GE Nuclear Energy

175 Curtner Avenue
San Jose, CA 95125

NEDC-32041P
Revision 2
Class III
April 1996

~~GE PROPRIETARY INFORMATION~~
Non-Proprietary Version

SAFETY REVIEW FOR EDWIN I. HATCH NUCLEAR POWER PLANT UNITS 1 AND 2 UPDATED SAFETY/RELIEF VALVE PERFORMANCE REQUIREMENTS

Michael R. Jonzen

Approved by: Hoa Hoang
H. X. Hoang
Project Manager

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between the customer and GE, as identified in the purchase order for this report, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than the customer or for any purpose other than that for which it is intended, is not authorized, and with respect to any unauthorized use, GE makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

PROPRIETARY INFORMATION NOTICE

This document contains proprietary information of the General Electric Company and is submitted in confidence solely for other purpose or purposes stated in the transmittal letter. No other use, direct or indirect, of the document or the information it contains is authorized. Furnishing this document does not convey any license, expressed or implied, to use any patented invention or, except as specified above, any proprietary information of GE disclosed herein or any right to publish or make copies of the document without prior written permission of GE.

Particularly sensitive information is marked in the report by a bar in the left margin next to the information of concern. Transmission of this information to third parties is not authorized.

TABLE OF CONTENTS

TABLES

FIGURES

SUMMARY

This report presents analyses for proposed changes to safety/relief valve performance requirements. An opening pressure setpoint Upper Limit is established, whereby valve openings below this pressure assure that vessel pressure remains below the 1375 psig criterion value for the most limiting reactor transient. New nominal opening setpoints for the valves are determined. Analyses are presented which support plant operation with one valve out of service. Current and proposed performance requirements are summarized in Table 1-1.

The proposed performance requirements shown in Table 1-1 are primarily Technical Specification changes which produce the desired operational flexibility (i.e., fewer Licensee Event Reports, reduced forced outages, and a decrease in maintenance and surveillance testing costs). Actual physical changes within the plant are minimal. The tolerance of ±3%, one automatic depressurization system, low-low set, or safety/relief valve out of service, and the Upper Limit value are "paper" changes related to the American Society of Mechanical Engineers upset limit testing criterion and the design basis analyses. The only physical change involves increasing the mechanical safety/relief setpoints to 1150 psig. Therefore, the proposed performance requirements involve few physical changes in order to greatly reduce the operational impact of setpoint drift, reactor hydrostatic testing and inoperable valves.

The vessel overpressure protection analysis and fuel thermal limits evaluations discussed in this report (Revision 2) are applicable to fuel cycles 9 and 10 for Hatch 2 and fuel cycles 13 and 14 for Hatch 1. The report also presents the vessel overpressure analysis performed for the Unit 1 and 2 licensing submittal for power uprate operation. Since vessel overpressure and thermal limits analyses are cycle-specific, it must be demonstrated prior to each reload that appropriate criteria are satisfied. All other information documented herein is cycle-independent.

1.0 INTRODUCTION

1.1 PURPOSE

This report presents the results of an evaluation to update the safety/relief valve (SRV) performance requirements at Plant Hatch, Units 1 and 2. The performance changes are selected to minimize the impact on plant operation from pressure relief system-related problems due to SRV setpoint drift and inoperable SRVs. Georgia Power Company (GPC) requested an analysis which (1) maintains the surveillance requirement tolerance to ±3% for the proposed nominal SRV setpoints, (2) establishes an Upper Limit on the SRV opening pressure for all SRVs, and (3) permits one SRV (in the safety mode, automatic depressurization system (ADS) mode or low-low-set (LLS) mode) to be out of service (OOS) during continuous operation. A comparison of the present and proposed performance requirements is shown in Table 1-1.

A description of the SRV configuration employed in the Hatch units is provided in Section 1.2. The current performance requirements for the SRVs are discussed in Section 1.3. Each of the present performance requirements pertinent to this analysis is identified, as well as the associated limitation and the remedial actions for exceeding the limit. Section 1.4 discusses proposed performance requirement changes, associated limits, and analyses required to support each proposed change.

Section 2.0 explains the analysis approach, listing the type of analyses performed and the basis for determining the Upper Limit and the new nominal setpoints.

Section 3.0 provides the bases and the results of analyses for the limiting overpressurization event. These analyses are used to establish a SRV opening pressure Upper Limit, which bounds results where valves open at lower pressures.

Section 4.0 describes the bases and results of analyses to determine the effects of the most limiting abnormal operational pressurization event on fuel thermal limits.

Section 5.0 presents results of a review of emergency core cooling system (ECCS)/loss of coolant accident (LOCA) analyses with increased SRV setpoint pressures and 1 SRV OOS. The most significant scenarios, viz., the design basis accident, small break LOCA, and steamline break outside of the containment, are addressed.

In Section 6.0, the impact of higher steamline pressures on the high pressure water restoration systems, i.e., high pressure core injection (HPCI) and reactor core isolation cooling (RCIC), is discussed.

The effects of increased SRV opening pressures on containment components are assessed in Section 7.0. Containment pressure and temperature response, as well as increased loads on the SRV piping, T-quencher, torus shell, and submerged torus structure, are addressed.

Section 8.0 addresses the ATWS mitigation capability with one SRV OOS.

Section 9.0 provides conclusions drawn from the evaluations.

Throughout the report, references are made to SRV opening pressures and SRV setpoints. For purposes of this engineering evaluation, any pressure at which a SRV opens is identified as the SRV opening pressure. The SRV setpoint is the opening pressure at which the SRV has been set.

1.2 BACKGROUND

The nuclear pressure relief system at Hatch consists of 11 Target Rock 2-stage SRVs for each unit located on the main steam line between the reactor vessel and the first isolation valve within the drywell. The SRVs provide the following primary functions:

(1) Overpressure relief operation. The SRVs open automatically to limit the vessel pressure excursion during a postulated pressurization transient event.

(2) Overpressure safety function (spring safety mode). The SRVs, functioning in the self-actuated safety mode, open to prevent reactor vessel overpressurization.

(3) Depressurization operation. The automatic depressurization system (ADS) function is performed by selected SRVs, and these valves open automatically as part of the emergency core cooling system for events involving small breaks in the reactor vessel process barrier. Seven of the eleven Hatch SRVs operate in the ADS mode.

(4) Low-Low-Set (LLS) operation. Following initial opening and closure of SRVs, selected SRVs operate in the LLS mode opening and closing at pressures lower than the other SRVs to reduce the amount of cycling on the non-LLS SRVs. Four of the eleven Hatch SRVs operate in the LLS mode.

1.3 PRESENT PERFORMANCE REQUIREMENTS

The present performance requirements for the SRVs and the respective limits are listed in Table 1-1, along with the proposed changes. The following sections discuss the potential impact of each of these requirements.

1.3.1 SRV Setpoint Tolerance (Table 1-1, Items 1 through 4)

In accordance with the Improved Technical Specifications (Ref. 1), the safety-function SRVs are required to open within ±3% of the nominal setpoints and within ±1% following testing. These changes were justified and approved during the Technical Specification Improvement Program. No changes to the ±3% or ±1% tolerance are proposed.

1.3.2 SRVs/ADS/LLS Valves OOS (Table 1-1, Items 5, 6 and 7)

The Technical Specifications require all eleven SRVs to remain operable in their safety mode, all seven SRVs to remain in their ADS mode, and all four SRVs to remain in their LLS mode during continuous operation. Plant operation with one or more of these valves OOS beyond the time limits of Tech. Specs. LCOs is not allowed. The Tech. Specs. require the plant to be placed in a cold shutdown condition when one or more of these valves are inoperable for longer than the duration specified in the applicable action statements. Therefore, changing Tech. Specs. to permit one SRV to be inoperable for the primary functions listed above will reduce the number of forced plant shutdowns.

1.3.3 SRV Nominal Setpoints (Table 1-1, Item 8)

The Technical Specifications require setpoints for the SRVs to be set as follows within a ±1% tolerance:

Hatch 1	Hatch 2
4 SRVs at 1110 psig	4 SRVs at 1120 psig
4 SRVs at 1120 psig	4 SRVs at 1130 psig
3 SRVs at 1130 psig	3 SRVs at 1140 psig

The proposed performance change is to increase the setpoints on all 22 SRVs to 1150 psig. This change will increase the "simmer margin" and reduce SRV pilot leakage which may occur over a typical operating cycle. (The "simmer margin" is defined as the difference between the SRV setpoints and the vessel steam dome pressure.) The proposed change will increase the simmer margin to approximately 115 psid and is consistent with GE recommendations in Reference 2.

1.3.4 Electronic Actuation Logic (Non-Safety)

Both units at Plant Hatch have an electronic actuation logic system which was installed several years ago. Unit 1 FSAR Section 4.4.5 and Unit 2 FSAR 5.2.2.2 describe this system in more detail. The system provides a backup to the mechanical actuation mode. The equipment serving this backup system was procured and installed as if it was safety-related, but the logic system is considered non-safety related and is not considered in the safety analysis. However, GPC plans to keep the electronic actuation logic setpoints for the eleven SRVs staggered and distributed among three groups. The current electronic setpoints are the same as the nominal mechanical setpoints described in Section 1.3.3. The proposed electronic setpoints are virtually unchanged from the current values. The only difference is that the Unit 1 electronic setpoints will be the same as Unit 2.

1.3.5 Low-Low-Set System Setpoints

The Technical Specifications currently require operability of 4 LLS SRVs with opening and closing setpoints, as shown below:

	Hatch 1		Hatch 2	
	Open	Close	Open	Close
Low Pressure Setpoint LLS Valve (psig)	≤1005	≤857	≤1010	≤860
Medium-Low Pressure Setpoint LLS Valve (psig)	≤1020	≤872	≤1025	≤875
Medium-High Pressure Setpoint LLS Valve (psig)	≤1035	≤887	≤1040	≤890
High Pressure Setpoint LLS Valve (psig)	≤1045	≤897	≤1050	≤900

There will be no changes to the opening and closing pressures for the LLS valves resulting from this study. However, the study justifies one LLS valve being out of service for continuous operation. At least three LLS valves must remain in service at the specified setpoints since these requirements are part of the system design basis. Not only does LLS reduce the duty factor on the non-LLS SRVs due to cycling open and closed during pressurization transients, the system acts to maintain vessel pressure within the operational domain of the high pressure water restoration systems, e.g., HPCI and RCIC, during high pressurization operational occurrences and hypothetical accident situations.

1.4 PROPOSED PERFORMANCE REQUIREMENT CHANGES

This section discusses the effect of each set of the proposed performance requirement changes and the analyses necessary to support the changes. The present and proposed SRV performance requirement changes are shown in Table 1-1.

1.4.1 Upper Limit (Table 1-1, Items 1 and 2)

The Upper Limit is defined as "the SRV opening pressure up to which plant performance has been analyzed." Consequently, as long as the SRV opening pressures remain below the Upper Limit, the plant remains within analyzed conditions, and the SRVs remain capable of performing their pressure relief function. Therefore, the number of LERs and safety evaluations caused by SRV opening pressures drifting from the nominal setpoints will be reduced.

To establish an Upper Limit, the safety concerns affected by SRV setpoint drift must be evaluated. These include assessments of overpressure protection, ECCS/LOCA performance,

thermal limits, HPCI and RCIC performance, and containment response. The overpressure protection analysis determines the highest opening pressure at which the American Society of Mechanical Engineers (ASME) upset overpressure criterion remains satisfied.

This value is then used to analyze the remaining safety concerns. The highest opening pressure that meets all safety concerns determines the value of the Upper Limit. The Upper Limit is further discussed in Sections 2.2 and 3.0 of this report.

1.4.2 ±3% Tolerance (Table 1-1, Item 3)

The ASME has expanded the acceptance criterion for SRV performance testing from ±1% to ±3% per ANSI/ASME OM-1-1981 (Ref. 3). The acceptance criterion defines the range of expected in-service performance of a SRV. Beyond this criterion, valve refurbishment is required, and additional valves must be removed from the plant for testing. The increased tolerance on the acceptance criterion potentially reduces the number of valves that will exceed the in-service performance testing requirements, thus reducing the cost of SRV surveillance testing. The ±3% tolerance criterion is contained in the Hatch Technical Specifications, and is retained for the revised nominal setpoints.

[redacted]

1.4.3 SRV Installation Tolerance of ±1% (Table 1-1, Item 4)

Prior to placing new or refurbished valves in s[illegible] the SRV setpoints are adjusted to within ±1% of the nominal settings. Installation of [illegible] thin a ±1% tolerance ensures that there is margin to the ±3% in-servic[illegible]ng crite[illegible] the Upper Limit opening pressure. In this manner, valve integrity and the benefits of the increased surveillance requirements tolerance and the Upper Limit are maintained from cycle to cycle. The ±1% tolerance is contained in the Hatch Technical Specifications, and is retained for the revised nominal setpoints.

1.4.4 One SRV/ADS/LLS Valve OOS (Table 1-1, Items 5, 6 and 7)

The conservative analysis supports one SRV being inoperable in its overpressure relief operation and/or safety function mode. The analysis also supports one ADS valve to be inoperable for continuous operation. This ECCS function is on seven of the 11 SRVs. It is designed to depressurize the reactor during a small break LOCA if HPCI fails or is unable to maintain the required water level in the vessel. This function can be performed with six ADS valve in service. Therefore, the ECCS function will be successfully performed with one ADS valve out of service. The analysis also supports one LLS valve to be inoperable for continuous operation. The LLS logic and instrumentation is described in the Technical Specification Bases and is designed to mitigate thrust loads on the SRV discharge lines by preventing subsequent

SRV actuations with an elevated water leg in the SRV discharge lines. The LLS logic will assign pre-set opening and closing setpoints to four pre-selected SRVs. These 4 SRVs are the remaining SRVs which do not perform an ADS function. The logic is armed when any one of the 11 SRVs is opened, as indicated by a signal from one of the redundant pressure switches located on its tailpipe, coincident with a high reactor pressure signal.

There is considerable redundancy in the SRV tailpipe pressure switches and reactor pressure instrumentation. In addition, the safety function can be accomplished with one LLS valve out of service. Therefore, continuous operation is permitted with one LLS valve out of service.

Summarizing, this study supports continuous operation with 1 SRV OOS in the safety/relief mode, 1 SRV OOS in its ADS mode and 1 SRV OOS in its LLS mode. [redacted]

1.4.5 New Nominal SRV Setpoints (Table 1-1, Item 8)

The nominal SRV setpoints are increased to 1150 psig to improve the SRV simmer margin. The proposed change will provide approximately 115 psid of simmer margin, which is consistent with recommendations made by GE and BWR Owners' Group. The new nominal setpoints are required to be less than the Upper Limit by at least 3%, the allowable drift tolerance.

Table 1-1

COMPARISON OF PRESENT TO PROPOSED PERFORMANCE REQUIREMENTS

Performance Requirement	Present Limit	New Limit
1. Opening pressure up to which the SRVs are capable of performing safety function.	Nominal +3%	Upper Limit (1195 psig)
2. Opening pressure for licensing basis analyses.	Nominal + 3%	Upper Limit (1195 psig)
3. Tolerance beyond which valve refurbishment and additional valve testing is required.	±3%	±3%
4. Tolerance on the as-left SRV setting prior to the valve being returned to service.	±1%	±1%
5. Number of SRVs permitted to be OOS for continued operation.	0	1
6. Number of ADS valves permitted to be OOS for continued operation.	0	1
7. Number of LLS valves permitted to be OOS for continued operation.	0	1
8. Nominal Setpoints (psig): Unit 1	4 at 1110 4 at 1120 3 at 1130	11 at 1150
Unit 2	4 at 1120 4 at 1130 3 at 1140	11 at 1150

2.0 ANALYSIS APPROACH

2.1 DISCUSSION OF ANALYSES

This report discusses analyses and evaluations for the following proposed pressure relief system performance requirements:

- New Nominal Setpoints
- SRV Opening Setpoint Tolerance ±3%
- Upper Limit (1195 psig)
- 1 SRV/ADS/LLS Valve OOS

The safety considerations for each of these requirements are examined at the most limiting conditions with respect to each issue. Consequently, safety concerns, where higher SRV opening pressures have the greatest impact (e.g., vessel overpressure margin, ECCS/LOCA performance, and containment response), are analyzed at the Upper Limit with 1 SRV OOS. For other concerns, such as thermal limits and HPCI and RCIC performance, the most limiting conditions are more dependent upon the SRV nominal setpoints. In this regard, the set of performance requirements which represents the most limiting conditions is analyzed, thereby bounding the less limiting performance requirements. Exceptions to this approach are noted within the applicable section.

2.2 ESTABLISHMENT OF AN UPPER LIMIT

The value of the Upper Limit is initially determined from the vessel overpressure analysis. [redacted]. This margin allows for variations in the peak vessel pressure which were calculated for past cycles (Table 3-1) and may be predicted for future fuel cycles. The value of the Upper Limit determined from the vessel overpressure analyses is then used as the SRV opening pressure when analyzing the remaining safety concerns.

2.3 DETERMINATION OF NOMINAL SETPOINTS, ALLOWABLE TOLERANCE, AND OPERATION WITH VALVES OOS

[redacted]

[redacted]

3.0 VESSEL OVERPRESSURE ANALYSIS

The ASME code requires peak vessel pressures to be less than the upset transient limit of 1375 psig during transient events. The limiting overpressure event is the main steam isolation valve (MSIV) closure with flux scram (Ref. 4). For this transient, a multiple component failure of the MSIV position scram signal is assumed. Therefore, the reactor scrams on a subsequent high neutron flux signal due to the collapse of voids following vessel pressurization.

Considering the proposed performance requirement changes, the greatest challenge to the ASME Upset Limit is provided by assuming that the SRVs have drifted up to the Upper Limit with 1 SRV OOS. [redacted]

[redacted] Table 3-1 presents historical information on the peak vessel pressure obtained for this event for previous cycles with 1% drift. It can be seen that there was a large margin to the ASME limit of 1375 psig and that the predicted peak vessel pressure remains relatively constant over the three fuel cycles indicated.

3.1 OVERPRESSURE ANALYSIS ASSUMPTIONS

The following assumptions and initial conditions were used in analyzing the MSIV closure with flux scram (MSIVF) for all cases:

- Reactor initially operating at 2485 MWt (102% of pre-uprate rated power),
- Conservative end-of-cycle (EOC) nuclear dynamic parameters used (Cycle 13, 14 for Unit 1, and Cycle 9, 10 for Unit 2)
- 1 SRV out-of-service,
- Recirculation pump trip (RPT) capability out-of-service.
- SRV opening pressure at Upper Limit

In addition, one case was repeated with a 5% increased core flow (ICF) to demonstrate that the appropriate margin to the ASME Upset Limit can be maintained with the plants operating in

that condition. It is conservative not to assume reduced feedwater temperature for an overpressure analysis.

Finally, two additional cases were run at power uprate conditions based on Unit 2, Cycle 12. The cases were run at ICF conditions, assuming all SRVs are at their Upper Limit with and without one SRV out of service. The initial (pre-transient) reactor operating pressure was also increased to the maximum allowed by the new Technical Specifications.

The GE thermal-hydraulic and nuclear kinetics coupled transient code, ODYN (Ref. 5), was used to obtain the system response and peak vessel pressure.

3.2 OVERPRESSURE ANALYSIS RESULTS

The reactor response for the two plants with the SRVs at the Upper Limit is shown in Figures 3-1 through 3-4. When both plants are analyzed with the same SRV setpoints (Upper Limit of 1195 psig), the pressure response is virtually the same. Figure 3-5 is the case with ICF. Note that for this event, operating with ICF has almost no effect upon peak vessel pressures.

The MSIVF event is initiated as the MSIVs begin to close and the pressure increases. The voids collapse, which produces a neutron flux increase. A scram signal on high flux occurs approximately 1.5 seconds into the transient. Vessel pressure continues to rise until the SRV opening pressures are reached. Subsequent SRV actuations terminate the pressurization transient.

Figures 3-6 and 3-7 demonstrate similar responses for 102% of power uprate and ICF conditions with and without one SRV out of service. The case with one SRV out of service (Figure 3-6) is 10 psi higher than the corresponding case performed at 102% of pre-uprate rated power (Figure 3-5). This increase is attributed primarily to the higher initial vessel pressure assumed in the uprate case. However, the limiting case at uprate conditions is still well below the ASME overpressure limit of 1375 psig. Table 3-2 shows the resultant peak vessel pressures for each case analyzed.

The value of the opening pressure Upper Limit is 1195 psig. This opening pressure maintains an approximate 50 psi margin to the ASME Upset Limit of 1375 psig, assuming one SRV-OOS for the MSIV closure with flux scram event. Therefore, each set of performance requirement changes can be implemented without impacting vessel overpressure safety concerns.

Table 3-1
PREVIOUS PEAK PRESSURE HISTORY FOR LIMITING OVERPRESSURE EVENT
(Nominal Settings +1%)

HATCH 1	
Cycle	Peak Vessel Pressure (psig)
12	1242
13	1244
14	1242
HATCH 2	
Cycle	Peak Vessel Pressure (psig)
8	1241
9	1245
10	1250

Note: This historical information is presented to show the relatively small variation in peak pressure from cycle to cycle.

Table 3-2
RESULTS OF VESSEL OVERPRESSURE PROTECTION EVALUATION

	Peak Vessel Pressure (psig)	
Case	Hatch 1	Hatch 2
1. Upper Limit w/1 SRV OOS, pre-uprate	1311 Cycle 13	1308 Cycle 9
2. Upper Limit w/1 SRV OOS, pre-uprate	1312 Cycle 14	1312 Cycle 10
3. Case 1 with ICF	1311 Cycle 13	
4. Case 1 with ICF, power uprate, 1058 psig initial dome pressure	-	1321 Cycle 13*
5. Case 4 with all SRVs in service	-	1280 Cycle 13*

* Simulated with Cycle 12 core.

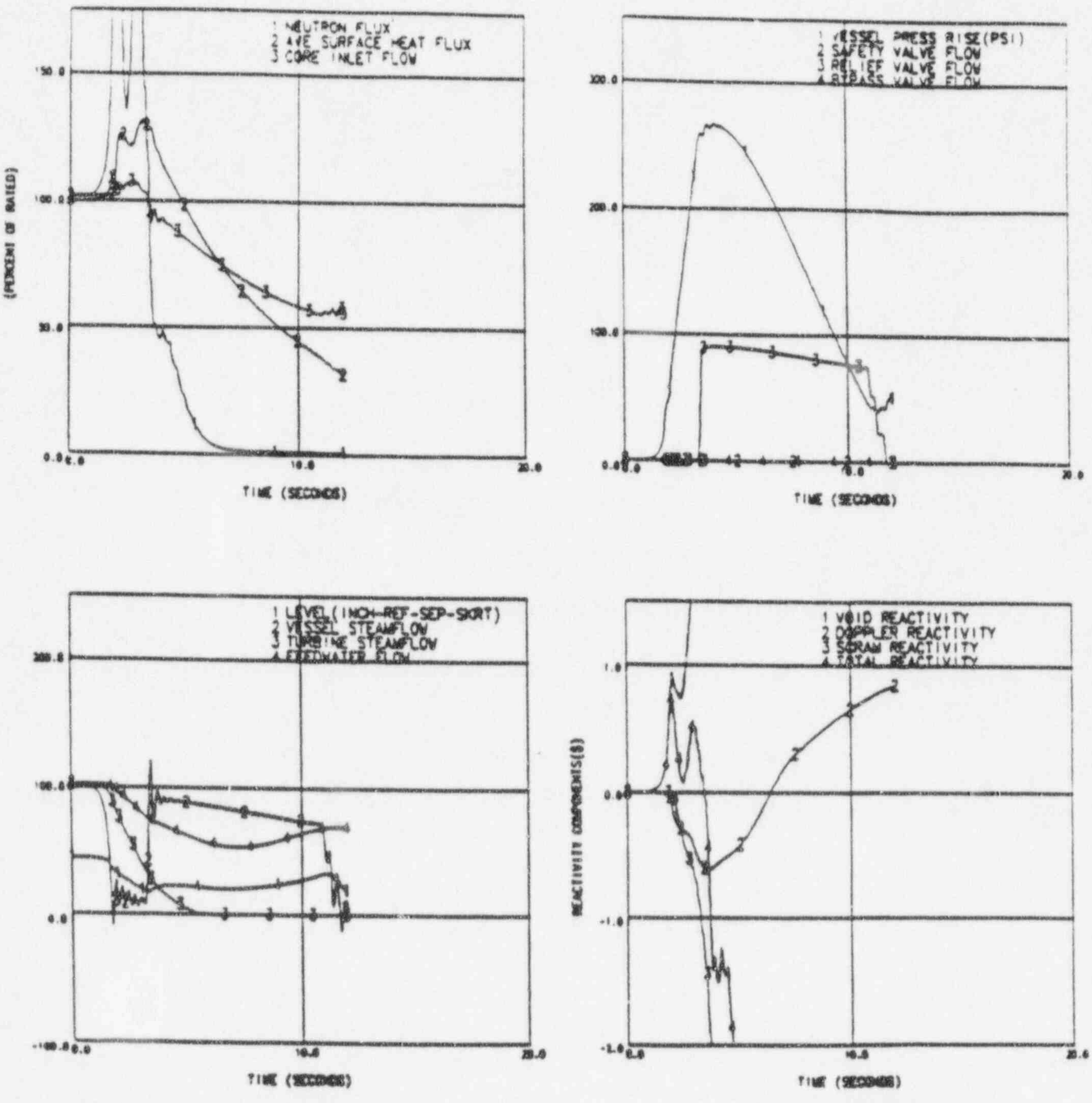

Figure 3-1. Hatch 1 - MSIV Closure, Flux Scram, 102% of Pre-Uprate Power, 100% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 13

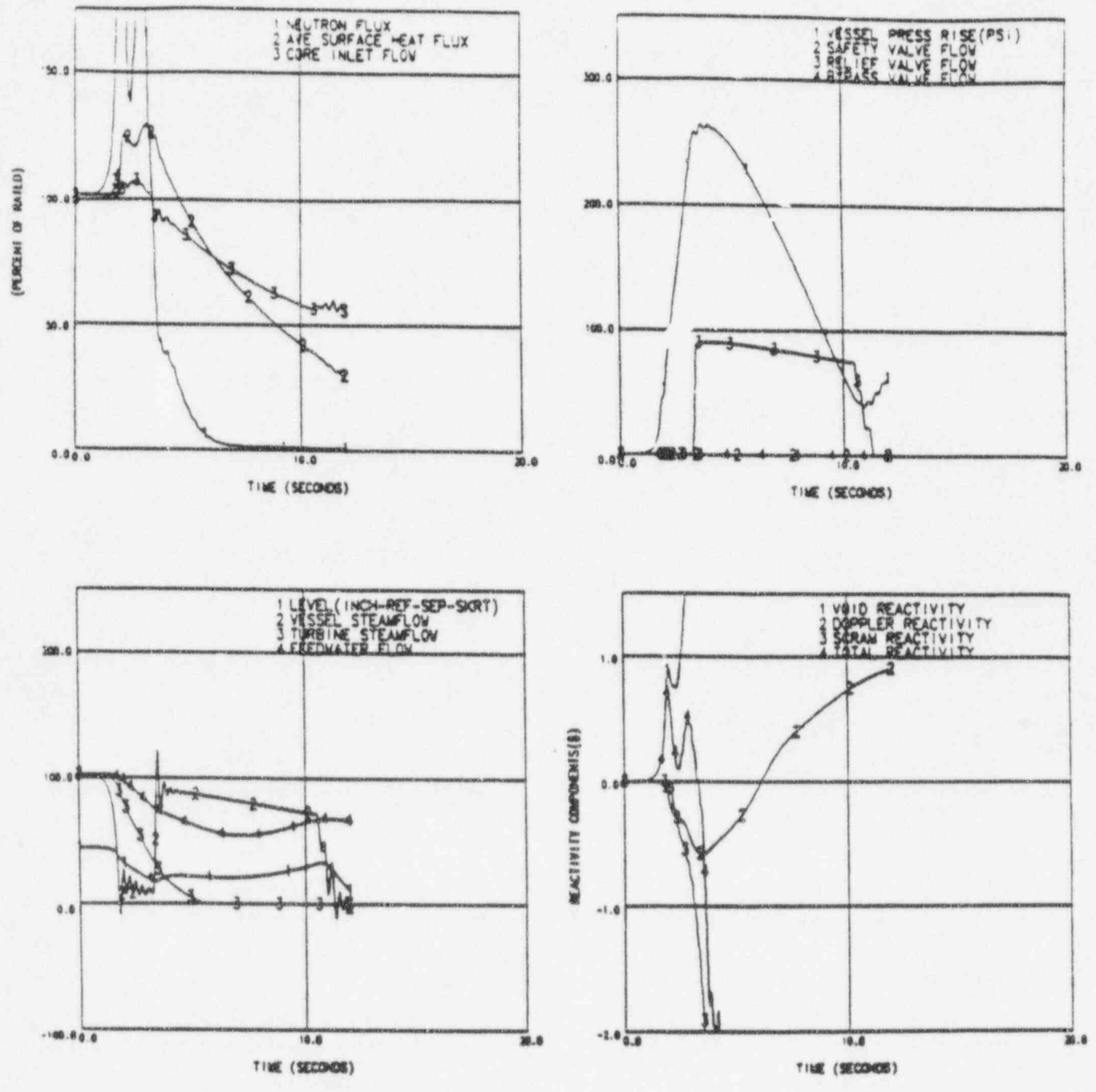

Figure 3-2. Hatch 1 - MSIV Closure, Flux Scram, 102% of Pre-Uprate Power, 100% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 14

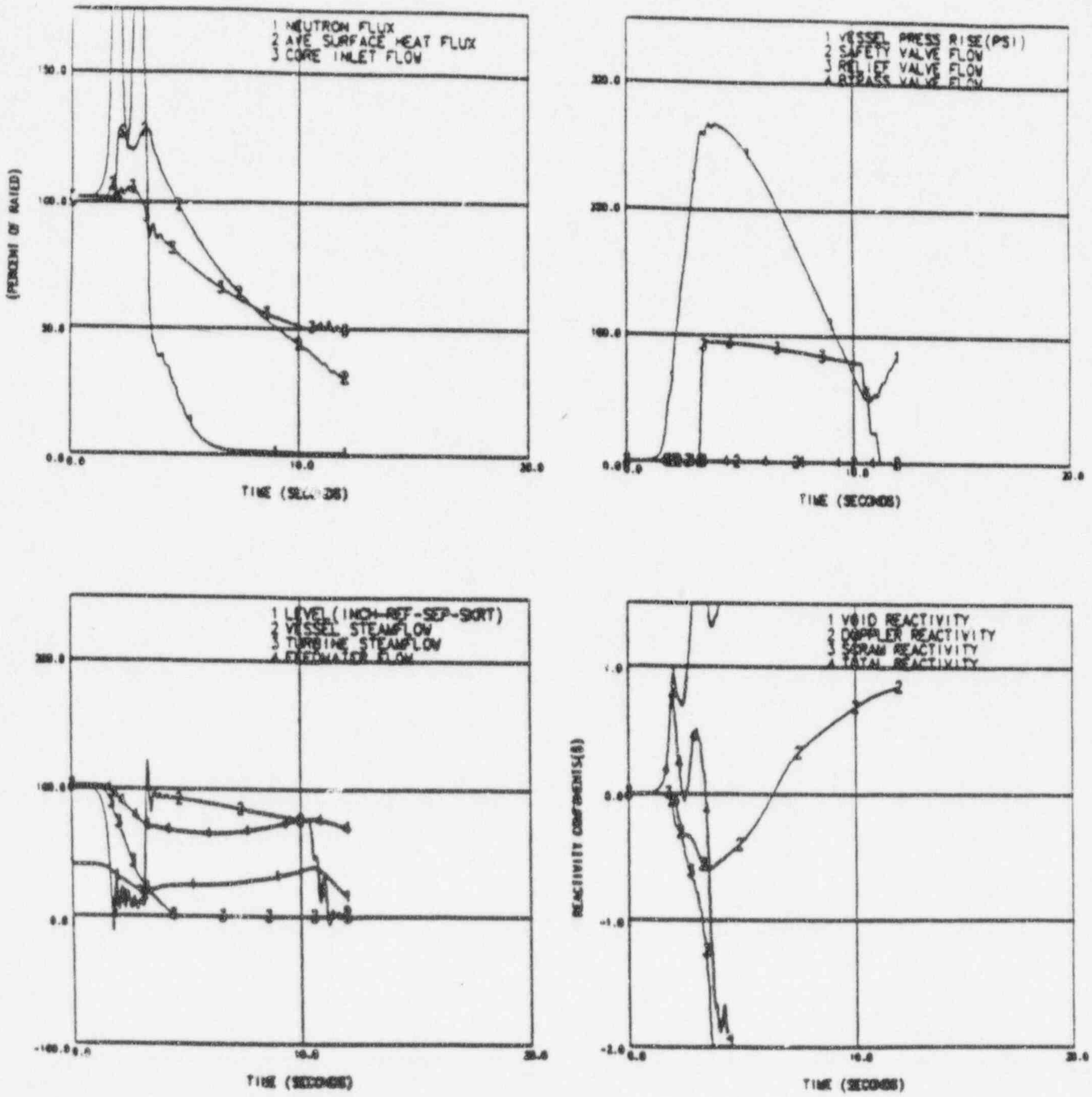

Figure 3-3. Hatch 2 - MSIV Closure, Flux Scram, 102% of Pre-Uprate Power, 100% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 9

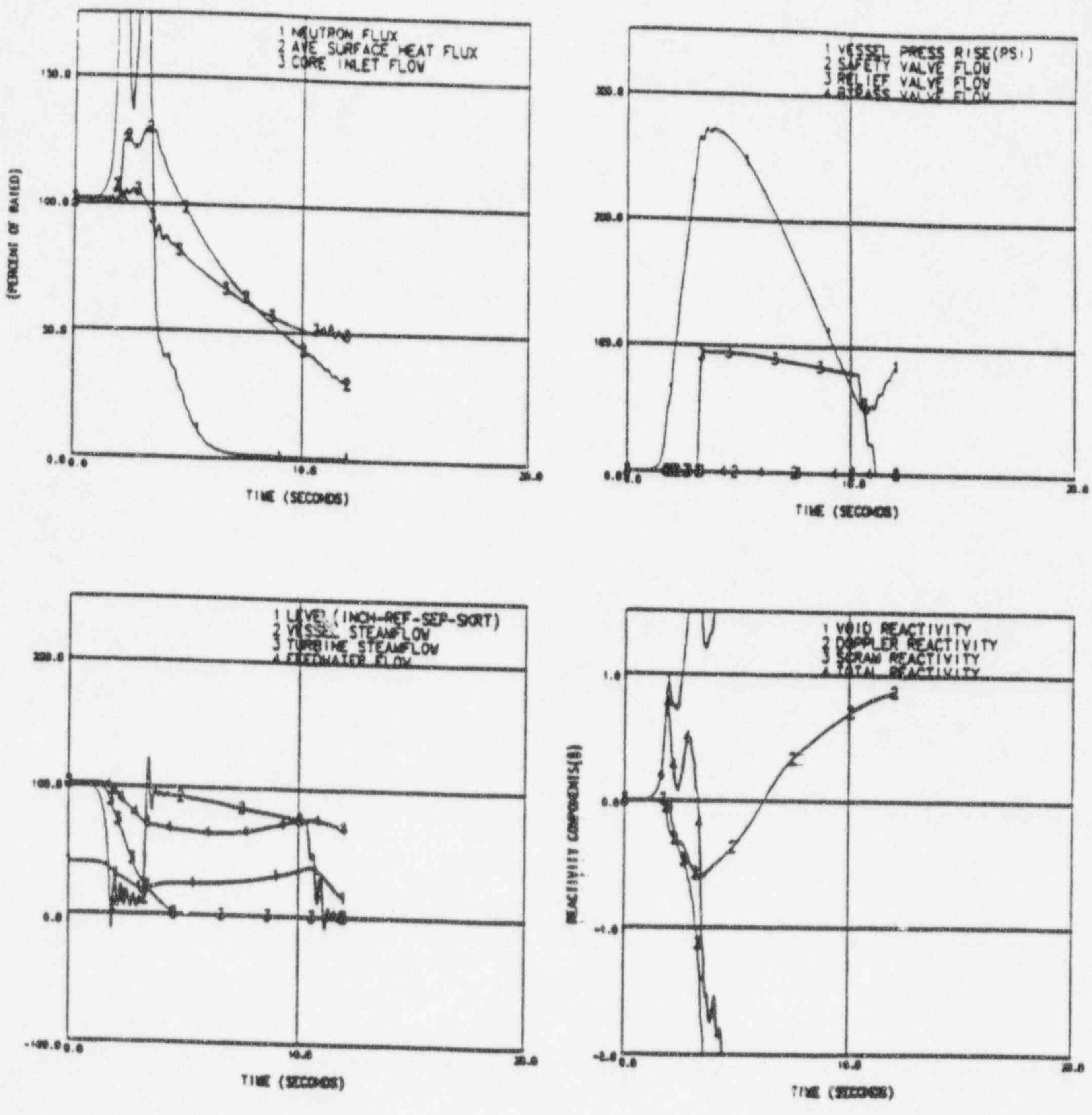

Figure 3-4. Hatch 2 - MSIV Closure, Flux Scram, 102% of Pre-Uprate Power, 100% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 10

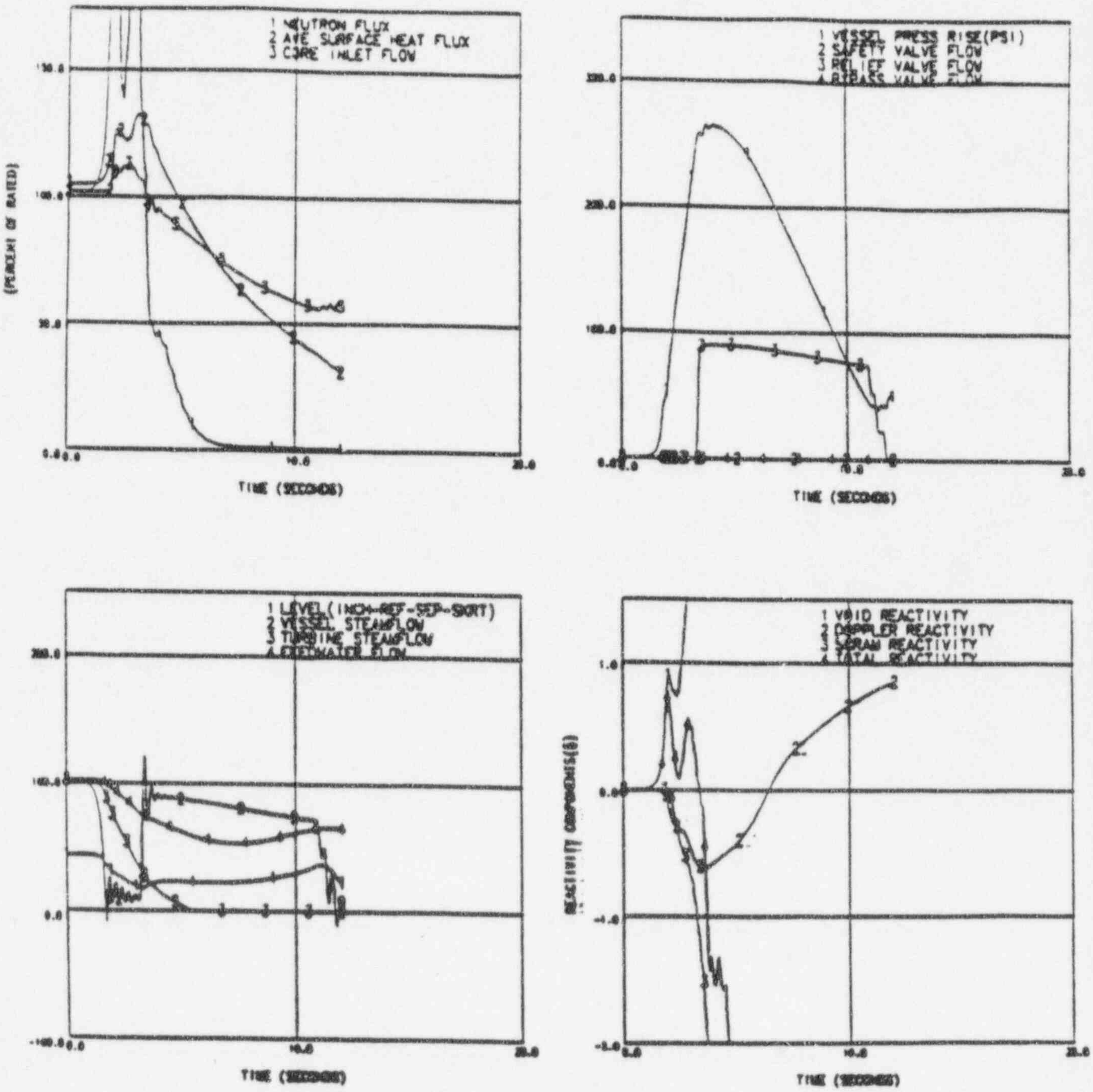

Figure 3-5. Hatch 1 - MSIV Closure, Flux Scram, 102% of Pre-Uprate Power, 105% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 13

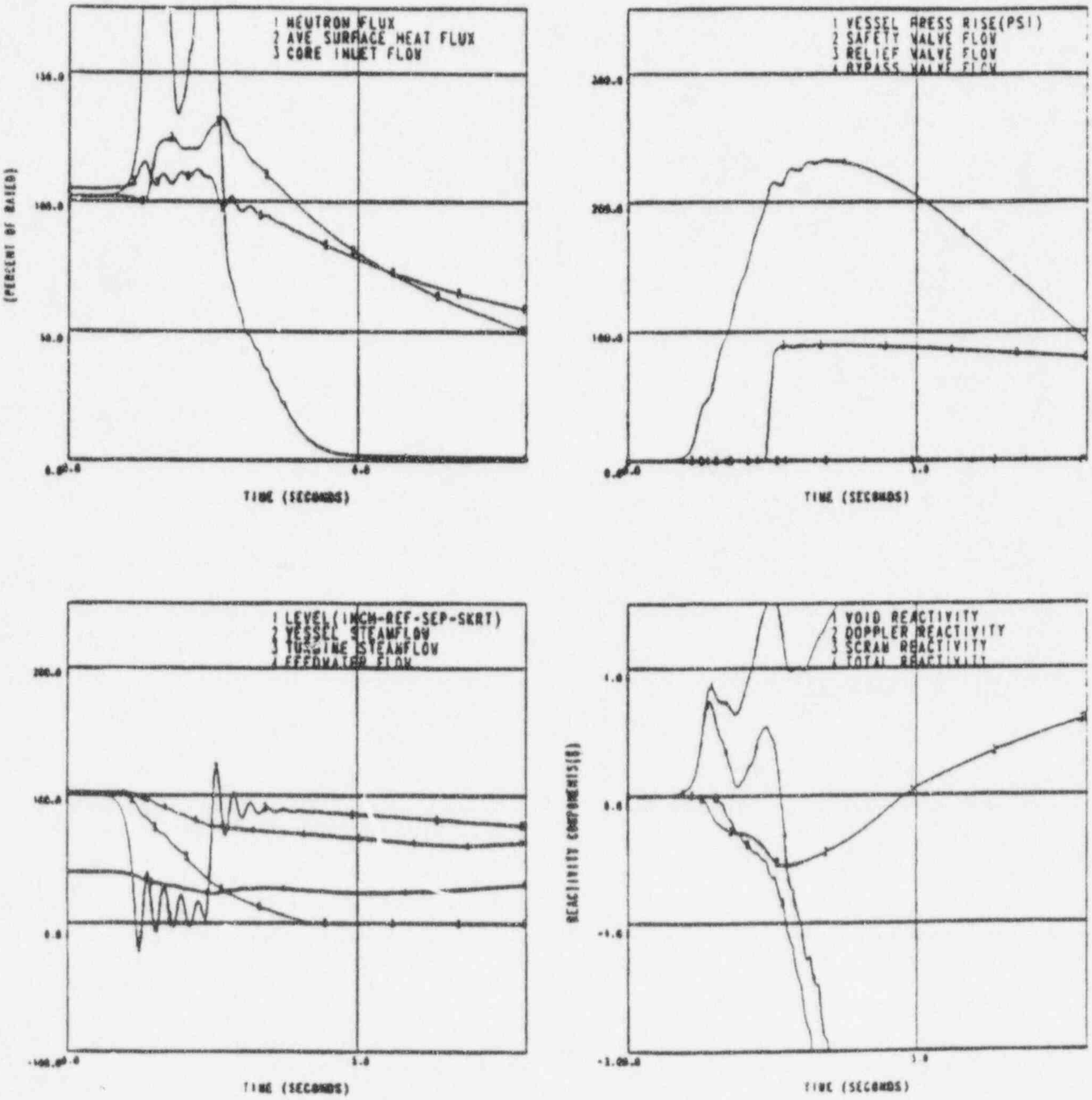

Figure 3-6. Hatch 2 - MSIV Closure, Flux Scram, 102% Power, 100% Core Flow at EOC without RPT, Upper Limit - 1 SRV OOS, Cycle 13 (Simulated with Cycle 12 Core)

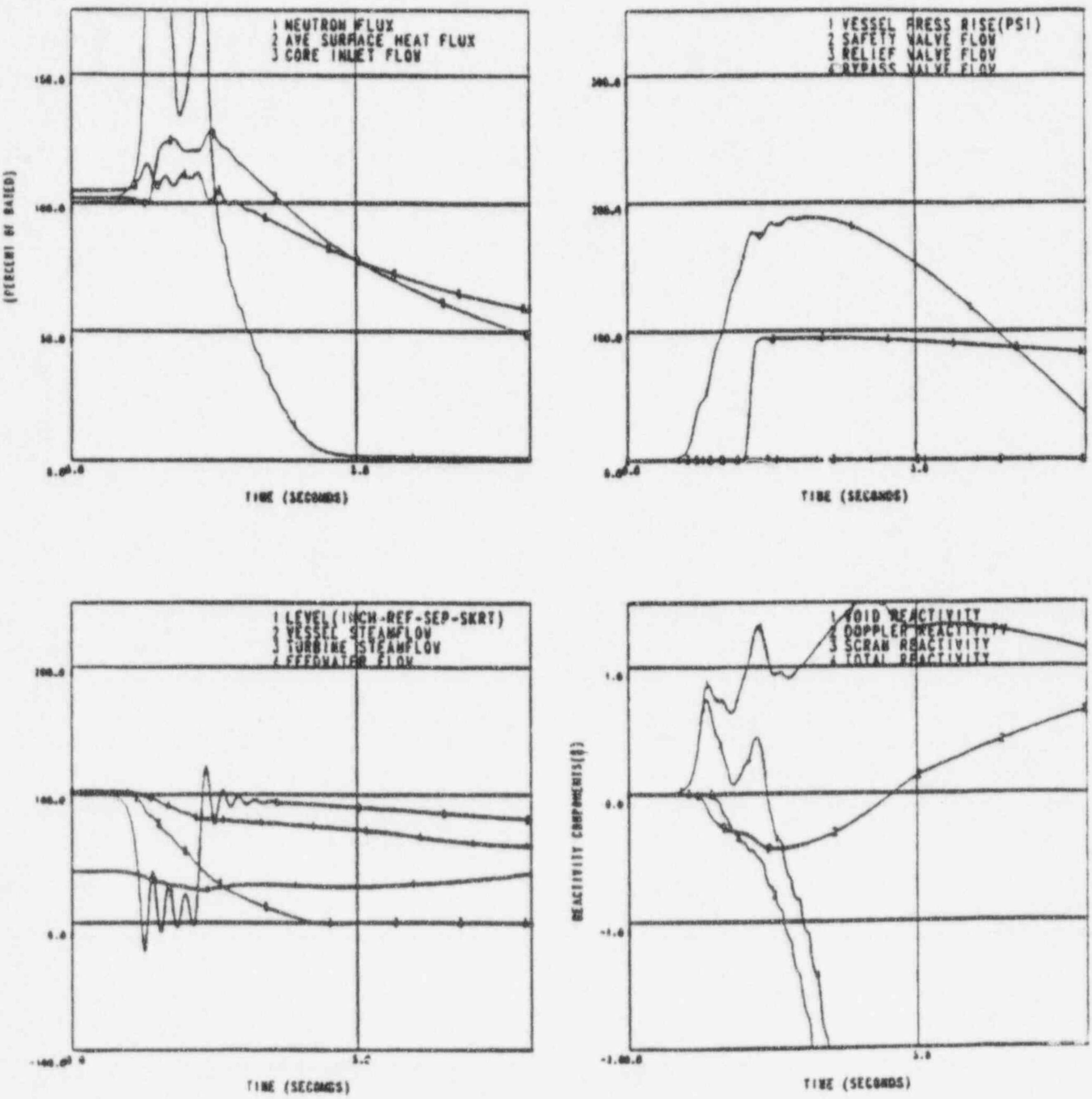

Figure 3-7. Hatch 2 - MSIV Closure, Flux Scram, 102% Power, 100% Core Flow at EOC without RPT, Upper Limit, Cycle 13 (Simulated with Cycle 12 Core)

4.0 THERMAL LIMITS

Important in the study to increase the tolerance on SRV opening setpoints are the effects on the reactor fuel of pressurization transients. Surveys have been done, therefore, to determine the transients which have the greatest impact on fuel thermals limits. The most limiting abnormal operational occurrence for both units was found to be the generator load rejection without bypass (LRNBP) (Refs. 6-11). This event was used to determine the most operationally limiting thermal limits for the core. The minimum critical power ratio (MCPR) is the most significant thermal limit for evaluation.

[redacted]

[redacted]

[redacted]

The phenomenon described above is confirmed by the transient analysis performed for the proposed SRV setpoints at the new nominal settings to demonstrate that the SRVs at the increased opening pressures actuate after the peak surface heat flux. The results can then be applied to higher SRV opening pressures (i.e., the Upper Limit). The Hatch 1 response to an LRNBP is also representative of Hatch 2. The results of these analyses are discussed in a following section.

4.1 THERMAL LIMITS ASSUMPTIONS

The following assumptions and initial conditions were used in analyzing the effect of the new setpoints listed in Table 1-1:

- Reactor initially at 100% of pre-uprate rated power,
- Conservative end-of-cycle nuclear dynamic parameters for Cycle 13 and 14, Hatch 1,

- One SRV OOS,
- Increased core flow assumed,
- Reduced feedwater temperature not assumed,
- RPT feature in service.

The GE thermal-hydraulic and nuclear kinetics coupled transient code, ODYN, was used to model the reactor response to the LRNBP.

4.2 THERMAL LIMITS RESULTS

[redacted]

SRV opening pressures higher than the current nominal setpoint case will actuate later for the same event. Therefore, the results above also demonstrate that SRV opening pressures at either the new nominal or the Upper Limit setpoints have no impact on thermal limits for either the present or past cycles.

Table 4-1
RESULTS OF THERMAL LIMITS EVALUATION FOR THE LIMITING EVENT (LRNBP)

	Cycle 13	Cycle 14
Peak Heat Flux (% rated at sec)	117.8 at 0.89	116.8 at 0.89
SRV Actuation (sec)	1.33	1.34
Peak Vessel Pressure (psig at sec)	1215 at 2.03	1214 at 2.09
Peak Dome Pressure (psig at sec)	1186 at 2.07	1185 at 2.07

Note: The results for Hatch 1 are shown above to demonstrate that SRV actuation occurs after peak heat flux. It is noted that the peak heat flux occurs approximately at the same time as MCPR.

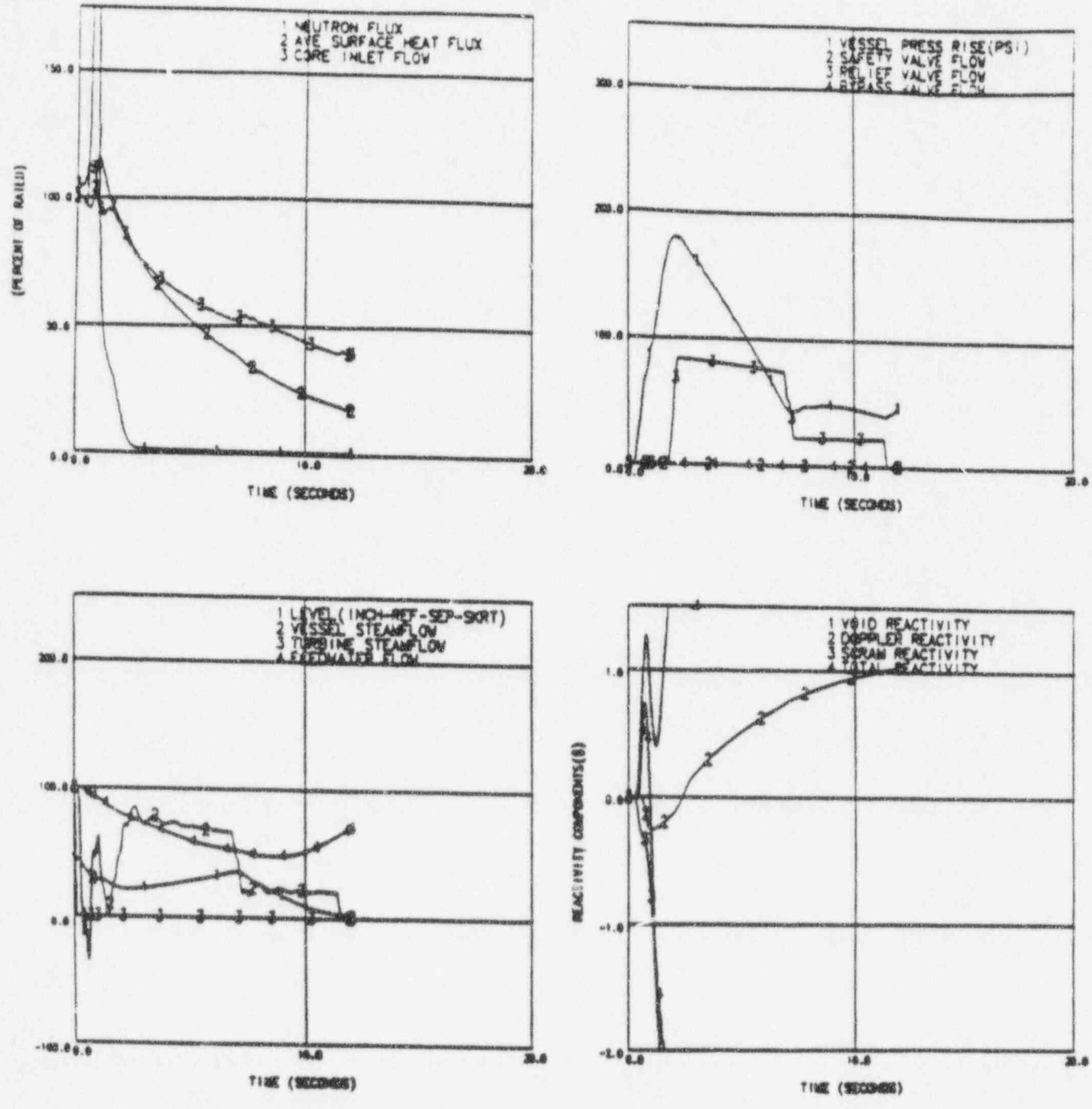

Figure 4-1. Hatch 1 - Generator Load Rejection without Bypass, 100% of Pre-Uprate Power, 105% Core Flow at EOC, with RPT, New Nominal Setpoints - Cycle 13

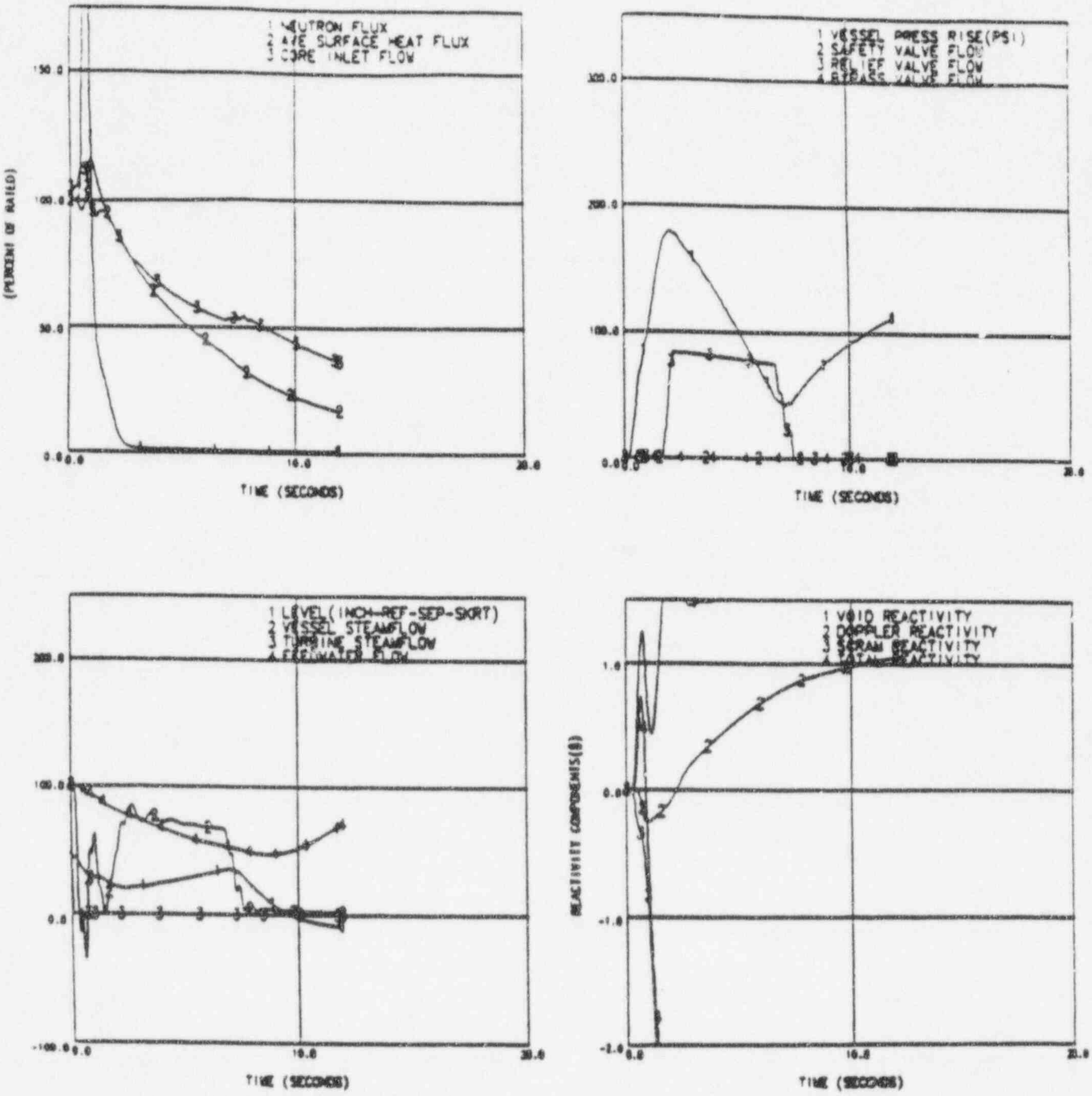

Figure 4-2. Hatch 1 - Generator Load Rejection without Bypass, 100% of Pre-Uprate Power, 105% Core Flow at EOC, with RPT, New Nominal Setpoints - Cycle 14

5.0 ECCS/LOCA PERFORMANCE EVALUATION

The ECCS/LOCA (SAFER/GESTR) analysis (Ref. 12) has been reviewed to determine the impact of an increase in the SRV opening pressures and 1 SRV OOS. The SRV opening pressures were assumed to drift up to the Upper Limit value of 1195 psig. The results of the review also apply to lower opening pressures (i.e., the new nominal opening setpoints).

The ECCS is designed to protect fuel integrity by limiting the core response parameters below the requirements of 10CFR50.46, which specifies that the calculated peak cladding temperature (PCT) may not exceed 2200°F. A change in the SRV opening pressure can only affect the PCT for pipe break events in which SRV actuations occur. Likewise, the assumption of 1 ADS/SRV OOS need only be assessed for those events where ADS/SRVs are actuated. A review of the SAFER/GESTR analysis, referenced above, identified those breaks that are potentially impacted by the proposed changes. They are evaluated in the following sections to demonstrate that impact on PCT will not be sufficient to challenge the limiting break PCT. The following breaks are addressed:

- DBA (design basis accident), or limiting break LOCA, which is a recirculation line break with a battery failure,
- Small break LOCA (specifically those with ADS/SRV actuations),
- Steamline break outside of the containment.

5.1 LIMITING BREAK LOCA

[redacted] Typically, this type of event depressurizes very rapidly through the large break, never experiencing pressures within the operating range of the SRVs nor requiring ADS. Consequently, there is no impact on the limiting PCT due to the proposed changes, and this event need not be reanalyzed.

5.2 SMALL BREAK LOCA

An examination of the small breaks analyzed in the SAFER/GESTR study shows which breaks have SRV or ADS actuations and thus, are potentially impacted. Two cases are considered: a 0.05 ft^2 and a 0.10 ft^2 break in a recirculation discharge line. These are breaks so small that vessel depressurization does not occur until inventory depletes to the water level setpoint for MSIV closure. Vessel isolation follows, causing pressurization to the SRV opening

setpoints, as in the case of the 0.05 ft^2 break. ADS valves may actuate before the actuation of the non-ADS SRVs, depending on the rate the water is dropping to the ADS (Level 1) initiation setpoint. [redacted]

5.2.1 Impact of Upper Limit SRV Setpoints on Recirculation Line Breaks

[redacted]. If SRV actuation was delayed an additional 50 seconds (a conservative estimate of the additional time to the 1195 psig Upper Limit setpoint), there would still be no heatup. Also, ADS would occur at about the same time, providing depressurization to the low pressure ECCS initiation setpoints, should they be required.

5.2.2 Impact of One ADS/SRV OOS

[redacted] An assumed ADS/SRV OOS would slightly delay the onset of low pressure ECCS initiations. However, for small breaks such as this, if the core uncovers, the resulting heatup rate is mild (~1 or 2 degrees per second). Considering a margin of approximately 400 degrees to the limiting PCT, it is not feasible that the small delay in the onset of low pressure ECCS operation due to decreased ADS capacity would cause any significant increase in PCT to challenge the present limiting PCT.

5.3 STEAMLINE BREAK OUTSIDE CONTAINMENT

For non-recirculation breaks, the one of specific interest is a break in a main steamline downstream of the inboard MSIVs. High steamline flow would initiate MSIV closure, which would isolate the vessel and terminate steam flow out the break. The vessel would pressurize to the SRV setpoints. The SRVs would cycle open and closed, gradually depleting inventory. Actuation of ADS by the Level 1 setpoint would terminate the SRV cycling and depressurize the vessel to the range of the low pressure ECCS. Inventory depletion could uncover the core, resulting in a mild core heatup.

[redacted]

5.4 IMPACT ON LLS

Regarding LLS valve operation, the increased SRV opening pressures will only affect the timing of first SRV actuations. Once the logic is initiated, the opening and closing setpoints of pre-selected SRVs are automatically reset to lower values by the LLS logic. This logic is unaffected by the setpoint tolerance change or the Upper Limit since the logic acts on the relief mode of SRV actuation and not on the safety mode of operation.

5.5 CONCLUSIONS FOR ECCS/LOCA EVALUATIONS

Based on the above discussion, SRV actuations at the Upper Limit and 1 ADS/SRV OOS have no impact on the limiting LOCA analyses and have only a negligible impact upon the PCT for the specific cases which experience ADS/SRV actuations. Because the large recirculation line break is unaffected by the SRV setpoints and is the most limiting event in terms of regulatory limits, the plant licensing basis PCT remains unchanged.

6.0 HIGH PRESSURE SYSTEM PERFORMANCE

HPCI and RCIC performance were evaluated for SRV setpoint drift to the Upper Limit value of 1195 psig. Operation at the Upper Limit provides a greater challenge to the HPCI and RCIC piping, pumps, and turbines than SRVs at the new nominal setpoints. These evaluations assure satisfaction of performance requirements for operation at both the Upper Limit and at the new nominal setpoints since operation at the Upper Limit bounds operation at the proposed nominal setpoints.

Both HPCI and RCIC systems are important in mitigating reactor vessel isolation and loss of feedwater events. The HPCI system is important in LOCA evaluations, whereas the RCIC system is not since no credit is given for RCIC operation in LOCA analyses. (Note that the Reference 12 LOCA analysis does not take credit for either HPCI or RCIC.)

6.1 IMPACT OF HIGHER SRV SETPOINTS ON HPCI AND RCIC PERFORMANCE

[redacted]

[redacted]

The potential concern during the startup transient is system availability. If the HPCI and RCIC turbines do trip during the startup, manual actions are required to reset the turbine trips. For HPCI, the turbine can be reset in the control room. For RCIC, the turbine must be reset locally.

The above considerations assume that HPCI and RCIC would initiate and operate when the reactor pressure is at the Upper Limit. This is a highly unlikely scenario. First, there is a very low probability that all eleven SRVs will drift high to their upper limits. Second, the non-safety electronic actuation logic described in Section 1.3.4 would have to fail. This system is redundant, highly reliable and will initiate SRV actuations at 1120 psig. Thirdly, in more probable sequences of events for pressurization transients, SRVs would actuate before HPCI and RCIC initiate. This reduces vessel pressure to a value well within the operational ranges of these systems. After first actuation of the SRVs, the LLS system will maintain vessel pressure in the worst case at no more than 1050 psig (see LLS setpoints in Section 1.3.4), which is readily within the discharge capabilities of the HPCI and RCIC systems. In cases where RCIC and/or HPCI initiate before first actuation of the SRVs, operation of these systems at high pressures would be for a relatively short period, i.e., until pressure increases to the point where SRVs actuate at the Upper Limit, and then pressures would be reduced. And as indicated above, LLS would then maintain vessel pressures at values below 1050 psig. It should be noted that actuations of SRVs in the worst case at the Upper Limit and the following operation of the LLS valves provide conservatisms which ensure appropriate performance of the HPCI and RCIC systems for the conditions examined in this report.

6.2 HPCI AND RCIC PERFORMANCE FOR LOSS-OF-FEEDWATER EVENTS

[redacted]

6.3 HPCI PERFORMANCE FOR LOCA EVENTS

[redacted]

[redacted]

The discussions above demonstrate that SRV setpoint drift up to the Upper Limit has an insignificant impact on HPCI and RCIC performance.

7.0 CONTAINMENT EVALUATION

7.1 CONTAINMENT PRESSURE AND TEMPERATURE RESPONSE

The impact of an increase of the SRV opening pressures up to the Upper Limit of 1195 psig on the containment response was assessed. The impact of SRV opening pressures at the Upper Limit bounds the impact of SRV opening pressures at the new nominal setpoints. Therefore, the results of this assessment apply to all of the performance requirement changes.

The effects on the peak drywell pressure and temperature response for the respective limiting events were considered. The most severe event in terms of peak drywell pressure is the design basis LOCA, a double-ended guillotine break of the recirculation line. An increase in SRV opening pressures has no effect on this event because the vessel rapidly depressurizes without any SRV actuations (see LOCA discussion in Section 5.1). Therefore, there is no impact on the peak drywell pressure.

The most severe event in terms of peak drywell temperature is a steamline break inside the drywell. However, the steamline break that produces the limiting peak drywell temperature is large enough to depressurize the vessel through the break, without requiring SRV actuations. Therefore, the limiting peak drywell temperature event is insensitive to an increase of the SRV opening pressures.

For smaller steamline breaks that require SRV actuations, the resultant drywell temperatures are well below that of the limiting steam line break. Furthermore, the peak drywell temperatures occur late in the event following many SRV actuations. The peak drywell temperature is governed by the total energy released into the drywell. Since the SRVs will return to the nominal setpoints following the first actuation, an increase in the SRV opening pressures will only affect the very beginning of the event that will have a negligible impact on the total energy released to the drywell. Therefore, an increase in the SRV opening pressures will also have an insignificant impact on the peak drywell temperature for the non-limiting drywell temperature events.

7.2 CONTAINMENT INTEGRITY

The Plant Unique Analysis Report (PUAR) for Hatch Units 1 and 2 (Refs. 14 and 15) and other applicable documents (Refs. 16 and 17) have been reviewed to determine the impact of an increase in SRV opening pressures on the containment. This review considered the known conservatisms in the forcing function used as input to the PUAR structural calculations and used PUAR documented margins to allowable stresses where available.

The PUAR was reviewed to determine the impact of SRV setpoint drift up to the Upper Limit of 1195 psig. That report indicates that the analyses bound SRV opening pressures at

current nominal SRV setpoints + 3%. Therefore, the results of this review are applicable to the performance requirement changes.

7.2.1 Components Evaluated

When an SRV actuates, pressure and thrust loads are exerted on the SRV discharge piping and T-quenchers. In addition, the expulsion of water and then air into the suppression pool through the T-quenchers results in pressure loads on the submerged portion of the torus shell and drag loads on submerged structures. These SRV discharge loads have the potential to be affected by an increase in the SRV setpoints due to an increase in SRV flow rates. Reference 16, the Mark I Containment Program Load Definition Report (LDR), summarizes the following SRV discharge loads:

- Thrust loads on SRV discharge piping and T-quenchers,
- Torus shell pressures,
- Water jet loads and air bubble induced drag loads on submerged structures.

[redacted] The PUAR determines the limiting load combination, the actual load, and the allowable load for a given structure. The evaluation performed here assesses the impact of increased SRV opening pressures on the actual load for the limiting load combination on a structure-by-structure basis, as addressed in the PUAR.

7.2.2 Methods of Evaluation

[redacted]

[redacted] The minimum nominal SRV opening setpoint pressure was 1080 psig for Hatch Unit 1 and 1090 psig for Hatch Unit 2 when the PUAR analysis was performed. The corresponding increase in the SRV loads due to an increase in the SRV open pressure to 1195 psig is approximately 10.5% for Hatch Unit 1 and 9.5% for Hatch Unit 2.

7.2.3 Results of Evaluations

Thrust loads on SRV piping and T-quenchers were determined in References 14 and 15 using the RVFOR computer model. [redacted]

[redacted]

[redacted]

Based on the discussion above, an increase in SRV opening pressures up to the Upper Limit does not result in the allowable containment loads being exceeded. Therefore, SRV setpoint drift up to the Upper Limit has no significant impact on containment integrity.

8.0 ATWS MITIGATION CAPABILITY

The potential impact of the proposed changes to the SRV performance requirements (Table 1-1) on the ATWS performance is an increase in the peak vessel bottom pressure during the vessel overpressure event due to one SRV being out of service. It is necessary to demonstrate compliance with the ASME vessel overpressure criterion of 1500 psig with one SRV-OOS.

In Reference 18, the limiting ATWS event (MSIV closure) was analyzed for the current power conditions with the SRV relief setpoints which are the same as the electronic relief settings described in Section 1..3.4 of this report. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Therefore, it is concluded that the ATWS performance with one SRV OOS is acceptable for Hatch.

9.0 CONCLUSIONS

Based on this study, the following pressure relief system performance requirement changes have no significant safety impact on vessel overpressure margin, thermal limits, ECCS/LOCA performance, HPCI and RCIC operability, containment response, or ATWS mitigation.

Hatch 1	Hatch 2
Nominal SRV Settings 11 at 1150 psig	Nominal SRV Settings 11 at 1150 psig
SRV Opening Tolerance at ±3%	SRV Opening Tolerance at ±3%
Upper Limit at 1195 psig	Upper Limit at 1195 psig
1 SRV/ADS/LLS Valve OOS	1 SRV/ADS/LLS Valve OOS

Analyses were performed to examine cycle-dependent safety concerns, i.e., vessel overpressure margin and thermal limits. It was demonstrated that SRV setpoint drift up to the Upper Limit value of 1195 psig during the previous cycles has no significant impact upon plant safety. For future cycles, it is recommended that the reload licensing evaluations verify the cycle-specific applicability of these analyses. Evaluations for ECCS/LOCA, high pressure systems performance, containment response and ATWS mitigation need not be re-performed for future cycles, as long as the above SRV performance requirements are met.

10.0 REFERENCES

1. Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Technical Specifications.

2. W. J. Roit (GE) to John Dale (BWROG Projects), "Applicability of SIL 196 Including Supplements 1 to 16, to the Target Rock 2-Stage SRV," dated May 11, 1995.

3. ANSI/ASME OM-1-1981, as referenced in Subsection IWV-3500 of the ASME Code, Section XI, 1986 Edition.

4. Edwin I. Hatch Units 1 and 2 Updated Final Safety Analysis Report.

5. Qualification of the One-Dimensional Core Transient Model for BWRs, NEDO-24154, Vols. 1 and 2, and NEDE-24154P, Vol. 3, October 1978.

6. Supplemental Reload Licensing Report For Unit 1, Reload 11/Cycle 12, 23A5939, Rev. 0, GE Nuclear Energy, September 1988.

7. Supplemental Reload Licensing Submittal For Unit 1, Reload 12/Cycle 13, 23A6504, GE Nuclear Energy, April 1990.

8. Supplemental Reload Licensing Submittal For Unit 1, Reload 13/Cycle 14, 23A7131, GE Nuclear Energy, September 1991.

9. Supplemental Reload Licensing Report For Unit 2, Reload 7/Cycle 8, 23A5884, Rev. 0, GE Nuclear Energy, January 1988.

10. Supplemental Reload Licensing Submittal For Unit 2, Reload 8/Cycle 9, 23A6470, GE Nuclear Energy, November 1989.

11. Supplemental Reload Licensing Submittal For Unit 2, Reload 9/Cycle 10, 23A6549, GE Nuclear Energy, April 1991.

12. Edwin I. Hatch, Units 1 and 2 SAFER/GESTR-LOCA Analysis, GE Nuclear Energy, NEDC-31376P, December 1986.

13. Nuclear Regulatory Commission Generic Letter 89-10, June 28, 1989.

14. Plant Unique Analysis Report, E. I. Hatch Nuclear Plant Unit 1, Rev. 2, December 1989, Docket 50-321.

15. Plant Unique Analysis Report, E. I. Hatch Nuclear Plant Unit 2, Rev. 2, December 1989, Docket 50-366.

16. NEDO-21888, Mark I Containment Load Definition Report, November 1981.

17. NEDE-21878P, Mark I Containment Program Analytical Model for Computing Air Bubble and Boundary Pressures Resulting from an SRV Discharge through a T-Quencher Device, Task 7.1.1.2, January 1979.

18. GE-NE-A00-05389-06, "Power Uprate Task Report for Edwin I. Hatch Plant Units 1 and 2 Anticipated Transients Without Scram (ATWS) Analysis," February 1995.

19. NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG 0460 Alternate No. 3)," December 1979.

Enclosure 6

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Safety/Relief Valve Setpoint Change

GE Affidavit for NEDC-32041P

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report GE-NEDC-32041P, Revision 2, Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2, Updated Safety/Relief Valve Performance Requirements, Class III (GE Company Proprietary Information), dated April 1996. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of the loss-of-coolant accident for the BWR.

The development and approval of the BWR loss-of-coolant accident analysis computer codes used in this analysis was achieved at a significant cost, on the order of several million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 29TH day of APRIL 1996.

David J. Robare
David J. Robare
General Electric Company

Subscribed and sworn before me this 29th day of April 1996.

PAULA F. HUSSEY
COMM. #1046120
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires DEC 1,1998

Paula F. Hussey
Notary Public, State of California