of cooling units currently used in plant operation, normal operating temperatures in containment have not reached the 120°F limit. Representative operating temperatures recorded in containment for May through July 1993 range approximately between 101°F and 117°F for Unit 3 and 100°F and 117°F for Unit 4. These temperature ranges are reasonably conservative as they include full-power operation in summer months.

In addition, in the unlikely event that containment operating temperature were to exceed 120°F, but not 125°F, the Technical Specifications allow operation to continue for a cumulative 336 hours per year at a temperature not to exceed 125°F.

5.5.3.3 Conclusions

The increase in containment total heat load and operating temperature due to uprate will not impact the capability of the NCCS to maintain containment operating temperature below the design basis of 120°F. The design capacity for NCCS and CRDM cooling units exceeds the heat load expected at uprated conditions.

5.5.4 Emergency Containment Cooling and Filtering Systems

5.5.4.1 Introduction

The safety-related Emergency Containment Cooling and Filtering Systems (ECCFS) is used in conjunction with the Containment Spray System (CSS) to provide adequate heat removal capability in containment following a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). In addition, the ECCFS is used to provide adequate air recirculation capability in containment following a LOCA to reduce the iodine concentration and prevent hydrogen concentration buildup. The ECCFS is comprised of two systems; the Cooling System and the Filtering System.

The ECCFS is comprised of three Emergency Containment Coolers (ECCs) only one of which is required to remove heat from containment atmosphere to keep the containment temperature and pressure from exceeding design limits. In addition, the Cooling System provides air recirculation for hydrogen dispersion following a LOCA, to impede hydrogen accumulations from reaching flammable or explosive concentrations in the containment. The Cooling System's minimum heat removal capability is modeled in the Containment Integrity analysis as a function of temperature and the performance of each ECC is used as an input in determining the LOCA long-term pressure and temperature transient effects. The Cooling System's maximum heat removal capability is modeled in the Component Cooling Water System (CCWS) post-accident thermal analysis to limit CCW temperatures.

The ECCFS is also comprised of three Emergency Containment Filters (ECFs), any two of which must operate following a LOCA with failed fuel to remove free iodine from the containment's atmosphere. Each ECF contains a spray system, which is used to remove decay heat from the charcoal filters in the event of a loss of forced air flow through the charcoal filter. The Filtering System's iodine reduction capability is modeled in the Environmental Consequences of a Loss-of-Coolant Accident analysis (e.g., offsite dose analysis).

5.5.4.2 Description of Analyses/Evaluation Performed

The ECCFS evaluation consisted of determining if uprate affects the ability of the ECCFS components based on the Containment Integrity Analysis, the Hydrogen Concentration Analyses, and the Offsite Dose Analysis results. In addition, the effect of a 3% increase in CSS flow temperature due to uprate on the heat removal capability of the ECF spray system was determined using existing limiting CSS flow parameters.

Uprate was determined not to affect the design of the ECCs or the equipment associated with them as the Containment Integrity, Offsite Dose, and Hydrogen Concentration Analyses yield acceptable results that do not impact the existing design of the ECCFS components.

The uprate ECC CCW inlet and outlet temperatures are bounded by the existing design temperatures. As such, changes in CCW flow parameters due to uprate will not affect the ECC equipment design.

The Hydrogen Generation Analysis for the Turkey Point Units 3 and 4 uprating is discussed in Section 3.6.2.

The Offsite Dose analysis for the plant thermal uprate is based upon the existing ECF flow rates and filter efficiencies. Therefore, there is no impact on the ECF's iodine reduction capability at the uprated power level.

In utilizing the existing CSS inlet flow conditions at uprate, the CSS flow ΔT was determined not to increase. However, the maximum inlet and outlet temperature will increase by approximately 5°F. Based upon a maximum inlet flow temperature of 205°F, the charcoal filters were found to be maintained at less than the design basis limit of 250°F.

In addition, the time the ECF spray systems are required for post-accident conditions were found to not be impacted by plant uprate because the pre-uprate analysis is based upon a core power of 2300 MWt (plus 2%).

5.5.4.3 Conclusions

The plant thermal uprate to a core power of 2300 MWt (plus 2%) will not impact the capability of the ECCFS to provide both adequate heat removal capability following an MHA, and adequate air recirculation to reduce the iodine concentration and provide hydrogen concentration control following a LOCA.

This conclusion is supported by the uprate Containment Integrity, Hydrogen Concentration, and Offsite Dose analyses which utilize existing ECCFS component data, as documented above, and yield results within existing design limits.

5.5.5 Spent Fuel Pool Cooling System

5.5.5.1 Introduction

The Spent Fuel Pool (SFP) Cooling System removes decay heat from the spent fuel assemblies stored in the SFP during plant operations and refueling. A small portion of the cooling flow can be diverted through a demineralizer and filters for purification of the water. Surface debris in the SFP is removed via two skimmers, a skimmer pump, and associated filters. Each Unit's spent fuel pool cooling loop consists of pumps, heat exchanger, filters, demineralizer, piping, and associated valves and instrumentation. The pump draws water from the SFP, circulates it through the heat exchanger where it is cooled by the Component Cooling Water (CCW) System.

The SFP cooling system is designed to maintain its cooling function during and after a seismic event, and to structurally withstand a design temperature of 212°F. The SFP is designed to withstand stresses associated with a steady-state gradient of 150°F.

With the installation of high density spent fuel storage racks, the SFP cooling system was reevaluated to determine the effect on the system of increasing the spent fuel storage capacity. The high density fuel storage racks increased the pool capacity from 4 2/3 cores to 9 cores (Note: the evaluation assumed 1413 assemblies which is 9 more assemblies than the actual maximum storage capacity of 1404 assemblies). This expansion of the spent fuel storage in the pool increased the decay heat load for each pool and the pool peak transient water temperature after refueling to less than 141°F. With a freshly discharged core, plus the heat load from the previously discharged fuel (i.e., 7 1/2 cores), the pool water temperature is maintained less than 180°F.

5.5.5.2 Description of Analyses/Evaluation Performed

The Thermal Uprate will increase the core power level from 2200 MWt to 2300 MWt. Since the decay heat rate of the spent fuel is a function of the core power level, the SFP cooling heat load will increase. This increase will result in higher heat loads transferred to the CCW system and increased operating temperatures in the spent fuel pool. The thermal power uprate is not expected to impact the impurity level in the spent fuel pool and the design of the purification loop will not be impacted.

The SFP cooling was evaluated at the uprated power level to determine the impact on the SFP heat load and resultant maximum bulk temperature. The following cases consistent with the UFSAR Appendix 14D and SRP guidelines were evaluated:

Case 1 Normal Refueling

1/2 core offload at 150 hours after shutdown

Case 2 Normal Operation

1/2 core offload at 36 days following shutdown

Case 3 Abnormal Operation with SFP Cooling

Full core offload at 150 hours following a forced shutdown with 1/2 core recently offloaded (36 days after a normal refueling shutdown)

Case 4 Abnormal Operation without SFP Cooling

Full core offload at 150 hours following a forced shutdown with 1/2 core recently offloaded (36 days after a normal refueling shutdown)

For this case the makeup rate to replace SFP inventory due to boil off should also be determined.

Based on the results of the evaluation, the impact of the uprated power level is as follows:

Case 1 Normal Refueling

The maximum expected SFP heat load and temperature for a 1/2 core offload at 150 hours after shutdown is 16.6 MBTU/HR and 147°F.

Case 2 Normal Operation

The maximum expected SFP heat load and temperature for a 1/2 core offload at 36 days following shutdown is 10 MBTU/HR and 130°F.

Case 3 Abnormal Operation with SFP Cooling

The maximum expected SFP heat load and temperature for a full core offload at 150 hours following forced shutdown with 1/2 core recently offloaded (36 days after a normal refueling shutdown) is 35.5 MBTU/HR and 194.5°F. The time to reach the maximum steady state temperature with SFP cooling is 24 hours.

Case 4 Abnormal Operation without SFP Cooling

The maximum expected SFP heat load and temperature for a full core offload at 150 hours following a forced shutdown with 1/2 core recently offloaded (36 days after shutdown) is 35.5 MBTU/HR and 212°F. The time to reach boiling with no SFP cooling is 4.5 hours. The maximum boil off (makeup) rate at 212°F is 76.3 GPM.

5.5.5.3 Conclusions

The existing SFP cooling will be adequate for the uprated conditions. The maximum expected temperature for a 1/2 core normal refueling is 147°F which is below the steady-state gradient design temperature of 150°F. The maximum temperature was calculated based on conservative decay heat loads, rapid core offload, maximum cooling water temperature, and a 1/2 core offload. The decay heat load evaluation indicates that the temperature would remain above 140°F for approximately 150 hours. Also experience from previous refuelings and data taken during the Unit 4 1994 outage, demonstrate that the expected temperature for a full core offload in the SFP will be below that calculated.

For the abnormal case of a full core offload following a recent normal refueling the maximum temperature calculated is 194.5°F with SFP cooling. The SFP cooling loop is designed to remain functional during and following a seismic event, and structurally withstand a design temperature of 212°F. With a complete loss of SFP cooling, the temperature will reach boiling (212°F) in about 4.5 hours. The makeup rate to replace water loss due to boiling is approximately 76.3 gpm. There is still sufficient time to provide makeup at an available makeup rate of 100 gpm to maintain the SFP inventory.

5.6 TURBINE-GENERATOR SYSTEMS

The Turkey Point turbine generator system designs have been evaluated to determine their operability under uprated conditions. The following provides a summary of each system's acceptability of performance under the proposed uprated conditions.

5.6.1 Component Evaluation

5.6.1.1 Turbine

The turbine has been evaluated for areas such as increased steam flow and variation in pressure, and generator heat balance. The turbine meets Westinghouse acceptance for continuous service at the total NSSS power of 2308 MWt .

5.6.1.2 Moisture Separator-Reheater (MSR)

It is expected that the current MSR will meet, or exceed, the requirements for the new heat balance for 2308 MWt.

5.6.1.3 Generator

The proposed uprate in turbine input power to the generator is within the limits of the generator's capability curve. Westinghouse has reviewed the Unit 4 generator that was rewound by ABB and since FPL has elected to operate the generator within the original capability curves, no modifications are required and Unit 4 is expected to perform the same as Unit 3.

5.6.1.4 Exciter and Voltage Regulator

The evaluation for the exciter and voltage regulator confirm that no modifications are required and that they can be operated at the uprated conditions.

5.6.1.5 Coolers

The lube oil coolers, generator seal oil cooler (hydrogen side), exciter air cooler, and hydrogen cooler have been evaluated and no modifications are required for operation at the proposed uprated conditions.

5.6.1.6 Miscellaneous Systems

The turbine control system, gland seal system, gland steam leakoff piping, cylinder heating steam system, valve leakoff piping, gland condenser, lubrication oil system, and rotor turning gear have been reviewed and evaluated and no modifications are required for acceptable operation at the proposed uprated condition.

5.6.1.7 Conclusions

In this study of the turbine generator systems for the uprating, a review was made of the following areas: the moisture separator-reheater, generator, exciter, voltage regulator, coolers (lube oil, generator seal oil, exciter air, and hydrogen), turbine control system, gland seal system, lubrication oil system, and the rotor turning gear. The basis for this evaluation was a review of the expected design conditions at the uprated power level. These conditions were compared to the applicable design criteria to determine the acceptability of operation at the higher power level. Previous modification records from both Westinghouse and FPL were checked to ensure that the latest plant conditions were evaluated. In cases where design margin was minimal, plant operating data was also considered to determine whether the component could be approved for uprating.

The study results show that all the turbine generator systems and turbine auxiliaries reviewed meet the design criteria for the 2308 MWt uprating. It is therefore acceptable from a systems viewpoint for the plant to operate at the uprated power level.

5.7 Conclusions (NSSS and Turbine Generator (TG) Systems Review)

The evaluations discussed above concluded that the design requirements of the NSSS fluid systems, Control and protection systems, TG systems and NSSS/BOP interface systems are met for the Uprating and associated primary temperature conditions.

CHAPTER 6

BALANCE OF PLANT (BOP) EVALUATIONS

6.0 BOP EVALUATIONS

6.1 INTRODUCTION

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 10, that applies to power uprating.

The power conversion systems were designed to utilize the energy available from the nuclear steam supply system. The original system and equipment sizing was based on an NSSS power rating of 2208 MWt and a steam flow of 9.60 x 10^6 lb_m/hr. Uprating will increase main steam flow to 10.061 x 10^6 lb_m/hr or approximately a 5% increase.

The system operating and design pressures and temperatures for uprated conditions were developed by preparing new heat balances to reconstitute a baseline and describe uprate conditions.

6.2 BOP SYSTEMS

6.2.1 Main Steam System

The Main Steam (MS) System from the steam generators up to and including the Main Steam Isolation Valve (MSIV) assemblies are safety related. The MSIV assemblies include the Main Steam Check Valves (MSCVs) and Main Steam Bypass Valves (MSBVs).

The Main Steam (MS) system design including the main steam isolation valve assemblies and main steam safety valves (MSSVs) were evaluated to ensure that system and component capabilities bound the main steam conditions at the 2308 MWt uprated power rating. The atmospheric dump valves and the condenser dump valves are discussed in Section 6.2.2.

The main steam design conditions of 1085 psig and 600°F remain unchanged and bound all predicted operating conditions for both the system and components. At 2308 MWt, the predicted main steam flow is 10,061,000 lb/hr, an increase of approximately 5% over the original Westinghouse maximum guaranteed steam flow of 9,600,000 lb/hr . The predicted uprate main steam flows are 0.2 % less than the original maximum calculated conditions. The changes to the predicted operating pressures and temperatures at the uprate power conditions have no negative effect on the system piping or design.

The predicted increase in the main steam operating flow was evaluated for increased erosion/corrosion concerns. Because of the small increase in the piping velocities associated with the uprate, the E/C impact will be small. The plant E/C program will continue to monitor for material degradation.

Four MSSVs are located outside containment on each of the three main steam lines to protect the steam generators and MS piping from over-pressure. The safety valves discharge to atmosphere are designed and manufactured in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Re-analysis of the Loss of External Load Transient Analysis (UFSAR Section 14.1.10) at the uprated conditions confirmed that the existing MSSV setpoints and capacities were adequate at the uprate power level. Other non-LOCA events that could potentially impact the design Steam Generator Pressure criterion were also reviewed (e.g., UFSAR Sections 14.11, Loss of Normal Feedwater; UFSAR 14.1.12, Loss of AC Power: UFSAR 14.1.9, Loss of Reactor Coolant Flow (Locked Rotor, Partial Loss of Flow and Complete Loss of Flow)). A setpoint tolerance of ± 3% was determined to be acceptable and all safety margins are met for the uprated power level.

MSSV discharge pipe backpressure will be higher at the uprated conditions requiring a modification to the MSSV discharge piping to ensure adequate margin at uprate.

The MSIV assemblies provide safety related isolation capability for the steam generators for Main Steam Line Breaks (MSLBs) and Steam Generator Tube Ruptures (SGTRs) events. One valve assembly is provided outside containment for each main steam line from the steam generators. Each valve assembly consists of a swing disc held open against flow by a pneumatic cylinder and a check valve downstream to stop reverse flow from the other two steam generators in the event of a steam break up-stream of the isolation valve.

The MSIVs are maintained closed by the Instrument Air System. On Unit 3, a safety related nitrogen supply subsystem functions as a backup to the Instrument Air System. On Unit 4, safety related air accumulators are provided to perform this backup function. The valve assemblies were evaluated for the rapid closure conditions associated with a postulated pipe break. Based on a review of the existing design reports, the MSIV and MSCV capabilities are acceptable for operation and transients at the uprated power level.

6.2.2 Steam Dump System

The Steam Dump System consists of four condenser dump valves (CDVs) on a line from the Main Steam (MS) System which dump MS to the main condenser as necessary to accommodate a reactor trip with turbine trip and three atmospheric dump valves (ADVs), one on each MS line upstream of the Main Steam Isolation Valve (MSIV).

For the uprating, the CDVs are capable of passing the required 26% and 27% of the uprate full-load MS flow at low T_{AVG} and high T_{AVG} operation, respectively. The ADVs provide for plant cooldown when the main condenser is unavailable. Two of the three ADVs will be capable of passing 10% of the rated steam flow at no load pressure and each ADV is required to pass 10% of its respective steam generator rated steam flow at 775 psia.

Additionally, the predicted MS pressure, temperature, and velocity at uprate will be below the steam dump system and component design.

6.2.3 Condensate and Feedwater System

The Condensate and Feedwater System automatically maintains the steam generator water level during steady state and transient operations. The systems do not perform any safety related functions, except for the feedwater isolation valves and those portions of the feedwater system downstream of the isolation valves to the steam generators.

All of the system operating conditions are bounded by the existing design conditions. The condensate/feedwater system temperatures will increase slightly at the uprate conditions. The operating pressures will decrease slightly at uprate due to condensate/feedwater pump head characteristics and increased pressure drop at increased flow rates.

The total Condensate and Feedwater System resistance was evaluated for the higher flow rates at the uprate power level. The steam generator pressure remains approximately the same as experienced with the existing power level. Based on the system pressure drop and feedwater control valve capability at uprated conditions, the existing pumps have sufficient head to overcome the increased total system resistance with two condensate and two feedwater pumps in operation at the uprated condition. This is the same pump alignment used at the existing power level.

The net positive suction head (NPSH) available at the suction of the feedwater and condensate pumps is adequate at the uprated conditions.

The effect of the increased condensate/feedwater flows associated with uprate is not expected to alter the E/C rates appreciably as the velocity increases. The existing Erosion/Corrosion monitoring program will be continued to ensure that this conclusion is correct.

6.2.4 Steam Generator Blowdown System

The Steam Generator Blowdown (SGBD) System does not perform a safety-related function, except for steam generator isolation and has no impact on the plant licensing bases. The SGBD System is used in conjunction with the chemical feed system to control the chemical composition of the steam generator feedwater within allowable parameters as specified by generator manufacturer. The system also controls the build-up of solids in the steam generator water. The evaluation consisted of comparing the feedwater system design parameters at uprate and the blowdown flowrates to the existing system and component design parameters.

The SGBD System is sized to provide adequate capacity to maintain steam generator secondary side water chemistry under normal conditions, and to recover chemistry to within allowable limits for expected plant transient conditions. The steam generator design conditions do not change as a result of the uprate and therefore the SGBD System design conditions will also not change. Similarly, the flash tank and the downstream piping design conditions are still bounded by the existing design. Since none of the flow design parameters have changed significantly, the uprate will have no effect on the

SGBD System. The potential for erosion/corrosion (E/C) will increase with the slight increase in blowdown flowrate and velocity due to the uprate. However, design E/C limits are not exceeded.

6.2.5 Extraction Steam System

The Extraction Steam (ES) System contains piping and valves that transport steam extracted from various stages of the main turbine to the shell-side of the Feedwater (FW) heaters.

Extraction steam temperatures and pressures predicted at uprate were determined to be bounded by the ES piping design. Additionally, the performance of the non-return valves serving the Nos. 3, 4, 5, and 6 FW heaters are not impacted by uprate.

The extraction steam flows at uprate will be slightly higher but are bounded by the Extraction Steam (ES) piping design, and are not expected to exceed erosion/corrosion rate limits.

6.2.6 Circulating Water System

The Circulating Water (CW) System is not safety related and has no impact on the plant licensing basis.

The CW System supplies the unit's two-shell condenser with cooling water. The CW System was evaluated to ensure its capability to maintain the condenser pressure below maximum turbine backpressure limits/turbine trip setpoint.

The CW System outlet temperature is expected to increase less than 1°F at uprate, however, the condenser has sufficient margin to maintain turbine backpressure below the maximum limits/turbine trip setpoint. The environmental impact on the canal system associated with the CW and ICW System's heat duty increase is discussed further in Section 7.0.

6.2.7 Turbine Plant Cooling Water System

The Turbine Plant Cooling Water (TPCW) System is a closed-loop cooling water system and provides cooling water, during normal operation, to various non-safety related equipment coolers.

The heat absorbed by the TPCW System is rejected to the Intake Cooling Water (ICW) System, which, in turn, rejects the heat to the plant cooling canals. The TPCW System is isolated following a design basis accident.

The TPCW System heat load that is expected to increase because of uprate is that associated with the generator hydrogen coolers. However, the two TPCW heat exchangers are capable of providing the increased heat removal and therefore bound the uprate conditions.

6.2.8 Intake Cooling Water System

The Intake Cooling Water (ICW) System provides cooling water to the safety-related Component Cooling Water (CCW) and non-safety related Turbine Plant Cooling Water (TPCW) heat exchangers. The system is designed for removal of normal operating heat loads from the CCW and TPCW systems. In addition, the ICW System is also required to remove the heat load associated with the CCW System during accident conditions to support both reactor heat removal and containment heat removal requirements. The ICW System also provides lube water to the circulating water pumps located in the Intake Area.

The ICW System will experience higher heat loads during normal operation, resulting in slightly higher ICW discharge temperatures to the canal system. However, the existing ICW design basis is not exceeded, as is supported by the CCW analysis (See Section 5.5.2).

6.2.9 Instrumentation and Control Valves

Instrumentation and control valves in the following BOP systems were reviewed to determine whether any modifications to the existing design would be required as a result of the uprating:

- Main Steam
- Main Condenser
- Condenser Air Removal
- Circulating Water
- Condensate Polishing
- Condensate and Feedwater
- Extraction Steam
- Feedwater Heater, Moisture Separator and Reheater, Vents & Drains
- Steam Generator Blowdown
- Auxiliary Feedwater
- Intake Cooling Water
- Spent Fuel Pool Cooling
- Turbine Plant Cooling Water
- Instrument Air
- Primary Water Makeup
- Auxiliary Steam
- Containment Purge
- Heating Ventilation and Air Conditioning
- Emergency Containment Cooling and Filtering
- Normal Containment Cooling

A comparison between existing operating parameters, uprate operating parameters and instrument ranges were made to evaluate whether the instruments are suitable for uprate conditions. The existing design conditions were used as the basis of comparison with uprate operating conditions.

Control valves and plant instrumentation were reviewed to determine the effects of uprate on their design and current setpoints. Operating flows, pressures and temperatures at uprate were reviewed to determine whether they are enveloped by existing design conditions.

Based on the instrumentation and control valve review it was concluded that the difference between uprate and the current operating conditions are negligible and the instrumentation and control valves are acceptable for uprate conditions with only the condensate storage tank level, ECC start logic and the turbine first-stage pressure signal comparators requiring setpoint and calibration changes.

6.2.10 Electrical Systems

The Turkey Point Unit 3 and 4 station electrical systems, which include the 240 kV switchyard and the 4.16 kV, and the 480V Systems, are designed to provide a simple arrangement of buses requiring a minimum of switching to restore power to a bus in the event the normal supply to the bus is lost.

It was determined that the main generator is operating within the generator capability curve with ample margin to handle the uprated power output.

It was also determined that the electrical distribution system is able to accommodate the uprate requirements without exceeding equipment ratings.

6.2.11 Heating, Ventilation, and Air Conditioning

The following Heating, Ventilation, and Air Conditioning Systems were evaluated to ensure that they are capable of supporting the plant thermal uprate conditions:

- Control Room
- DC Equipment/Invertor Rooms
- Cable Spreading & Computer Equipment Rooms
- Radwaste Building
- Fuel Handling Building
- 480V Load Centers & 4.16 kV Switchgear Rooms
- Auxiliary Building
- Unit 4 Emergency Diesel Generator Building
- Electrical Equipment Room
- Containment Penetrations

During normal plant operation, these systems cool, heat, and ventilate plant areas to maintain a suitable environment for plant personnel and equipment, as appropriate. These systems will continue to maintain normal operating temperatures at or below their respective maximum normal operating temperatures. This is due to the negligible changes in the environments they serve, and/or the excess margin specified in their original design.

6.2.12 Miscellaneous Systems

Evaluations of the following systems were performed to determine the impact of the thermal uprate:

- Instrument Air
- Primary Makeup and Demineralizer Water
- Auxiliary Steam and Condensate Recovery
- Post Accident Sampling
- Containment Purge
- Feedwater Heaters
- Condensate Polishing System
- Heater, Moisture Separator and Reheater Drain System
- Main Condenser

Except for associated containment isolation features, these systems do not perform any safety related function and continue to function as intended at the uprated conditions.

6.3 BALANCE OF PLANT COMPONENTS

6.3.1 Accident Shielding - Vital Access

The shielding provided by the walls of cubicles that house components carrying post-accident recirculation fluids serve the dual purpose of limiting the doses received by plant personnel during any planned post-accident vital mission, as well as reducing the post-accident radiation exposure of safety related components located adjacent to these cubicles.

The equipment qualification and vital access dose estimates are based on the reactor equilibrium core inventory assuming full power operation, source term guidance relative to post-accident core releases as provided in TID 14844, and plant specific mitigation system design features.

Core uprate impacts the equilibrium core inventory and therefore the post-accident radiological source term. An additional factor that can impact the equilibrium core inventory is the expected fuel burnup. The impact of a core uprate from 2200 MWt to 2300 MWt, and the potential use of a 24 month nominal fuel cycle, on the post-accident radiological source terms, was evaluated, to assess the impact on post accident exposure rates in various plant areas, and to demonstrate the acceptability of the existing plant shielding.

The existing.post-LOCA source terms which are conservatively based on a core thermal power of 2200 MWt, were compared to the source terms associated with the uprated (2300 MWt plus 2%), 24 month extended burnup. The comparison included source term A (containment atmosphere, i.e., 100% noble gases, 25% halogens diluted in containment atmosphere), source term B (pressurized LOCA liquid, i.e., 100% noble gases, 50% halogens and 1% remainder diluted in the RCS volume) and source term C (depressurized LOCA liquid or sump water, i.e., 50% halogens and 1% remainder diluted in the sump water volume).

The approach taken was to perform a comparison of the current design basis source terms and the core uprate source terms and estimate a percentage impact due to the change, rather than develop actual dose rate estimates at various locations/times using the new core inventory. For the unshielded case, the impact on post-accident dose rates was estimated by comparing the total energy release rates as a function of time for Source Term A, B, and C. In order to demonstrate the acceptability of the existing post-accident shielding requirements, the source terms were weighted by the concrete reduction factors for each energy group, for 1 and 2 feet of concrete (typical shield thickness), thus providing a basis for comparison of the post-LOCA spectrum hardness of source terms A, B, and C (when unattenuated, or attenuated through 1 ft and 2 ft concrete) with respect to time, for the original design basis versus the uprated source terms.

The evaluation indicated that there is a close match between the source terms based on the uprated core/24 month fuel cycle and the current design basis source term. The existing shielding and post-accident dose rate estimates are adequate for the uprated conditions and any variances from existing calculated values are insignificant.

6.3.2 Equipment Qualification - Radiological

Equipment Qualification dose estimates are based on the reactor equilibrium core inventory assuming full power operation, source term guidance relative to post-accident core releases as provided in TID 14844, and plant specific mitigation system design features.

Core uprate impacts the equilibrium core inventory and therefore the post accident radiological source term. An additional factor that can impact the equilibrium core inventory is the expected fuel burnup. The impact of a core uprate from 2200 MWt to 2300 MWt, and the potential use of a 24 month nominal fuel cycle, on the post accident radiological source terms, was evaluated, to assess the impact on post accident radiation dose estimates in various plant areas, and to demonstrate the acceptability of the existing post accident equipment qualification dose requirements for safety related components.

The existing post-LOCA integrated gamma source terms which are conservatively based on a core thermal power of 2300 MWt, were compared to the integrated gamma source terms associated with the uprated (2300 MWt plus 2%), 24 month extended burnup cycle. The comparison included source term A (containment atmosphere, i.e., 100% noble gases, 25% halogens diluted in containment atmosphere), source term B (pressurized LOCA liquid, i.e., 100% noble gases, 50% halogens and 1% remainder diluted in the RCS volume) and source term C (depressurized LOCA liquid or sump water, i.e., 5% halogens and 1% remainder diluted in the sump water volume).

The approach taken was to perform a comparison of the current design basis integrated gamma source terms to the core uprate integrated gamma source terms, and estimate a percentage impact due to the change rather than develop actual integrated gamma dose estimates at various locations/times using the new core inventory.

The current "shielded" gamma design basis source terms are essentially equal in energy spectrum hardness (within 1%) to the corresponding extended burnup source terms. Consequently, the percentage impact on equipment qualification gamma doses is considered to be the same whether the controlling contribution is the result of an unshielded or a shielded source. The impact on post accident integrated gamma doses was therefore estimated by comparing the total unshielded integrated energy releases as a function of time for Source Term A, B, and C, between the design basis core versus the uprated, extended burnup core.

The impact on beta doses was assessed by a dose model consistent with the semi-infinite cloud model outlined in Regulatory Guide 1.4. A region of air with a very small exhaust rate (to prevent quiescence), was modelled. The appropriate fractions of core inventory associated with source terms A, B and C was "PUFFED" into this region and allowed to decay for 721 hours. Region volumes and densities were addressed for consistency, but the exact values were not considered important for this evaluation since the results were ratioed to develop the estimated impact. The calculated beta doses are for comparison purposes only and are not intended to replace the existing beta dose values that support equipment qualification.

The evaluation indicated that there is a close match between the integrated gamma source terms based on the uprated core/24 month fuel cycle and the design basis source term.

Since the gamma energy spectra for all three source terms are essentially equal in hardness (whether shielded or unshielded), throughout the entire accident, the gamma equipment qualification doses calculated based on design basis source terms are essentially unaffected by the uprate and use of extended burnup fuel, and is valid for unshielded as well as shielded components.

Therefore, it is concluded that the existing equipment qualification gamma and beta dose estimates are adequate for the uprated conditions and any variances from existing calculated values are insignificant and that the total integrated dose to safety related equipment from an accident remains unchanged from that previously evaluated.

6.3.3 Radwaste Systems

The liquid and gaseous radwaste activity is influenced by the reactor coolant activity which is a function of the core power level. This section discusses the impact of the uprate on the existing liquid and gaseous radwaste system for normal operational releases. The accident releases are discussed in Sections 3.2.14 and 3.2.15.

Potentially radioactive liquid waste from Units 3 and 4 chemistry laboratories, containment sumps, floor drains, showers, and miscellaneous sources are collected in waste hold up tanks. The liquid waste is processed through demineralizers and the effluent stored in the waste monitoring tanks. Laundry waste is normally segregated and sent to monitor tanks. Liquid waste in the monitoring tanks are released after sampling and analysis in accordance with Technical Specification 3/4.11.1. The effluent discharge is monitored by a radioactive liquid effluent monitor.

The activity of the steam generator blowdown discharge to the blowdown flash tank is monitored and the releases are sampled and analyzed in accordance with Technical Specification 3/4.11.

Radioactive and potentially radioactive gases from Units 3 and 4 Containment Buildings, Auxiliary Building, Spent Fuel Pool, Radwaste Building, and Laundry area are released via the monitored plant vent. Radioactive gases from the plant primary systems are stored in the gas decay tanks. The gases are held up to reduce the activity levels by radioactive decay prior to release. The gaseous waste are released after sampling and analysis in accordance with Technical Specification 3/4.11.2.

The limits placed on plant radioactive effluent release by 10 CFR 20 and 10 CFR 100 have been considered in the design and operating plans for the plant, with the objective to maintain release concentration at the site boundary below natural background activity and thus only a minute fraction of 10 CFR 20 limits. To achieve these objectives, the facility has been designed and is operated as follows:

1. Liquid wastes are collected in tanks and processed by the waste disposal demineralizer. Waste evaporators are also provided if necessary. The waste processes provided can reduce activity well below established limits and represent a design for reducing activity to the lowest practicable value.

2. Gaseous wastes are stored in decay tanks for natural decay. Gases will be released through the monitored plant vent, and at the site boundary the annual dose will not exceed the regulatory limits.

   The quantity of radioactivity contained in each decay tank is restricted to provide (a) assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem, and (b) assurance that the concentration of potentially explosive gas mixtures contained in the Gas Decay Tank System is maintained below the flammability limits of hydrogen and oxygen.

The existing design of the radwaste systems is based on the core power level of 2300 MWt. The uprate does not require changes to the existing design and/or operation of the radwaste systems. There is expected to be minimal impact on the frequency of and the amount of waste processed, however the radwaste process capability to meet the existing Technical Specification limits is not impacted . No Technical Specification changes are required.

Uprating to a core power level of 2300 MWt does not impact the ability of the radwaste systems to provide adequate processing and maintain the normal operational radioactive releases within regulatory limits.

6.3.4 Reference

1. TID 14844 entitled "Calculation of Distance Factors for Power and Test Reactor Sites", J. J. DiNuno, et. al. dated March 23, 1962.

6.4 ADDITIONAL BOP REVIEWS

6.4.1 High Energy Line Break

System operating parameters for uprate were evaluated against the system pressure and/or temperature parameters used in the existing plant bases to demonstrate the acceptability for High Energy Line Break (HELB) effects. The HELB review was conducted to evaluate the possible effects on the input to EQ analysis (pressure, temperature, and flooding), jet impingement forces on components and structures, and pipe rupture restraint reactions as a result of plant thermal uprate to 2308 MWt. For the Auxiliary and Standby Steam Generator Feedwater and Steam Generator Blowdown systems, the consequences of the dynamic effects of HELB were treated as independent of the system parameters, and dependent on the analysis of the potential targets.

The resulting conditions obtained (i.e., pressure, temperature, jet load, etc.), assuming the postulated failure of the affected piping systems, were acceptable at the uprate condition, if they were bounded by those conditions used in the existing design bases. The resulting conditions associated with the HELB were considered bounding if the internal pipe operating conditions used in the previous HELB analysis at the existing 2208 MWt rating were bounded by the same operating modes at the 2308 MWt uprate conditions.

The uprate review considered the consequences of postulated breaks outside and inside containment for the following high energy piping systems:

- Main Steam (MS) System,
- Main Feedwater (FW) System,
- Auxiliary Feedwater (AFW) System; consisting of the steam supply to the pump turbine and the AFW discharge,
- Standby Steam Generator Feedwater System,
- Steam Generator Blowdown (SGBD) System, and
- Chemical and Volume Control System (CVCS); consisting of the letdown and charging lines.

Core uprate will not change the temperature and pressure environment used as the basis for pipe break analysis. System operating parameters for uprate are bounded by the original (existing) 2208 MWt analyses and no additional analysis was required.

6.4.2 Piping and Supports

The purpose of the piping and support review is to evaluate piping systems for the effects resulting from thermal uprated conditions in order to demonstrate design basis compliance. Operation at the uprated conditions may increase piping stresses caused by slightly higher operating temperatures, pressures and flow rates. Additionally, pipe supports and equipment nozzles may be subjected to slightly increased loadings due to the thermal uprate condition.

The specific piping systems evaluated for thermal uprated conditions are as follows:

<u>Safety Related Piping Systems</u>
Auxiliary Feedwater Supply
Auxiliary Feedwater Pumps Steam Supply
Chemical and Volume Control
Component Cooling Water
Condensate Storage Tanks and Transfer Pumps
Containment Spray and Containment Emergency Filters
Feedwater
Intake Cooling Water
Main Steam
Pressurizer Safety and Relief Valve Piping
Primary Water and Demineralized Water
Safety Injection and Residual Heat Removal
Spent Fuel Pool Cooling
Steam Generator Blowdown
Waste Disposal - Liquid

<u>Non-Safety Related Piping Systems</u>
Circulating Water
Condensate
Extraction Steam
Feedwater Heater, Moisture Separator and Reheater (Vents, Drains and Relief Valves)
Turbine Plant Cooling Water

The piping and support review concluded that each piping system remains acceptable and continues to satisfy design basis requirements when considering the effects resulting from thermal uprated conditions. The evaluations also document that no piping or pipe support modifications are required as a result of the increased power level.

6.4.3 Structures

The effects on structures due to the thermal power uprate of Turkey Point Units 3 and 4, are reflected in changes to the loads transmitted from equipment, systems, and components, and from the normal operating and postulated accident environments. UFSAR Appendices 5B and 5A describe the design bases of the containment and the other Class I structures, including the loadings used in their design. The loadings associated with equipment, systems, and components and the normal and accident environments, which are influenced by thermal uprate, were evaluated in detail.

The loads which are the basis for the design of Class I structures are described in UFSAR Appendix 5A and are a subset of those described for the containment. These loads are the dead loads, pipe rupture loads, piping reactions, earthquake loads and wind loads. The pipe rupture loads and the piping reactions are affected by the thermal uprate and were evaluated.

The design basis pressure for the containment, 55 psig, is based on a LOCA and was compared to the results of the Containment Integrity Analysis (Section 3.5). The limiting case for the calculated containment peak LOCA pressure for the thermal uprate conditions is the Double-Ended Hot Leg (DEHL) break resulting in a pressure of 48.1 psig which is less than the containment design basis value.

Since the uprate calculated peak LOCA containment atmosphere temperature is below that calculated for existing conditions and the durations of the temperature transients are similar it is concluded that the design basis containment liner temperature and wall thermal gradients shown in UFSAR Figure 5.1-8 are not exceeded by core uprate.

The existing analysis of the consequences of the high energy line breaks (Section 6.4.1) is not changed by thermal uprate. These consequences include both the magnitude and types of loads (reactions, jet impingement, and whip) and the locations of the breaks. The pipe support reactions resulting from thermal uprate (Section 6.4.2) are acceptable and no significant changes to these loads were identified.

Therefore, the proposed thermal uprate will not adversely affect structures as reflected in changes to the loads transmitted from equipment, systems, and components, and from the normal operating and postulated accident pressure and temperature environments.

CHAPTER 7

ENVIRONMENTAL CONSIDERATIONS

7.0 ENVIRONMENTAL EVALUATION

This section discusses the need for the thermal power uprate and the potential impact the thermal power uprate will have on the environment. The onsite and offsite radiological and non-radiological environmental effects are evaluated.

Turkey Point Units 3 and 4 are currently licensed for a core power level of 2200 MWt and the proposed thermal power uprate will increase the licensed core power level to 2300 MWt which will result in an increase in electrical generation output of approximately 30 MWe per unit. Appendix B of the Turkey Point Units 3 and 4 operating licenses provide for changes in facility design and operation provided such changes do not involve an unreviewed environmental question. This section discusses the environmental evaluation of the impact of the thermal power uprate and documents that the thermal power uprate neither constitutes an unreviewed environmental question nor will have a significant impact on the quality of the human environment.

Environmental issues associated with the issuance of an operating license for both Turkey Point Units 3 and 4 were originally evaluated in the "Final Environmental Statement (FES) related to the Operation of the Turkey Point Plant" (Reference 1). A further evaluation of impacts was performed in connection with the proposed license amendments which recaptured the construction period for the operating license (Recapture Amendments) (Reference 2). The approval of the Recapture Amendments allows FPL to operate Turkey Point Units 3 and 4 for a full 40 year operating period (an additional 5.25 and 6 years, respectively, beyond the previously approved operating period). The NRC's Environmental Assessment and Finding of No Significant Impact (Reference 3) related to the operating license extension concluded that the proposed action will not have a significant effect on the quality of the human environment.

The environmental review conducted for the proposed thermal power uprate considered the need for the power uprate and the resulting environmental impact associated with it. This included considering the operating license and NPDES permit limits and the information contained in the FES and the evaluations associated with the Recapture Amendments. This evaluation included determining whether the power uprate would cause the plant to exceed discharge limits and NPDES permit conditions associated with the operation of the plant. In addition, a review of the recent Turkey Point Units 3 and 4 Annual Radioactive Effluent Release Reports was undertaken to evaluate whether a small increase in discharge amounts is acceptable. Slight increases in discharge amounts, if any, associated with the proposed thermal power uprate are acceptable, as annual discharges will continue to be a small percentage of the allowable limits and the FES estimates.

7.1 NEED FOR ACTION

The proposed action would increase the electrical output of each Turkey Point unit by approximately 30 MWe, and thus, would provide additional electric power to service commercial and domestic loads on the Florida Power and Light Company grid. The thermal power uprate is needed to accommodate the annual growth rate in the FPL service territory while avoiding major capital expenditures for new generating capacity. The thermal power uprate program will result in direct displacement of higher cost fossil fuel generation with lower cost nuclear fuel generation.

7.2 OFFSITE RADIATION EXPOSURE

Offsite radiation exposures from normal operation and accidents are assessed and documented in the Turkey Point Units 3 and 4 Updated Final Safety Analysis (UFSAR) with additional information contained in the FES and evaluation associated with the Recapture Amendments.

7.2.1 Normal Operation Exposure

The offsite radiation exposure from various pathways to the maximally exposed individual member of the general public has been evaluated for the proposed uprate.

Section V.D. of the FES projected doses and anticipated annual release of radioactive material as characterized in Table III-2 and III-3 resulting from radioactive materials released to the environment from routine operations of the two reactors. Title 10 CFR Part 50, Appendix I, which provides guidelines for meeting as low as reasonably achievable (ALARA) doses from the reactors, is incorporated in the Turkey Point Units 3 and 4 Technical Specifications and Offsite Dose Calculation Manual (ODCM).

The results of operating experience in effluent, offsite dose calculation results, and the radiological environmental monitoring program demonstrate the minimal radiological impact upon the general public from the operation of the two reactors.

The liquid effluent from the plant are discharged into a closed cooling canal system.

Gaseous waste from routine operations are collected, compressed, and stored in holdup tanks at the plant. The holdup tanks allow for the decay of short half-life radionuclides prior to release through high efficiency particulate absolute (HEPA) filters to remove particulate material.

Turkey Point Units 3 and 4 have consistently been operated well within the requirements of 10 CFR 50 Appendix I for all types of releases as documented in the Turkey Point Units 3 and 4 Annual Radioactive Effluent Release Reports.

The Turkey Point Units 3 and 4 Radiological Effluent Technical Specifications (RETS) are also in compliance with the goal of maintaining radiation exposure ALARA. The capability of the Turkey Point Units 3 and 4 to meet the required Effluent Technical Specifications and maintain radiation exposure ALARA, as analyzed in the FES and evaluations associated with the Recapture Amendments, will not be impacted by the thermal power uprate.

7.2.2 Accident Exposure

Offsite radiation exposures from postulated accidents are assessed and documented in Section 3.0, consistent with the analysis in the FES and the evaluation associated with the Recapture Amendments. The offsite doses for the exposure postulated under accident conditions remain within the guidelines of 10 CFR 100.

7.3 ONSITE RADIATION EXPOSURE AND RADIOACTIVE WASTE PRODUCTION

The thermal power uprate is not expected to increase the day-to-day radiation exposures encountered by plant workers since the in-plant radiation levels will not change significantly, with respect to the evaluations in the FES and the evaluations associated with the Recapture Amendments.

FPL has developed and implemented comprehensive ALARA programs at its nuclear power plants. Three types of waste are generated at Turkey Point Plant: gaseous, liquid, and solid. Each of these types of waste is discussed in Section 6.3.3 and below with respect to their impact on waste treatment.

The gaseous radwaste systems are designed to assure that the airborne release of such waste is maintained ALARA during normal plant operation. The RETS ensure that the equipment required to maintain the offsite doses ALARA will be operable and will be operated as required to maintain the releases ALARA.

The liquid waste treatment systems at Turkey Point Units 3 and 4 are designed to meet the ALARA goals. These systems are also subject to the RETS for assurance of operability.

Operation of Turkey Point Units 3 and 4 at the uprated power level may result in additional solid Low Level Radioactive Waste (LLRW) that will have to be shipped for disposal. However, the annual volume of LLRW is not expected to increase significantly. Additionally, Turkey Point's LLRW disposal volume is well below the median value for similar two unit pressurized water reactor (PWR) sites. Over the years, significant improvements have been made in the way that LLRW is handled and disposed. Turkey Point Plant also uses volume minimization techniques and other volume reduction processes to minimize the volume of LLRW for final disposal. These techniques should further minimize any impact power uprate might have on the generation of additional LLRW.

7.4 NON-RADIOLOGICAL EFFECTS

The FES (Reference 1) and the evaluations associated with the Recapture Amendments (References 2 and 3) assessed the non- radiological impacts of plant operation as a function of plant design features, relative loss of renewable resources, and relative loss or degradation of available habitat.
Environmental impacts associated with forty year operating licenses were originally evaluated in the FES. The FES and the evaluations associated with the Recapture Amendments concluded that, after weighing the environmental, economic, technical, and other benefits against environmental costs and

considering available alternatives, and subject to certain conditions, from the standpoint of environmental effects, the issuance of operating licenses for Turkey Point Units 3 and 4 was an acceptable action. These assessments, and the assumptions on which they were based, remain valid and are not impacted as a result of the thermal power uprate.

Protection of the environment is assured by compliance with permits issued by federal, state, and local agencies.

7.5 NATIONAL POLLUTANT DISCHARGE ELIMINATION SYSTEM (NPDES) PERMIT IMPACT

The Turkey Point Plant consists of two fossil fuel units (Units 1 and 2) and the two nuclear units (Units 3 and 4). The four units obtain their cooling water from and discharge to a closed cooling canal system. All water used at the plant is recycled within the closed canal system except station make-up which is purchased from the local municipal utility. The thermal loading on the canal from the four units is approximately 14 x 10^9 Btu/hr.

The Turkey Point Units 3 and 4 were licensed for an initial licensed power level of 2200 MWt with an ultimate thermal generating capacity identified in the Final Environmental Statement of 2300 MWt. There are no discharges to Biscayne Bay or Card Sound from the plant site and therefore the Turkey Point NPDES permit does not place any operating limits on either flow or temperature.

The heat duty increase associated with uprate is mainly associated with the Circulating Water System and will be approximately 440 x 10^6 Btu/hr. This represents a 4.4% increase over the present power level but is insignificant when compared to the heat load from all four units and the incident solar radiation heat gain to the canal.

For normal Circulating Water System operation, which includes the reduction in circulating water flow caused by the existing condenser tubes plugged, the maximum temperature increase expected as a result of the uprate between inlet and outlet will be approximately 0.7°F over existing plant operation. Therefore, the thermal power uprate of the Turkey Point Units 3 and 4 will have no adverse impacts on the environment or result in exceeding NPDES permit limits.

7.6 ALTERNATIVE TO THE PROPOSED ACTION

The principal alternative would be "no action" with respect to the requested amendments for the thermal power uprate. No action would not significantly reduce the environmental impact of plant operations, but would restrict operation of the Turkey Point facility to the currently licensed power level. No action would prevent the facility from generating the additional approximately 30 MWe for each Turkey Point unit that is needed for present and future system loads.

7.7 ALTERNATIVE USE OF RESOURCES

This action does not involve a significant increase in the use of resources not previously considered in the "Final Environmental Statement Related to the Operation of Turkey Point Plant," dated July 1972 (Reference 1) and the environmental evaluation performed to support the "Issuance of Amendments Re: Recapturing Construction Period in the License Term," dated April 20, 1994 (References 2 and 3).

7.8 SUMMARY OF ENVIRONMENTAL ANALYSIS

The radiological and non-radiological environmental impacts related to the proposed license amendments associated with the thermal power uprate have been analyzed and evaluated as follows:

- There will be no significant change in the types or in the amounts of any radiological effluents over those which have already been evaluated and found acceptable in the FES and evaluations associated with the Recapture Amendments. Similarly, there will be no significant increase in individual or cumulative occupational or population exposures.

- There will be no significant increase in the types or amounts of radioactive wastes over that already evaluated in the FES and evaluations associated with the Recapture Amendments.

- There will no significant increase in non-radiological impacts over those evaluated in the FES and evaluations associated with the Recapture Amendments.

Based on these analyses, it has been concluded that there are no significant radiological or non-radiological impacts associated with the thermal power uprate. The thermal power uprate will have no significant impact on the quality of human environment and does not involve an unreviewed environmental question as defined in Appendix B, the Environmental Protection Plan, of the operating licenses.

7.9 REFERENCES

1. "Final Environmental Statement Related to the Operation of Turkey Point Plant", dated July 1972, United States Atomic Energy Commission.

2. Letter, K. N. Harris (FPL) to USNRC, "Proposed License Amendments Operating License Expiration Date", dated February 25, 1992, L-92-31.

3. Letter, R. P. Croteau (USNRC) to J. H. Goldberg (FPL), "Environmental Assessment and Finding of No Significant Impact for Recapturing Construction Period in the License Term - Turkey Point Units 3 and 4", dated April 7, 1994.