# ATTACHMENT A

Proposed Change #185 to Technical Specifications and Bases

Affected Pages

- Technical Specification page 18
- Technical Specification page 120
- Bases page 142

## 1.2 SAFETY LIMIT

### 1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

## 2.2 LIMITING SAFETY SYSTEM SETTING

### 2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the ~~principle~~ principal process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve settings shall be as specified in Table 2.2.1. follows:

~~1 valve at ≤1080 psig~~
~~2 valves at ≤1090 psig~~
~~1 valve at ≤1100 psig~~
~~2 valves at ≤1240 psig (safety valves)~~

TABLE 2.2.1

PRIMARY SYSTEM RELIEF AND SAFETY VALVE SETTINGS

| Number and Type of Valve(s) | Lift Setting (1) |
|---|---|
| 1 safety relief valve | 1080 psig |
| 2 safety relief valves | 1090 psig |
| 1 safety relief valve | 1100 psig |
| 2 safety valves | 1240 psig |

Table 2.1.1 Note

1. As-left setpoint tolerance ±1%

   As-found setpoint tolerance ±3%

## 3.6 LIMITING CONDITIONS FOR OPERATION

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves ~~shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced to and maintained at or below 95% of rated power.~~

and atleast 3 of the 4 relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

## 4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

~~Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purposes of this limiting condition a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.~~

~~Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.~~

See Insert

Insert for <u>BASES</u> 3.6 and 4.6.D <u>Safety and Relief Valves</u>

Safety analyses have shown that only three of the four relief valves are required to provide the recommended pressure margin of 25 psi below the safety valve actuation settings as well as compliance with the MCPR safety limit for the limiting anticipated overpressure transient. For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. Safety analysis has been performed which shows that with safety and safety relief valves within $\pm 3\%$ of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

# ATTACHMENT B

Proposed Change #185 to Technical Specifications and Bases

New Pages

- Technical Specification page 18
- Technical Specification page 120
- Bases page 142

## 1.2 SAFETY LIMIT

### 1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

## 2.2 LIMITING SAFETY SYSTEM SETTING

### 2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the principal process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve settings shall be as specified in Table 2.2.1.

TABLE 2.2.1

Primary System Relief and Safety Valve Settings

| Number and Type of Valve(s) | Lift Setting(1) |
|---|---|
| 1 safety relief valve | 1080 psig |
| 2 safety relief valves | 1090 psig |
| 1 safety relief valve | 1100 psig |
| 2 safety valves | 1240 psig |

Note:

(1) As-left setpoint tolerance ±1%.
As-found setpoint tolerance ±3%.

## 3.6 LIMITING CONDITIONS FOR OPERATION

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves and at least three of the four relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

## 4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Safety analyses have shown that only three of the four relief valves are required to provide the recommended pressure margin of 25 psi below the safety valve actuation settings as well as compliance with the MCPR safety limit for the limiting anticipated overpressure transient. For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. Safety analysis has been performed which shows that with all safety and safety relief valves within ±3% of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

# <u>ATTACHMENT C</u>

Proposed Change #185

Justification For Operation With
A Relaxed Safety Valve Setpoint Tolerance

and

100% of Rated Power Operation With An Inoperable SRV
For The Vermont Yankee Nuclear Power Station

## 1.0 EXECUTIVE SUMMARY

This report documents safety analyses performed in support of the Vermont Yankee Nuclear Power Station. The analyses were performed to support implementation of the following two changes:

1) The setpoint tolerance of safety relief valves and safety valves may be increased from ±1% to ±3%. As-found valve setpoints within ±3% of Technical Specification setpoints are acceptable.

2) Technical Specifications currently require that reactor power be maintained ≤ 95% of rated thermal power when one of the four safety relief valves is inoperable. The supporting safety analyses demonstrate that this restriction is not necessary to meet the applicable acceptance criteria.

Justification for these changes is provided by presenting the results of safety analysis for those postulated events which may challenge the safety relief valves or safety valves. Safety analysis results demonstrate that applicable acceptance criteria are met for the above changes.

Section 2 of this report describes the scope of the safety analyses performed, the related acceptance criteria, and the computer codes used. The applicable acceptance criteria for the changes include:

- transient reactor pressure vessel pressure ≤ ASME Boiler Vessel Code limit (110% of design)
- no significant increase in safety relief valve or safety valve challenges
- fuel integrity assured by MCPR ≥ the safety limit value during overpressure transients
- LOCA design basis criteria for emergency core cooling systems, containment design basis, containment heat removal, radiological releases, and LOCA induced structural loads
- associated structures, systems, and components remain intact during safety relief valve discharge

Section 3 describes the analyses of postulated overpressure events and the results. The discussion includes a description of the SRV and SV modeling changes required to simulate the effects of a increased setpoint tolerance and an inoperable SRV. The analyzed cases are organized by the acceptance criteria which they support.

Section 4 describes the Loss of Coolant Accident (LOCA) analysis and the results.

Section 5 provides a mechanical loads analysis performed to verify the continued integrity of the SRV discharge piping and Torus during operation with a increased setpoint tolerance for all SRVs.

Section 6 provides the conclusions derived from the results of analyses. The above two changes may be implemented separately or in combination. Safety analysis results demonstrate that all the applicable acceptance criteria are met.

## 2.0 INTRODUCTION

### 2.1 Scope of Analyses

The Vermont Yankee reactor vessel safety/relief configuration includes 4 Safety Relief Valves (SRVs) which discharge to the Torus and 2 Safety Valves (SVs) which discharge to the drywell. The current Technical Specification setpoints are 1 SRV @ 1080 psig, 2 SRVs @ 1090 psig, 1 SRV @ 1100 psig, and 2 SVs @ 1240 psig. The current setpoint tolerance on all these valves is ±1% as required by the ASME Boiler and Pressure Vessel Code. These setpoints and their relationship to other relevant plant pressure parameters are shown in Figure 2-1. The proposed changes to the configuration include:

1) A change to allow as-found setpoints for the SRVs and SVs to be ±3% of the Technical Specification values. The current practice of meeting ±1% tolerance for the as-left condition as required by the ASME Boiler and Pressure Vessel Code would continue. The range of valve setpoints with ±3% tolerance is also shown in Figure 2-1.

2) A change to the Technical Specifications to remove the current requirement to operate at ≤ 95% rated power with an inoperable SRV. Operation at full power with an inoperable SRV would be permitted.

Safety analyses are performed to demonstrate that operation with these changes will not result in violation of applicable acceptance criteria. The safety analyses include:

a) overpressure transient analysis to demonstrate that the ASME overpressure limit (110% of design) continues to be met. This analysis is discussed in Section 3.2;

b) overpressure transient analysis to demonstrate no significant increase in challenges to SRVs and SVs. This analysis is discussed in Section 3.4;

c) a hot channel analysis to determine the impact of the changes on the operating Minimum Critical Power Ratio (MCPR) limits specified in the Core Operating Limits Report (COLR). This analysis is discussed in Section 3.3;

d) Loss of Coolant Accident (LOCA) analyses to demonstrate continuing compliance to LOCA design basis criteria for emergency core cooling systems, containment design basis, containment heat removal, radiological releases, and LOCA induced structural loads. This analysis is discussed in Section 4; and

e) mechanical loads analysis of the SRV discharge piping and Torus to demonstrate they remain intact during SRV discharge. This analysis is discussed in Section 5.

### 2.2 Acceptance Criteria and Their Applicability

The following set of acceptance criteria are applied in the safety analysis of the increased setpoint tolerances and full power operation with an inoperable SRV.

#### 2.2.1 ASME Overpressure Limit

The ASME Boiler and Pressure Vessel Code Section III-A permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig). The limiting overpressure Abnormal Operational Transient (AOT) analyzed is a Main Steam Isolation Valve (MSIV) closure at End of Full Power Life (EOFPL) without credit for reactor trip on MSIV position sensing. The next most limiting overpressure AOT is the Generator Load Rejection Without Bypass (GLRWOBP) at EOFPL. These two transients are re-analyzed in this report to demonstrate continued compliance with the ASME overpressure limit (110% of design). Specifically, the maximum transient pressure in the bottom of the reactor vessel must be ≤1375 psig.

#### 2.2.2 Operating MCPR Limits

Operational restraints on the minimum critical power ratio (MCPR) are placed in the Core Operating Limits Report (COLR)

to assure that safety limits on CPR are not violated during AOTs. The impact of the changes described in Section 2.1 on MCPR limits is determined by performing a hot channel analysis for the overpressure transient which yields the largest transient drop in CPR (ΔCPR). This transient is typically the GLRWOBP. The analysis is discussed in Section 3.3.

### 2.2.3 SRV and SV Challenges

In addition to the acceptance criteria on maximum overpressure (see Section 2.2.1) and MCPR (see Section 2.2.2), other operational restraints are imposed to provide assurance that SRV and SV challenges are minimized.

From Technical Specification 3.6.D.1[1], reactor power currently must be maintained at or below 95% rated power when one of the four SRVs is inoperable. This requirement was established to assure that SVs and fuel integrity are not challenged during AOTs while operating with an inoperable SRV. Since operating MCPR limits are established to assure fuel integrity during AOTs, fuel integrity is already assured for all AOTs. Since SVs discharge to the drywell and the discharge may contain low levels of radioactivity , it is prudent to minimize SV challenges. In Section 3.4, overpressure transient analysis is performed with the changes discussed in Section 2.1 to demonstrate challenges to SVs are unlikey during AOTs while operating at full power with an inoperable SRV. For this purpose, a GLRWOBP is the limiting AOT. Demonstration that SV challenges are unlikey is accomplished by analysis results indicating that peak steam line pressure remains a minimum of 25 psi below the SV setpoint.

As set forth in NUREG 0737, Section II.K.3.16, licensees are committed[3] to minimizing challenges to the SRVs. In Section 3.4, an evaluation of the impact of the two changes discussed in Section 2.1 is provided to demonstrate no significant increase in SRV challenges.

### 2.2.4 LOCA Limits

The various acceptance criteria that must be met are 10CFR50.46 criteria for emergency core cooling systems, containment design basis, containment heat removal, radiological releases, and LOCA induced structural loads.

The LOCA ECCS analysis demonstrates compliance with 10CFR50.46 requirements. These requirements include:

- Peak cladding temperatures below 2200°F,
- Total cladding oxidation below 17% at peak location,
- Hydrogen generated in the core below 1%, and,
- Core retaining a coolable geometry.

The LOCA analysis will emphasize small to intermediate breaks in the recirculation loops which cause the SRVs to open as well as steam line breaks outside containment. Results are discussed in Section 4.0. In the LOCA analysis, inoperability of 1 SRV was not considered since there were no cases for which all four SRVs were challenged. Other LOCA design basis criteria and compliance with these criteria are also discussed in Section 4.0.

### 2.2.5 Mechanical Loads

The affected structures, systems, and components (SSCs) within the scope of work discussed in this report have been previously analyzed as part of the Mark I Containment Long Term Program[4,5]. The analyses were performed in accordance with the ASME Boiler & Pressure Vessel Code, Section III, Division I, with addenda through Summer 1977. Acceptance criteria for all analyses conform to the applicable requirements developed in support of the Mark I Program[6].

All analyses performed in support of this initiative to revise SRV setpoint pressures to a common value and increase the setpoint tolerance were performed in accordance with the methods and acceptance criteria as discussed above. Results are discussed in Section 5.0.

## 2.3 Computer Codes Used

### 2.3.1 Non-LOCA NSSS Transient Performance

All Non-LOCA NSSS transient response analyses are performed with the RETRAN[7] computer code. The code has received generic approval from the Nuclear Regulatory Commission (NRC) for application to non-LOCA transients. Current YAEC BWR analysis methods with RETRAN, including the application of one-dimensional kinetics, are documented in Reference 8 and approved by the NRC in Reference 9.

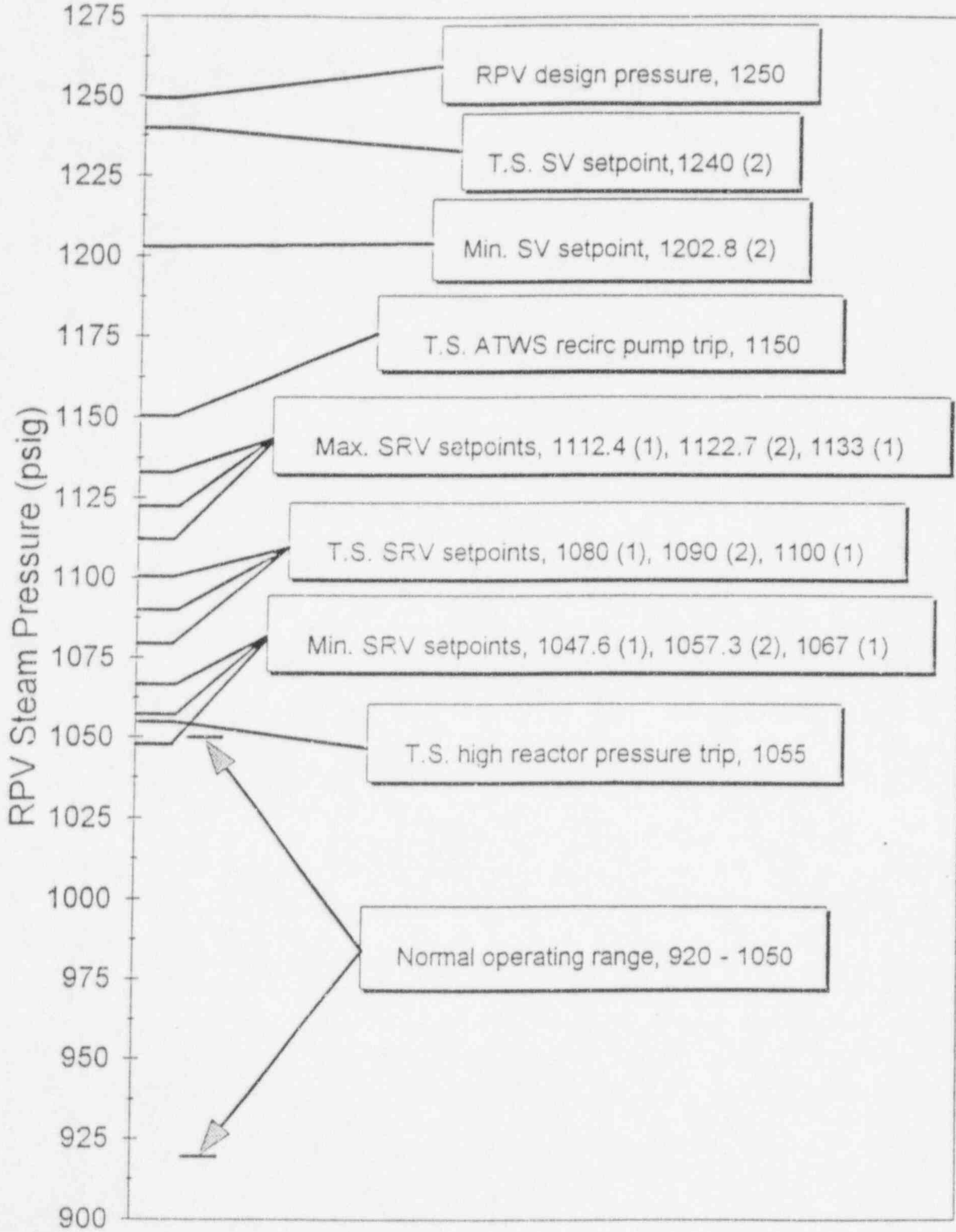
### 2.3.2 LOCA Analysis

The LOCA methods consist of the RELAP5YA[10] (BWR version) thermal-hydraulic analysis method and the FROSSTEY-2[11] fuel performance analysis method. The NRC approved the LOCA analysis methods for performing evaluation model licensing application covering the entire spectrum of LOCAs in References 12 and 13.

### 2.3.3 Mechanical Loads Analysis

The STARDYNE computer code was used in previous analyses for structures, systems, and components (SSCs) which are affected within the scope of work discussed in this report. This code was fully verified and industry accepted for the applications used. Analyses performed included dynamic response for time varying loads as well as static and thermal load cases. Computer analysis performed within the scope of this report also utilized the STARDYNE code.

FIGURE 2-1
OVERPRESSURE AND REACTOR PROTECTION SYSTEM SETPOINTS
WITH 3% TOLERANCES



## 3.0 NON-LOCA TRANSIENT ANALYSIS

In this section, non-LOCA transient analysis is presented. The analysis supports the following three changes:

1) SRV and SV setpoint tolerance increase from ±1% to ±3%; and,

2) operation at full rated power with an inoperable SRV.

The analysis is performed with approved methodology as discussed in Section 2.3.1. This section of the report is organized as follows:

- In Section 3.1, the features of the plant computer model which had to be modified to perform the analysis are described.

- In Section 3.2, analysis is presented which demonstrates continuing compliance with the ASME overpressure limit (110% of design).

- In Section 3.3, analysis is presented which demonstrates continuing compliance with fuel integrity limits by assuring MCPR ≥ the safety limit value .

- In Section 3.4, analysis and discussion is provided which demonstrate SV challenges are unlikely during an AOT and no significant increase in SRV challenges.

## 3.1 SRV & SV Modeling Changes

The first step in evaluating the impact of the changes is to modify the representation of the SRVs and SVs in the base RETRAN corewide model used in the current safety analysis. Only the model features important to this analysis are discussed.

### 3.1.1 Changes to Evaluate 3% Setpoint Tolerance

Plant model changes are necessary to accommodate analyzing the impact of increasing the as-found SRV & SV setpoint tolerance to ±3%.

The peak transient overpressure is maximized when SRV & SV opening setpoints are high. Also, higher transient pressures will cause greater core void collapse and yield more positive reactivity feedback from moderator density. It is conservative to assume high SRV and SV setpoints. All new transient analyses presented in this report will include SRV & SV operating characteristics associated with operation at the upper limit of the ±3% setpoint tolerance, except the analysis for SV challenges described in Section 3.1.3.

### 3.1.2 Change to Evaluate An Inoperable SRV

A goal of this calculation is to support power operation with one inoperable SRV. An inoperable SRV is assumed to be unable to perform its safety relief function within 3% tolerance of its setpoint. An additional change to the set of changes described in Section 3.1.1 above is developed here to evaluate this mode of power operation. An inoperable SRV is simulated by specifying an arbitrarily high opening setpoint ($1 \times 10^9$ psia) for one of the SRVs. With this arbitrarily high setpoint, the valve will fail to open during an overpressure transient.

3.1.3 Changes to Evaluate SV Challenges

Margin to SV lift is analyzed with the limiting AOT for overpressure, the GLRWOBP. This event results in the greatest overpressure for AOTs and should not be confused with the MSIV closure event without direct scram on MSIV position switch. The MSIVC is analyzed specifically for demonstration of ASME overpressure compliance. For evaluating SV challenges the analysis is typically performed with best estimate assumptions to demonstrate that a minimum of 25 psi margin exists to SV lift. The plant model is changed to reflect the expected tolerances of the SRVs and SVs. As found testing has demonstrated the expected tolerances of the SVs and SRVs to be less than 1%. For purposes of demonstration of no SV lift with an inoperable SV, a +1% tolerance is applied to the SRVs and a -1% tolerance to the SVs. This application of expected tolerance and use of the remainder of the assumptions for the ASME code and MCPR compliance analysis provides a conservative prediction of the margin to SV lift.

## 3.2 Compliance With The ASME Overpressure Limit

In this section, the reactor system response to design basis AOTs is discussed. The two transients analyzed are the MSIVC and the GLRWOBP. These are the two most limiting overpressure AOTs. The results for the GLRWOBP also provide boundary conditions for the hot channel (MCPR) analysis discussed in Section 3.3.

3.2.1 MSIVC Transients

The MSIVC transient is examined with RETRAN to demonstrate continuing compliance with the ASME overpressure limit discussed in Section 2 of this report. The limiting case for the current cycle is an MSIVC @ EOFPL with 67B scram times without credit for reactor trip on MSIV position sensing. Four cases were analyzed:

- The first case is the limiting MSIVC for the current plant design. Results are shown in Table 3-1 as the "Reference" case.
- The second case is the limiting MSIVC with 3% SRV and SV tolerance.
- The third case is also the limiting MSIVC but includes both changes, i.e., 3% SRV and SV tolerance, and an inoperable SRV.
- The fourth case is the limiting MSIVC with an inoperable SRV, and 15% SRV and SV setpoint tolerance. This case provides the limiting setpoint tolerance which yields no margin to the ASME overpressure limit.

Results for these cases are summarized in Table 3-1. Each case demonstrates that the ASME overpressure limit is met. Each case also produces a similar transient as typified by results for the limiting MSIVC with both changes (the third case). The sequence of events and transient parameters for this case are shown in Table 3-2 and Figure 3-1. With the SVs assumed to have an opening setpoint at the high end of the 3% tolerance, they do not open during the transients. Although the SVs would be challenged at their Technical Specification setpoint of 1240 psig, these results show they are not required to actuate in order to meet the ASME overpressure limit. Future SRV & SV testing may yield as-found setpoints above the 3% setpoint tolerance limit. Results for the fourth case with 15% tolerance may be used to support regulatory reporting requirements for this contingency.

3.2.2 GLRWOB Transients

The GLRWOBP @ EOFPL is also examined with RETRAN to demonstrate continued compliance with the ASME limit. After the MSIVC, it is the second most limiting overpressure transient. It is also the typical overpressure transient which yields the most limiting $\Delta$CPR. Four cases were analyzed:

- The first case is the limiting GLRWOBP for the current plant design. Results are shown in Table 3-1 as the "Reference" case.

- The second case is the limiting GLRWOBP with 3% SRV and SV tolerance.

- The third case is also the limiting GLRWOBP but includes both changes, i.e., 3% SRV and SV tolerance, and an inoperable SRV. This case provides the transient core boundary conditions for the hot channel analysis discussed in Section 3.3.

- The fourth case is a GLRWOBP, at licensed power plus calormetric uncertainties, with an inoperable SRV, +1% SRV tolerance, and -1% SV tolerance to demonstrate no SV lift.

Results for these cases are summarized in Table 3-1. Each case demonstrates that the ASME overpressure limit is met. Each case also produces a similar transient as typified by results for the limiting GLRWOBP with both changes (the third case). The sequence of events and transient parameters for this case are shown in Table 3-2 and Figure 3-2. With the SVs assumed to have an opening setpoint at the high end of the 3% tolerance, they do not open during the transients. The SVs would not be challenged at their Technical Specification setpoint of 1240 psig and these results confirm they are not required to actuate in order to meet the ASME overpressure limit. These cases are bounded by the MSIVC cases previously discussed in Section 3.2.1. The fourth case, the GLRWOBP, with 1% SRV and SV tolerances resulted in greater than 25 psi margin to SV lift.

### 3.3 Operating MCPR Limits

The current hot channel analysis was reviewed and the following information extracted. The transients which generate the most limiting drop in critical power ratio (ΔCPR) include the Turbine Trip Without Bypass (TTWOBP), the Generator Load Rejection Without Bypass (GLRWOBP), and the Loss of FeedWater Heater (LOFWH). The LOFWH event does not involve RV pressurization. The TTWOBP and the GLRWOBP are pressurization transients in which the SRVs are challenged.

Earlier SRV operation during these events by a lower opening setpoint would reduce core power through moderator reactivity feedback. The resulting MCPR would be less limiting.

Delayed SRV operation during these events by a higher opening setpoint would increase core power through moderator reactivity feedback. The limiting ΔCPR occurs in the GLRWOB transient. A hot channel analysis was performed with the core boundary conditions from the GLRWOBP @ EOFPL with 67B scram times and both changes (see case 3 in Section 3.2.2). Results show that the combined effects of a 3% setpoint tolerance increase and an inoperable SRV cause the ΔCPR to increase by 0.02. This is caused by positive moderator feedback from higher pressure in the top part of the core as control rods are inserted. Continuing compliance with fuel integrity limits is obtained when either or both changes are implemented by appropriate changes to the various operating MCPR limits identified in the Core Operating Limits Report (COLR).

### 3.4 SRV and SV Challenges

#### 3.4.1 SRV Challenges

As set forth in NUREG 0737, Section II.K.3.16, licensees are committed to minimizing challenges to the SRVs. In this section, each of the changes discussed in Section 3.1 is evaluated with respect to a potential increase in SRV challenges.

Current practice regarding SRV and SV setpoints is to assure ±1% tolerance is met as required by the ASME Boiler & Pressure Vessel Code. As-left setpoints always meet the ±1% tolerance. This report demonstrates that as-found setpoints within ±3% are acceptable and supported by the safety analysis. However, as-left setpoints will continue to meet ±1% tolerance as required by the ASME Boiler and Pressure Vessel Code. Thus, the probability of a SRV actuation by virtue of a low SRV setpoint is not increased by as-found setpoint tolerance increase from ±1% to ±3%.

The change to allow full power operation with an inoperable SRV has no significant impact on the probability of SRV actuation.

In conclusion, both changes meet the commitment to NUREG 0737, Section II.K.3.16 regarding minimizing challenges to

SRVs.

### 3.4.2 SV Challenges

Plant overpressure response is analyzed to assure that SV challenges are not likely to occur during AOTs. For this purpose, a GLRWOBP identified as the limiting AOT for overpressure was analyzed (see Section 2.2.3). The analysis is typically performed with best estimate assumptions to demonstrate that a minimum of 25 psi margin exists to SV lift for the most limiting AOT. The analysis was conservatively performed assuming:

- licensed power level plus calormetric uncertainties
- MST Scram Times
- an inoperable SRV
- +1% SRV setpoint tolerance
- -1% SV setpoint tolerance

Results for this case are summarized in Table 3-1. The case demonstrates that the SVs are not challenged at their Technical Specification setpoint of 1240 psig minus a 1% tolerance. The results show greater than 25 psi margin to SV lift. In conclusion, operation at full power with an inoperable SRV will not cause SV challenges during an AOT. The current Technical Specification limit of ≤ 95% rated power with an inoperable SRV is not required.

## TABLE 3-1
## CORE WIDE ANALYSIS SUMMARY

| Transient Description[a] | Peak Normalized | | Peak Pressure[b] (psig) | |
|---|---|---|---|---|
| | Power | Heat Flux | RV[c] | Steam Line |
| Reference MSIVC; EOFPL; 67B[d] | 1.9994 | 1.1995 | 1261 | 1230 |
| MSIVC; EOFPL; 67B; 3% Tolerance | 1.9997 | 1.1942 | 1283 | 1252 |
| MSIVC; EOFPL; 67B; 3% Tolerance; Inoperable SRV | 1.9997 | 1.1942 | 1316 | 1290 |
| MSIVC; EOFPL; 67B; 15% Tolerance; Inoperable SRV | 1.9997 | 1.1942 | 1375 | 1348 |
| Reference GLRWOBP; EOFPL; 67B[d] | 3.0883 | 1.2322 | 1233 | 1194 |
| GLRWOBP; EOFPL; 67B; 3% Tolerance | 3.0883 | 1.2322 | 1252 | 1216 |
| GLRWOBP; EOFPL; 67B; 3% Tolerance; Inoperable SRV | 3.0883 | 1.2322 | 1269 | 1237 |
| GLRWOBP; EOFPL; MST; 1% Tolerance; Inoperable SRV;[e] | 2.6318 | 1.1813 | 1233 | 1194 |
| ASME Overpressure Limit | ---- | ---- | 1375 | ---- |

(a) The transient cases are described in more detail in the text. The peak values are extracted from RETRAN output.

(b) The ASME overpressure limit is 1375 psig at the lowest elevation of the reactor coolant system (the lower hemisphere of the reactor vessel).

(c) The pressure in the bottom of the reactor vessel is determined by adding 2.87 psi to the lower plenum pressure in the RETRAN model to account for the elevation effect.

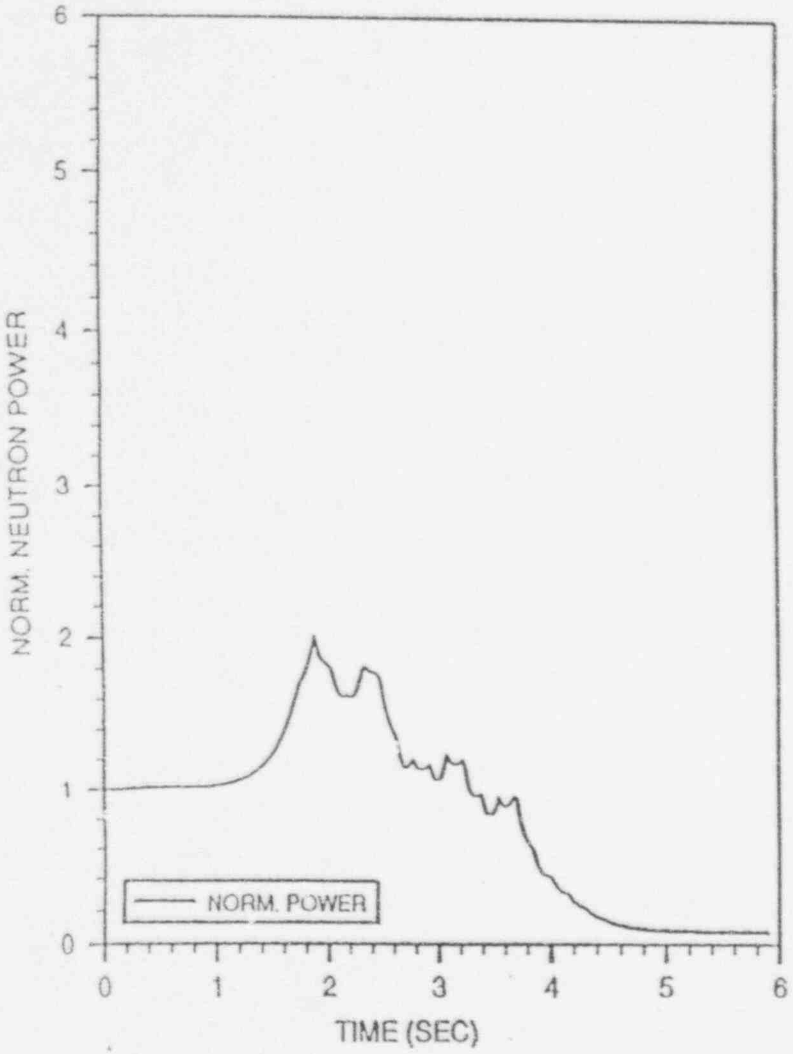
(d) These cases were rerun from the current cycle analysis. The results are shown here for reference in evaluating the effects of the two plant changes.

(e) This case demonstrates the recommended 25 psi margin to SV lift at the Technical Specification setpoint of 1240 psig minus 1% tolerance.

<u>TABLE 3-2</u>
<u>CHRONOLOGICAL SEQUENCE OF EVENTS</u>

| Transient Case | Event | Time (Seconds) |
|---|---|---|
| MSIVC; EOFPL; 67B; 3% Tolerance; Inoperable SRV | Start MSIV closure; bypass valves fail closed | 0.00 |
| | High flux scram setpoint reached | 1.45 |
| | Control rods begin insertion | 1.73 |
| | Peak core power occurs | 1.88 |
| | ATWS Recirc pump trip setpoint reached | 2.38 |
| | Threes SRVs open | 2.67 |
| | MSIVs fully closed | 3.00 |
| | Recirc pump field breakers open | 3.38 |
| | Peak RV pressure occurs | 4.80 |
| | Peak steam line pressure occurs | 5.10 |
| MSIVC; EOFPL; 67B; 3% Tolerance; Inoperable SRV; Direct Reactor Trip | Start MSIV closure; bypass valves fail closed | 0.00 |
| | Reactor scram on 10% MSIV closure setpoint reached | 0.71 |
| | Control rods begin insertion | 0.99 |
| | Peak core power occurs | 2.04 |
| | ATWS Recirc pump trip setpoint reached | 2.47 |
| | Three SRVs open | 2.71 |
| | MSIVs fully closed | 3.00 |
| | Recirc pump field breakers tripped | 3.47 |
| | Peak steamline pressure occurs | 3.96 |
| GLRWOBP; EOFPL; 67B; 3% Tolerance; Inoperable SRV | Start TCV closure; bypass valves fail closed; Rx tripped | 0.00 |
| | Control rods begin insertion | 0.28 |
| | TCVs fully closed | 0.31 |
| | Peak core power occurs | 0.80 |
| | ATWS Recirc pump trip setpoint reached | 1.40 |
| | Three SRVs open | 1.49 |
| | Recirc pump field breakers tripped | 2.40 |
| | Peak steam line pressure occurs | 3.10 |

MSIVC; EOFPL; 67B; 3% TOL; INOP. SRV
618tl
1 OF 5




MSIVC; EOFPL; 67B; 3% TOL; INOP. SRV
618tl
2 OF 5

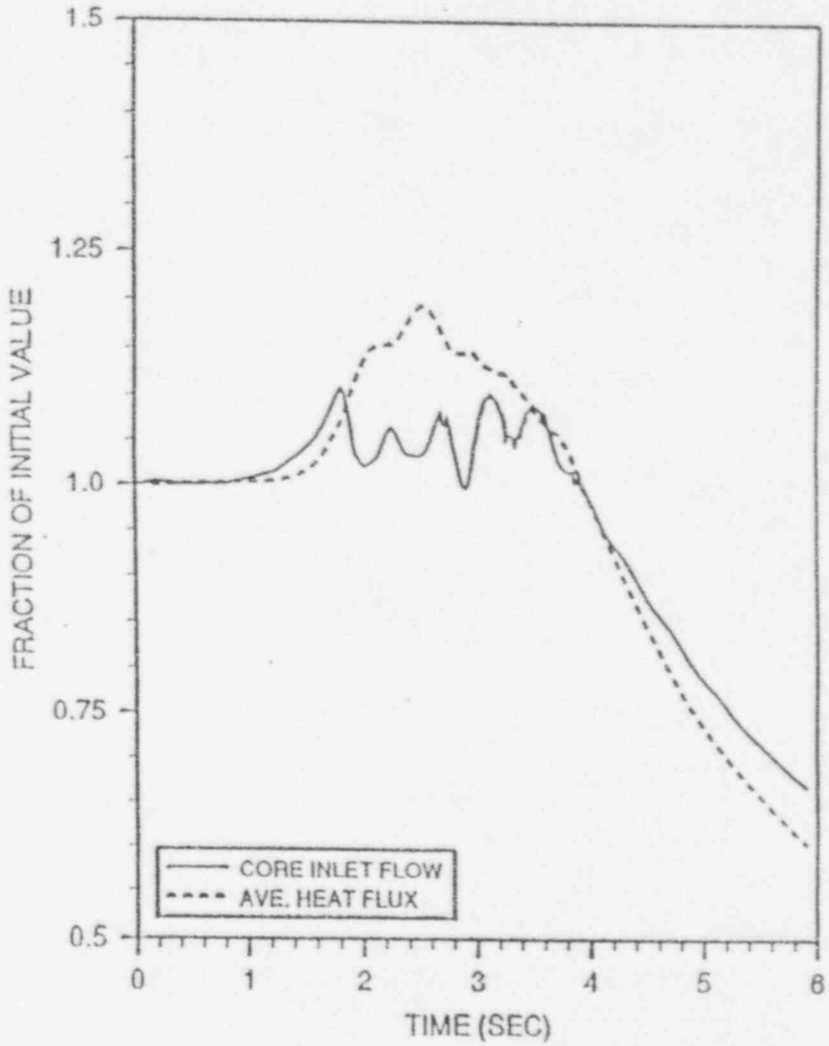
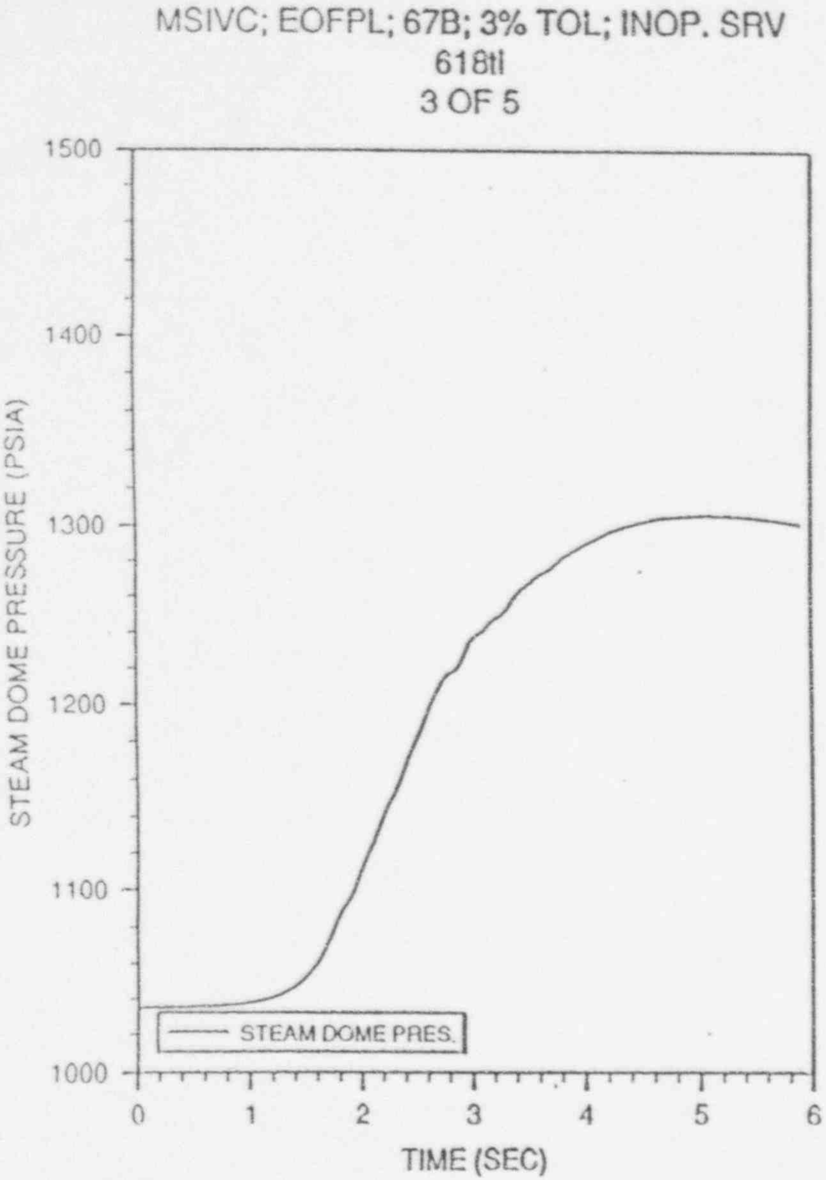



FIGURE 3-1



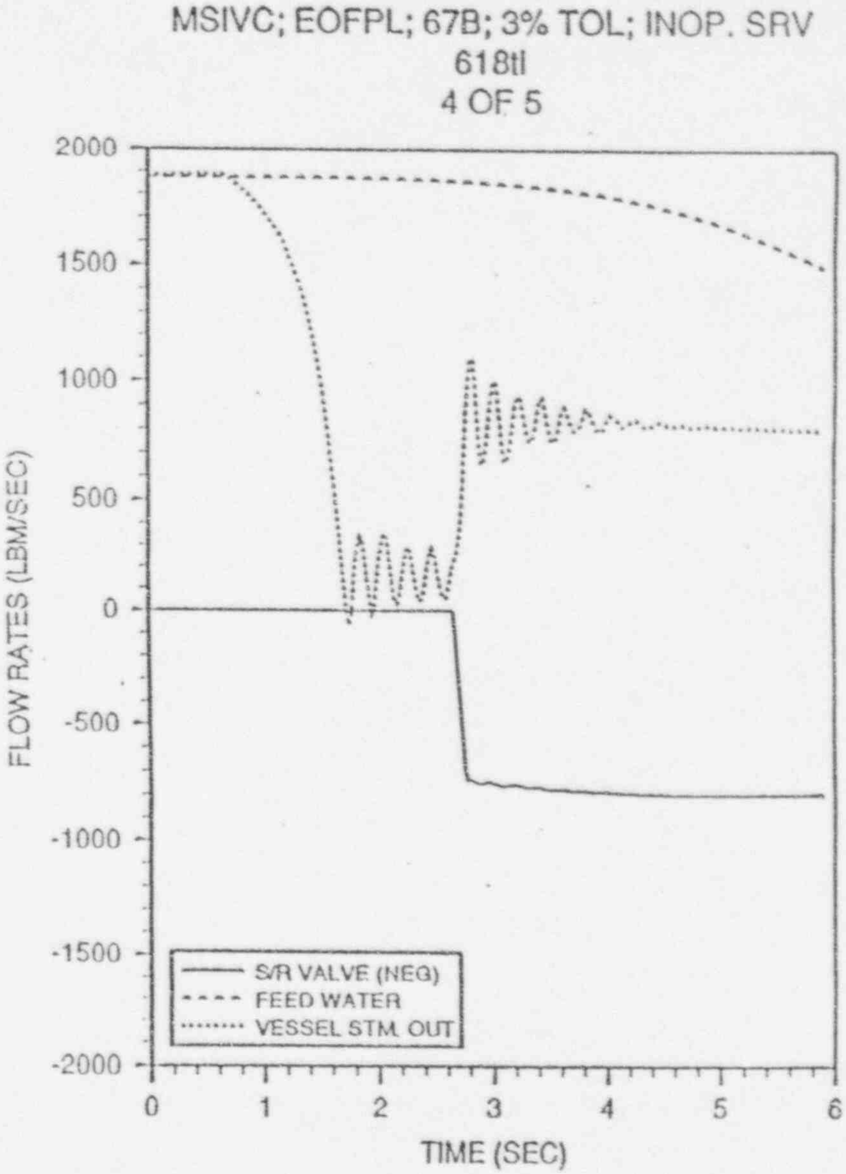
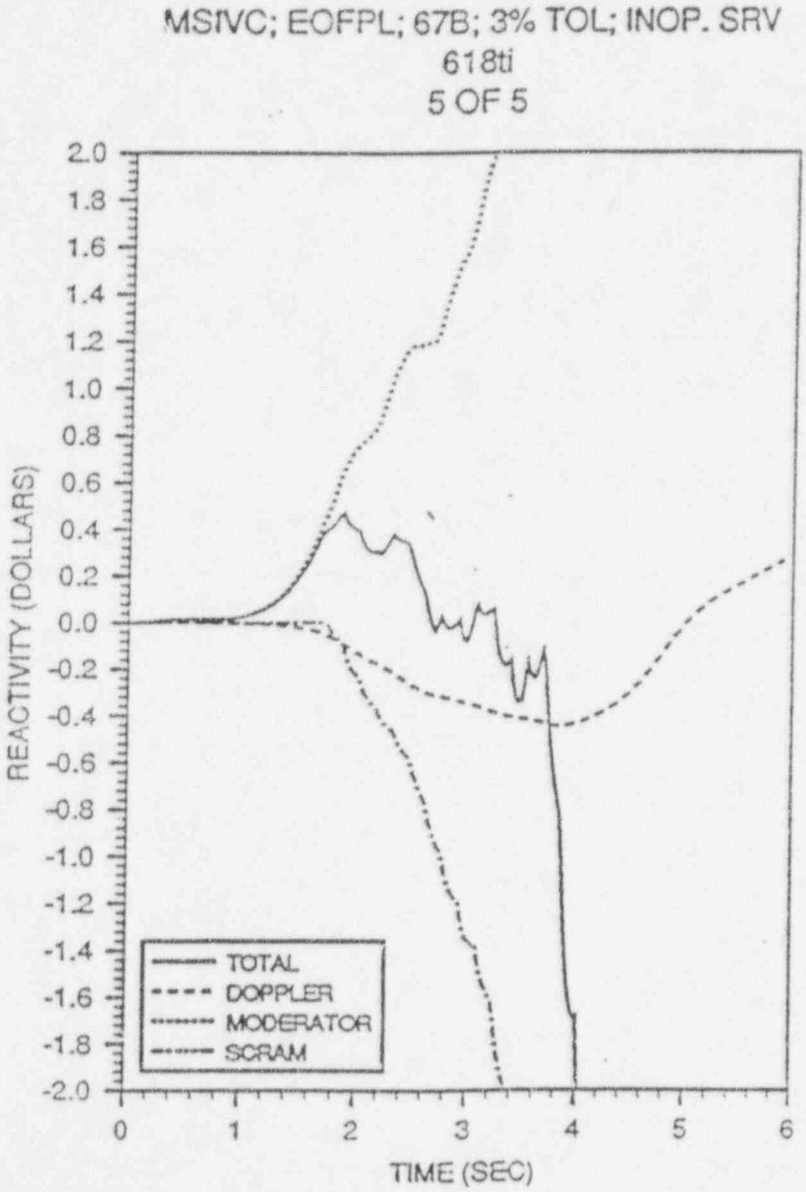
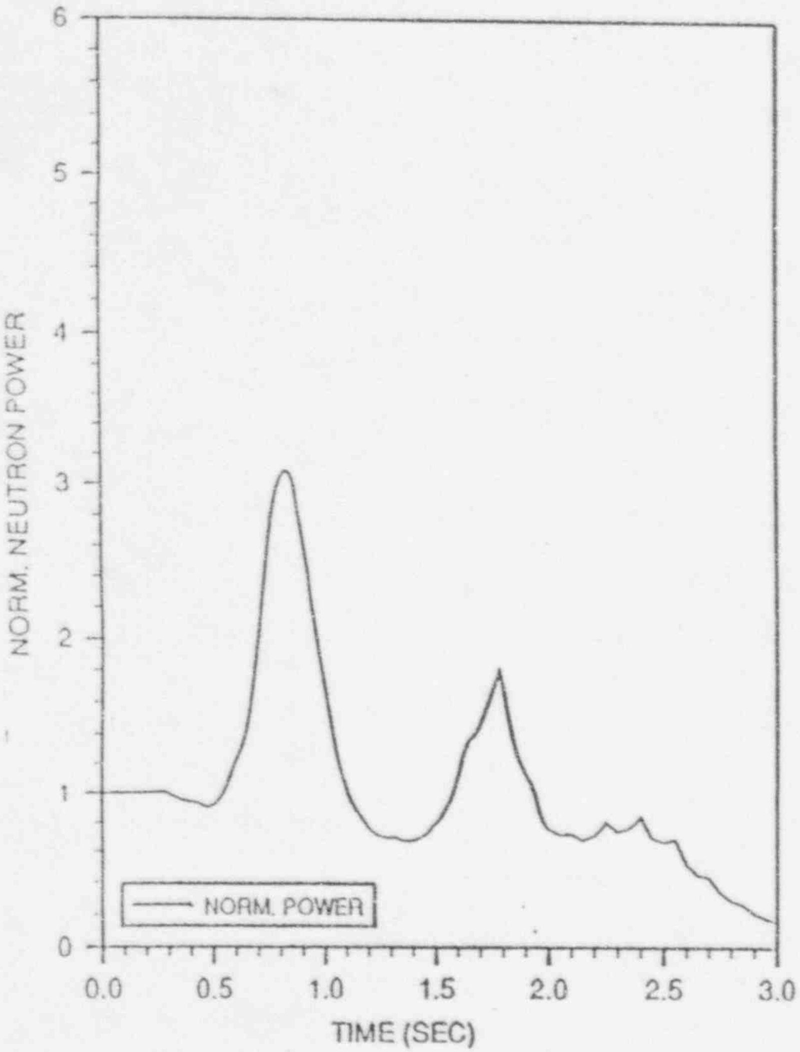




FIGURE 3-1 (CONTINUED)


MSIVC; EOFPL; 67B; 3% TOL; INOP. SRV
618ti
5 OF 5
REACTIVITY (DOLLARS)
2.0
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
-0.2
-0.4
-0.6
-0.8
-1.0
-1.2
-1.4
-1.6
-1.8
-2.0
0
1
2
3
4
5
6
TIME (SEC)
TOTAL
DOPPLER
MODERATOR
SCRAM

GLRWOBP; EOFPL; 67B; 3% TOL; INOP. SRV
610tl
1 OF 5




GLRWOBP; EOFPL; 67B; 3% TOL; INOP. SRV
610tl
2 OF 5

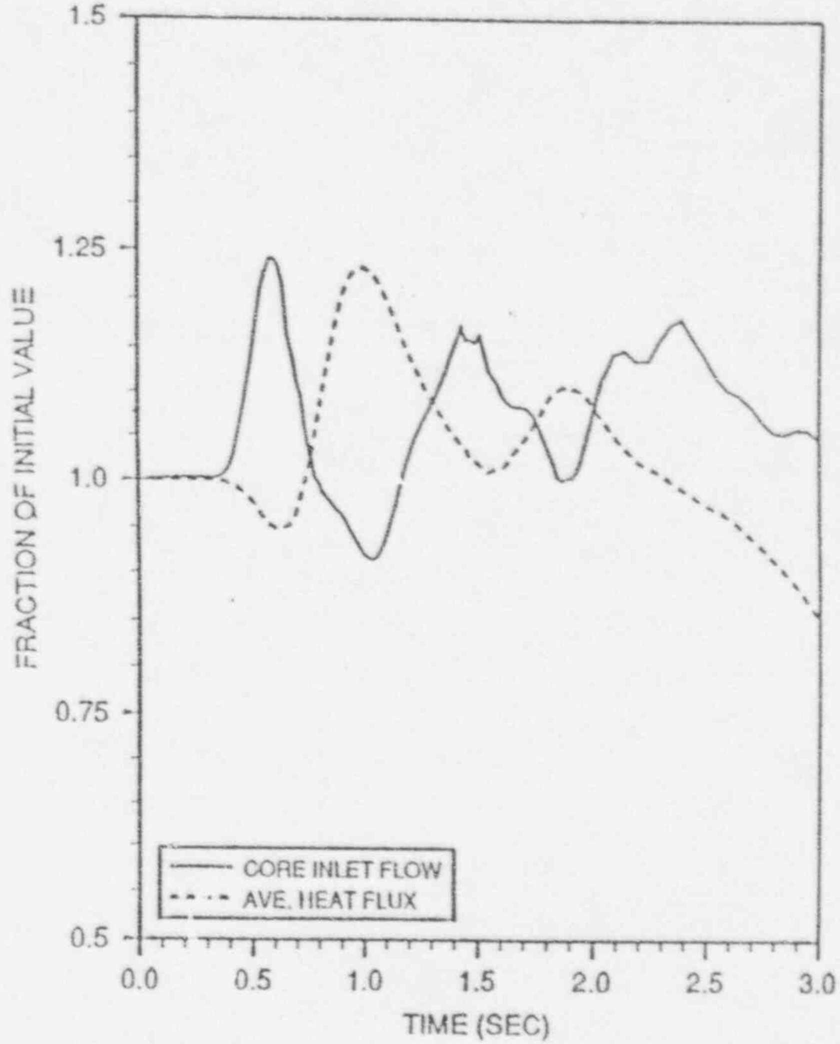
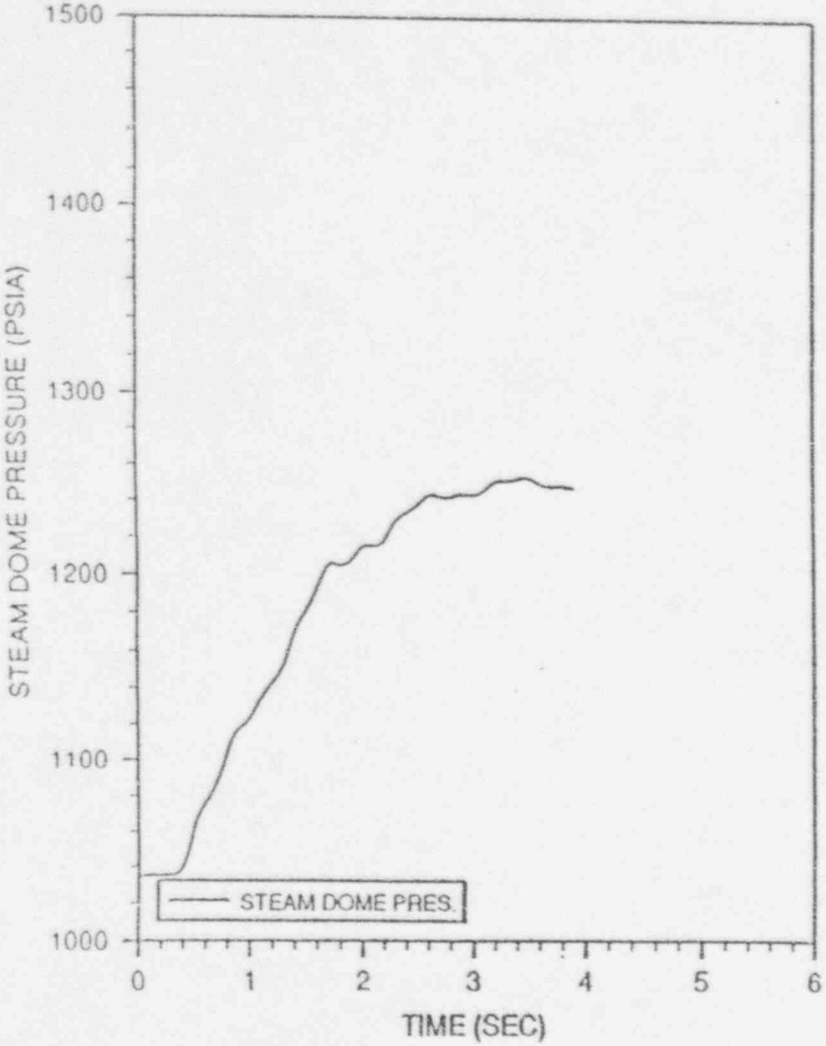



FIGURE 3-2

GLRWOBP; EOFPL; 67B; 3% TOL; INOP. SRV
610tl
3 OF 5




GLRWOBP; EOFPL; 67B; 3% TOL; INOP. SRV
610tl
4 OF 5

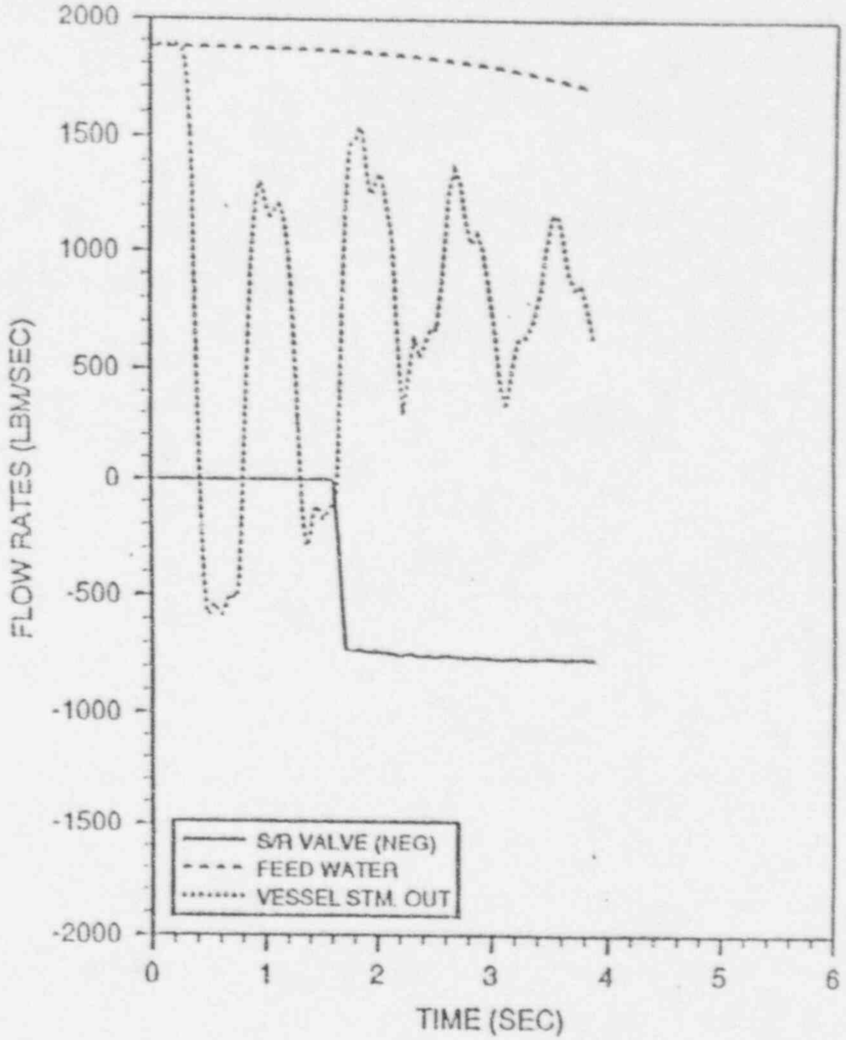
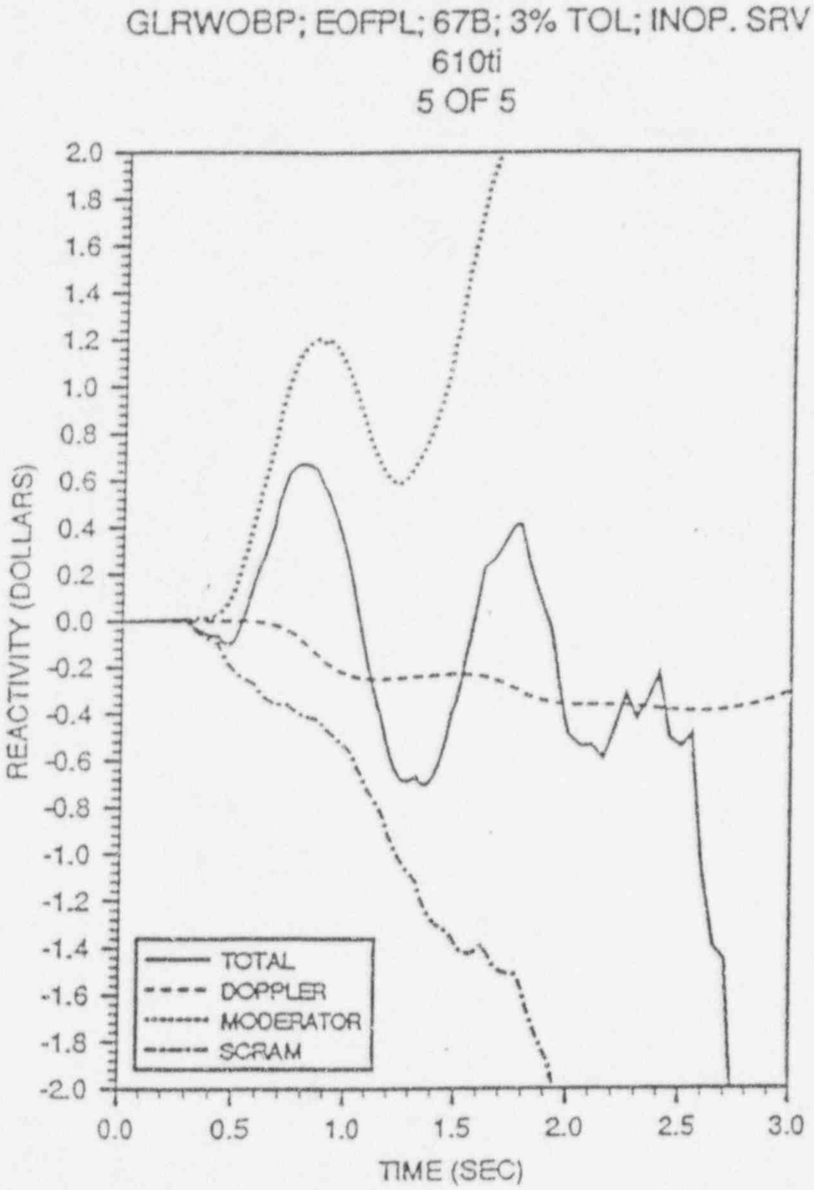



FIGURE 3-2 (CONTINUED)

FIGURE 3-2 (CONTINUED)



## 4.0 LOCA ANALYSIS

This section provides an assessment for LOCA events to verify the acceptability of operation with increased SRV and SV setpoint tolerances of ±3%. Operation at full power with an inoperable SRV was not addressed since for all LOCA cases only one SRV is challenged. Since only one SRV is challenged, an inoperable SRV would not change the results of the LOCA analysis. The various acceptance criteria defined in the VY FSAR that must be met are: 10CFR50.46 criteria for emergency core cooling systems; containment design basis; containment heat removal; radiological releases (10CFR100); and LOCA induced structural loads.

### 4.1 Scope of Analysis

The current LOCA licensing analysis[14], performed in conformance with 10CFR50.46 requirements, ensures that the most limiting combination of break size, break location and single failure have been considered in meeting the 10CFR50.46 acceptance criteria. The analysis shows that the limiting LOCA event is a break in the recirculation loop, with a break area of 0.6 $ft^2$, at the pump discharge location, with loss of one train of DC power as the single failure. The analysis of Reference 14 also shows that breaks at other locations, including the steam line, result in much lower peak cladding temperatures (PCT) than breaks in the recirculation line. Further, the analysis also shows that for breaks in the recirculation line larger than 0.4 $ft^2$, the SRVs would not be challenged because the energy removal at the break provides sufficient depressurization to overcome the effect of MSIV closure. Hence, in assessing the impact of changing as-found SRV setpoint tolerances on 10CFR50.46 criteria, only recirculation line breaks below 0.4 $ft^2$ need to be reassessed.

The Vermont Yankee design basis containment analysis is for a double ended break at the suction of the recirculation pump. For this large break size, the rate of depressurization is very fast and the SRVs are not challenged. Consequently, the SRV setpoint tolerance has no effect on the design basis containment analysis, and no further evaluation is required. Containment heat removal system design and Equipment Environmental Qualification are also based on the large double ended recirculation line break. The change in as-found SRV setpoint tolerances and operation at full power with an inoperable SRV does not affect large break LOCA events, and hence no further evaluation of these issues is required.

The Vermont Yankee design base accident for radioactive material releases and radiological effects is a complete severance of one main steam line outside the containment. For steam line breaks outside the containment, the MSIVs completely close and terminate radiological dose releases outside the containment, prior to SRVs being challenged. Hence, the changes to the as-found SRV setpoint tolerances and operation at full power with an inoperable SRV will not impact steam line breaks from the perspective of radiological releases.

LOCA induced structural loads are determined from the large double ended recirculation LOCA or steam line breaks inside the containment. SRVs are not challenged for these type of LOCA events. Consequently, the change in as-found SRV setpoint tolerances and operation at full power with an inoperable SRV does not affect LOCA induced structural loads, and hence no further evaluation of these issues is required.

### 4.2 Analysis Results

The LOCA analysis that forms the current licensing basis was used as the base analysis with which to evaluate the changes in as-found SRV setpoint tolerances and operation at full power with an inoperable SRV. The base analysis[14] was performed in conformance with 10CFR50.46 requirements and included a set of conservative initial operating conditions. In the base analysis, nominal values were used for the SRV setpoints. These were 1080 psig for one SRV, 1090 psig for two SRVs and 1100 psig for the remaining SRV. The SVs, which have a nominal setpoint currently at 1240 psig, were not modeled because the SRVs provided sufficient pressure relief to preclude challenging the SVs.

The current analysis, which consisted of a series of sensitivity studies on the base analysis, examined the impact of increasing the as-found SRV tolerances to ±3%. The resulting setpoints for opening and closing the SRVs are shown in Table 4-2.

The analysis examined break sizes below 0.4 $ft^2$ in the recirculation line, at the pump discharge location, for each case. Two limiting ECCS single failure assumptions were also addressed: (a) failure of an intact loop LPCI injection valve to open resulting in only two LPCS system available, and (b) failure of one train of DC power supply resulting in one LPCS system and one LPCI system. No credit was allowed for HPCI availability. These single failure assumptions were identical to those assumed in the base analysis of Reference 14.

The analysis evaluated break sizes down to 0.05 $ft^2$. The results of the various sensitivities, expressed in terms of peak cladding temperature (PCT), showed very little impact due to the revised as-found SRV setpoint tolerances of ±3%. As with the base analysis of Reference 14, all PCT values were well below the acceptance limit of 2200°F and all other 10CFR50.46 criteria continued to be met. The PCT differences between the sensitivity studies and the base case ranged from -28°F to +32°F regardless of which SRV is challenged. As a function of break size, this is shown in Figure 4-2. Break sizes below 0.05 $ft^2$ were not evaluated because the PCT shows a steady decreasing trend as break sizes decreased from 0.4 $ft^2$ to 0.05 $ft^2$. The conclusion of the analysis is that the current LOCA licensing basis will not be impacted by the change to the as-found SRV setpoint tolerances.

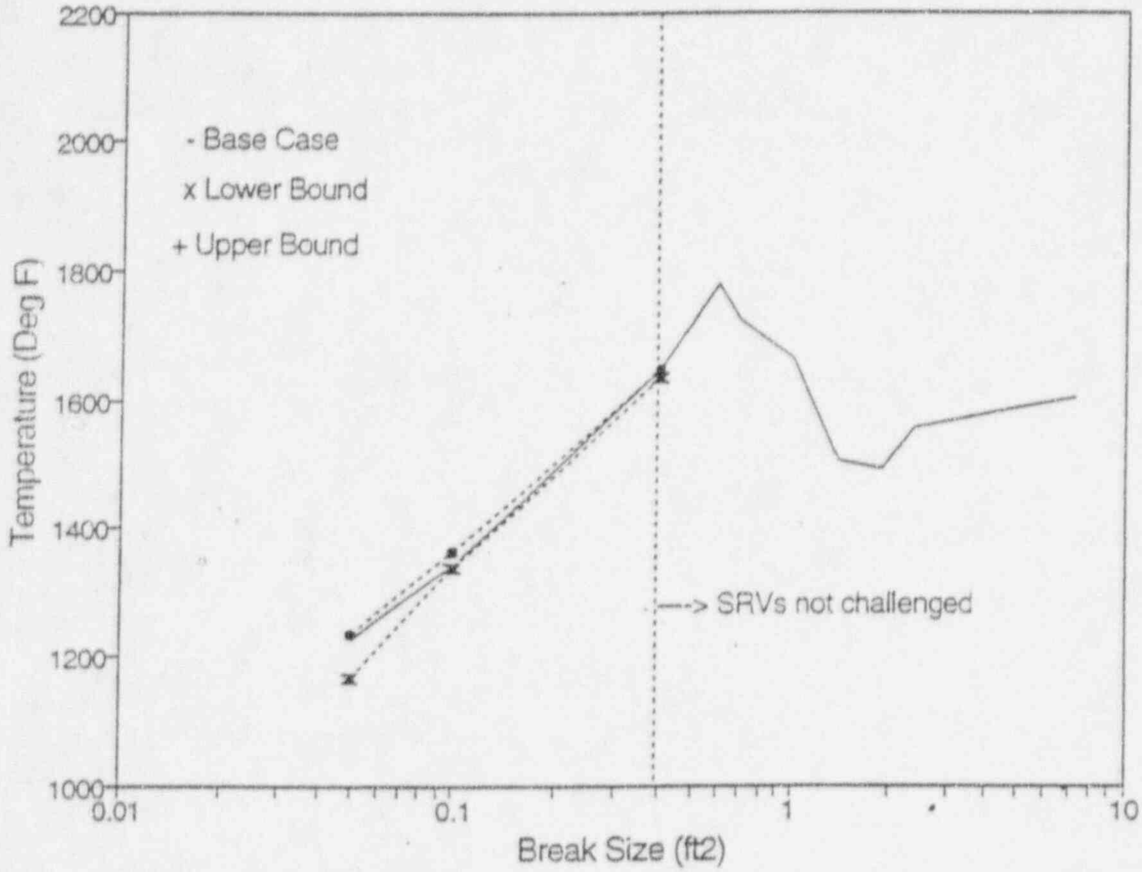
## TABLE 4-2

## SRV PRESSURE SETPOINTS WITH 3% TOLERANCE

| SRV Identification Numbers | Nominal Setpoints (psig) | Setpoints -3% Tolerance (psig) | | Setpoints + 3% Tolerance (psig) | |
|---|---|---|---|---|---|
| | | Open | Close[a] | Open | Close[a] |
| #1 | 1080 | 1047.6 | 1016.2 | 1112.4 | 1079.0 |
| #2 and #3 | 1090 | 1057.3 | 1025.6 | 1122.7 | 1089.0 |
| #4 | 1100 | 1067.0 | 1035.0 | 1133.0 | 1099.0 |

(a) Closing setpoints include 3% blowdown.

FIGURE 4-2
PEAK CLAD TEMPERATURE VERSUS BREAK AREA



## 5.0 MECHANICAL LOADS ANALYSIS

The purpose of the analysis is to determine the acceptability of as-found SRV setpoint tolerance changes on SRV piping/supports and the effect of increased discharge loads into the Torus.

The method used is to determine the load/stress increases in each of the affected areas and to compare these increases to the margins which exist in the current analyses. In cases where the stress margin exceeds the anticipated stress increase, no further analysis is required. In other cases, the original Mark I Program analyses were reviewed and conservatism removed to calculate lower, more accurate, actual stress values.

The results of this analysis show that the increase of as-found setpoint tolerance is acceptable for all four lines as they are currently installed at Vermont Yankee. This conclusion applies to all SRV piping, supports, and Torus structure.

## 6.0 CONCLUSIONS

The following conclusions regarding two changes are derived from the results of analyses documented in previous sections. The supporting safety analysis has been performed in a conservative way so that the two changes discussed below in Sections 6.1 and 6.2 may be implemented either separately or in combination.

### 6.1 SRV and SV Setpoint Tolerance Increase

The as-found SRV and SV setpoint tolerances may be increased from the current $\pm 1\%$ to $\pm 3\%$. Safety analyses assuming the increased tolerances verify the following acceptance criteria:

1) The ASME overpressure limit of 110% of design is met. This was verified by re-analyses of the two most limiting overpressure transients, i.e. the Main Steam Isolation Valve Closure (MSIVC) and the Generator Load Rejection Without Bypass (GLRWOBP).

2) SV challenges will be minimized during the most limiting AOT. This was verified by an analysis of a GLRWOBP.

3) The probability of SRV challenges during overpressure events will not be significantly increased as set forth in NUREG 0737, Section II.K.3.16.

4) Fuel integrity during overpressure transients will remain assured by adjusting the various operating MCPR limits specified in the Core Operating Limits Report (COLR) to reflect the effect of $\pm 3\%$ tolerance.

5) Re-analysis of the limiting Loss of Coolant Accidents confirms that acceptance criteria listed in Section 2.2.4 are met.

6) A mechanical loads analysis of the SRV discharge piping and the Torus confirmed the integrity of these structures, systems, and components during SRV discharge with the increased setpoint tolerance.

### 6.2 Full Power Operation With An Inoperable SRV

From Technical Specification 3.6.D.1, reactor power currently must be maintained $\leq$ 95% of rated power when one of the four SRVs is inoperable. This limit has been imposed to assure no challenge to the SVs in the event of an AOT. It is prudent to minimize challenges to the SVs which discharge to the drywell.

The limiting AOT discussed in Section 6.1, i.e. the GLRWOBP, was analyzed assuming an inoperable SRV while operating at full power. Results show no SV challenge. The 95% power restriction with an inoperable SRV in the Technical Specifications is not required.

Complete support for operation at full power with both changes is provided by performing all the safety analyses discussed in Section 6.1 assuming $\pm 3\%$ SRV and SV setpoint tolerances ($\pm 1\%$ for SV challenges) and an inoperable SRV. Results verify all the acceptance criteria previously discussed in Section 6.1 are met provided the various operating MCPR limits specified in the COLR are adjusted to reflect both changes. A hot channel analysis of the limiting $\Delta$CPR overpressure transient, typically the GLRWOBP, confirmed that a 0.02 increase in the operating MCPR limits bounds this effect.

These analysis results allow the conclusion that a 95% power restriction with an inoperable SRV is not necessary. This does not preclude the prudence of avoiding prolonged operation with an inoperable SRV.

## 7.0 REFERENCES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4. Technical Report TR-5319-1, Revision 2, Mark I Containment Program, "Plant Unique Analysis Report of the Torus Suppression Chamber for Vermont Yankee Nuclear Power Plant", November 30, 1983.

5. Technical Report TR-5319-2, Original Issue, Mark I Containment Program, "Plant Unique Analysis Report of the Torus Attached Piping for Vermont Yankee Nuclear Power Plant", October 3, 1983.

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7. EPRI NP-1850, Volume 1, Rev. 4, "RETRAN-02 - A Program for Transient Thermal- Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics (Revision 4)", November, 1988.

8. YAEC-1693, "Application of One-Dimensional Kinetics to Boiling Water Reactor Transient Analysis Methods", June, 1989.

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13. Letter, P. Sears (USNRC) to L. A. Tremblay (VYNPC), "Safety Evaluation for Vermont Yankee Nuclear Power Station RELAP5YA LOCA Analysis Methodology" (TAC No. M74595), October 21, 1992. (Docket No. 50-271).

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