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GENERAL ELECTRIC BOILING WATER REACTOR INCREASED SAFETY/RELIEF VALVE SIMMER MARGIN ANALYSIS FOR PILGRIM NUCLEAR POWER STATION UNIT 1

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1. INTRODUCTION AND SUMMARY

Steam leakage past the pilot disk of two-stage Target Rock safety/relief valves may result in degradation of valve performance. Extended operation with leakage may severely erode the disk/seat.

Operating data demonstrates that an increase in the valve simmer margin (difference between normal plant operating pressure and the valve set point) will reduce the probability of pilot valve leakage. Increasing valve set points, however, requires an investigation of the impact on plant transient and accident response.

A study was performed for Pilgrim to optimize the simmer margin without imposing additional restrictions on plant operation. The results of this study concluded that the operating limits derived for Pilgrim Cycle 6 operation are still valid for a proposed 30 psi increase in safety/relief valve (S/RV) set point and a 10 psi decrease in the reactor nominal operating dome pressure. The following are the reload 5 analysis and revised set pressures:

	Dome Pressure (psig)	S/RV Set Point (psig)	Number of S/RVs
Reload 5 Analysis	1034	1095 + 1%*	4
Revised	1024	1125 + 1%*	4

*A 1% increase in set pressure is normally added to the nominal S/RV set pressure in the analysis to cover the set point uncertainty.

2. SAFETY ANALYSIS

2.1 INTRODUCTION

The safety analysis for Pilgrim Cycle 6 provided in Reference 1 is used as the base case for determining which transients are limiting. The increase in SRV set point affects only those events which result in valve actuation to limit the system pressure. The selection of the limiting transients is consistent with previous reload analyses and is not expected to significantly change during plant life. The results of this analysis will be verified for subsequent cycles by using the revised plant parameters in the reload licensing analyses.

The events to be considered are as follows:

1. Load rejection without bypass for minimum critical power ratio (MCPR) evaluation;

2. Main steam isolation valve (MSIV) closure flux scram for vessel overpressure protection evaluation; and

3. Loss of coolant accident (LOCA-small break) for peak cladding temperature (PCT) evaluation.

In addition, the capability of the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system were reevaluated considering the increase in SRV set point.

The results of the analysis which demonstrate the acceptability of the increased simmer margin are given below. All analyses were performed using the same input parameters as documented in Refererence 1, with the exception of the new parameters displayed in Table 1. The dome pressure has been reduced by 10 psi relative to the cycle 6 calculation in order to increase the S/RV simmer margin. The turbine pressure has been decreased accordingly to yield

the same rated steam flow as reported in Reference 1. The slight increase in S/RV capacity is due to the increase in mass flow rate as a result of the higher pressure at the valve set point.

2.2 GENERATOR LOAD REJECTION WITH BYPASS FAILURE

This transient produces the most severe reactor isolation event described in Reference 2. Fast closure of the turbine control valve (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The TCV is required to close as rapidly as possible to prevent overspeed of the turbine-generator (T-G) rotor. This closing, concurrent with the failure of the bypass valve system, causes a sudden reduction in steam flow which results in a nuclear system pressure increase. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by a scram initiated from closure of the TCV and by a void increase after the S/RV have automatically opened on high pressure. The time response plot for this transient based on the new S/RV set point is shown in Figure 1. The data summary is given in Table 2.

The peak steam line pressure is increased by 2 psi to 1284 psig compared with the cycle 6 calculations as a result of the higher pressure actuation of the S/RV and lower reactor dome pressure. The change in the S/RV set point combined with the decrease in dome pressure yielded an increase of only 0.1% in peak heat flux which has no significant effect on the critical power ratio reported in Reference 1. Therefore, the operating MCPR limits established in Reference 1 are acceptable with the new set points.

2.3 VESSEL OVERPRESSURE PROTECTION EVALUATION

The pressure relief system must prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude the uncontrolled release of fission products. For the Pilgrim Nuclear Power Station, the pressure relief system includes two spring safety valves (SSV) and four dual function S/RV. These valves provide the capacity to limit nuclear system overpressurization.

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure:

1. A peak pressure of 110% of the vessel design pressure is allowed (1375 psig for a vessel with a design pressure of 1250 psig).

2. The lowest qualified safety valve set point must be at or below vessel design pressure.

3. The highest safety valve set point must not be greater than 105% of vessel design pressure (1313 psig for a 1250 psig vessel).

The two SSV are set to actuate at 1240 psig. The proposed set point for the four dual function S/RV is at 1125 psig. Thus, requirements 2 and 3 are satisfied.

Requirement 1 is evaluated by considering the most severe isolation event with indirect scram. The S/RV are assumed to be active. The event which satisfies this specification is the closure of all main steam line isolation valves with flux scram. The results of this event are given in Table 2 and shown in Figure 2. An abrupt pressure and power rise occurs as soon as the reactor is isolated. Neutron flux reaches scram level in about 1.55 seconds, initiating reactor shutdown. The S/RV open to limit the pressure rise at the bottom of the vessel to 1335 psig. This response provides a 40 psi margin to the vessel code limit of 1375 psig. Thus, requirement 1 is satisfied and adequate overpressure protection is provided by the pressure relief system.

2.4 SMALL BREAK ACCIDENT

Analysis of the design basis LOCA (large break) demonstrates that the pressure decays during the event, and the increase in S/RV set point will have no effect on the results. However, for small breaks, the reactor will remain pressurized until the initiation of the automatic depressurization system (assuming the single failure of the HPCI). The increase in S/RV set point will result in a slight increase in inventory loss through the break during this period.

A new analysis of the previously limiting small break (0.05 ft^2) was performed to quantify this effect. The results showed a 7°F increase in PCT from 1875°F to 1882°F due to the increase of 30 psi in S/RV set point and the decrease of 10 psi in reactor dome pressure from the base case.

The 7°F increase in PCT is still well within the 2200°F PCT limit. Thus, there is no significant impact on PCT as the result of the increase in S/RV set point and the decrease in reactor dome pressure.

2.5 EFFECT OF HIGHER S/RV SET POINT ON CAPABILITY OF THE RCIC AND HPCI SYSTEMS

The design objective of the RCIC system is to provide sufficient water to cool the core during reactor isolation when the normal reactor heat sink (the main condenser) and feedwater makeup flow is unavailable (HPCI not available). After isolation the S/RV cycle to maintain vessel pressure within acceptable limits and the RCIC system is automatically put into operation by the low water level (level 2) signal. The present RCIC system characteristics were evaluated and found to be capable of rated design flow with the increase in the S/RV set point.

Even though no impact on RCIC performance was found, a loss of feedwater flow transient was performed. During this event, the reactor is scrammed on low water level 3 and isolated on low water level 2 with the initiation of the RCIC system, assuming HPCI is not available. The RCIC flow rate was reduced to 80% of the flow provided for rated conditions, to demonstrate the margin available in pumping capacity.

The results of the analysis showed that even under this postulated most limiting condition for RCIC operation, the minimum water level reached remains more than 4.5 feet about the top of the active fuel. Hence, there is no concern with respect to RCIC performance with the revised S/RV set point of 1125 psig.

A similar evaluation of the HPCI pump system characteristics also indicated that the existing HPCI system is capable of providing an adequate flow rate if the S/RV set point is increased to 1125 psig.

Table 1

PLANT OPERATING/NEW INPUT PARAMETERS

Parameter	Value
Dome Pressure (psig)	1024
Turbine Pressure (psig)	968
SRV Set Point (psig)	1125 + 1%
SRV Capacity at Set Point (% NBR)	41.07

Table 2

TRANSIENT OUTPUT DATA SUMMARY

Event	Power (%)	Core Flow (%)	Peak Neutron Flux (% of Ref)	Peak Surface Heat Flux (% of Ref)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
Load Rejection w/o Bypass - Trip Scram	100	100	599	119	1284	1297
MSIV Closure - Flux Scram	100	100	468	123	1321	1335*

*40 psi margin to ASME Code Limit of 1375 psig

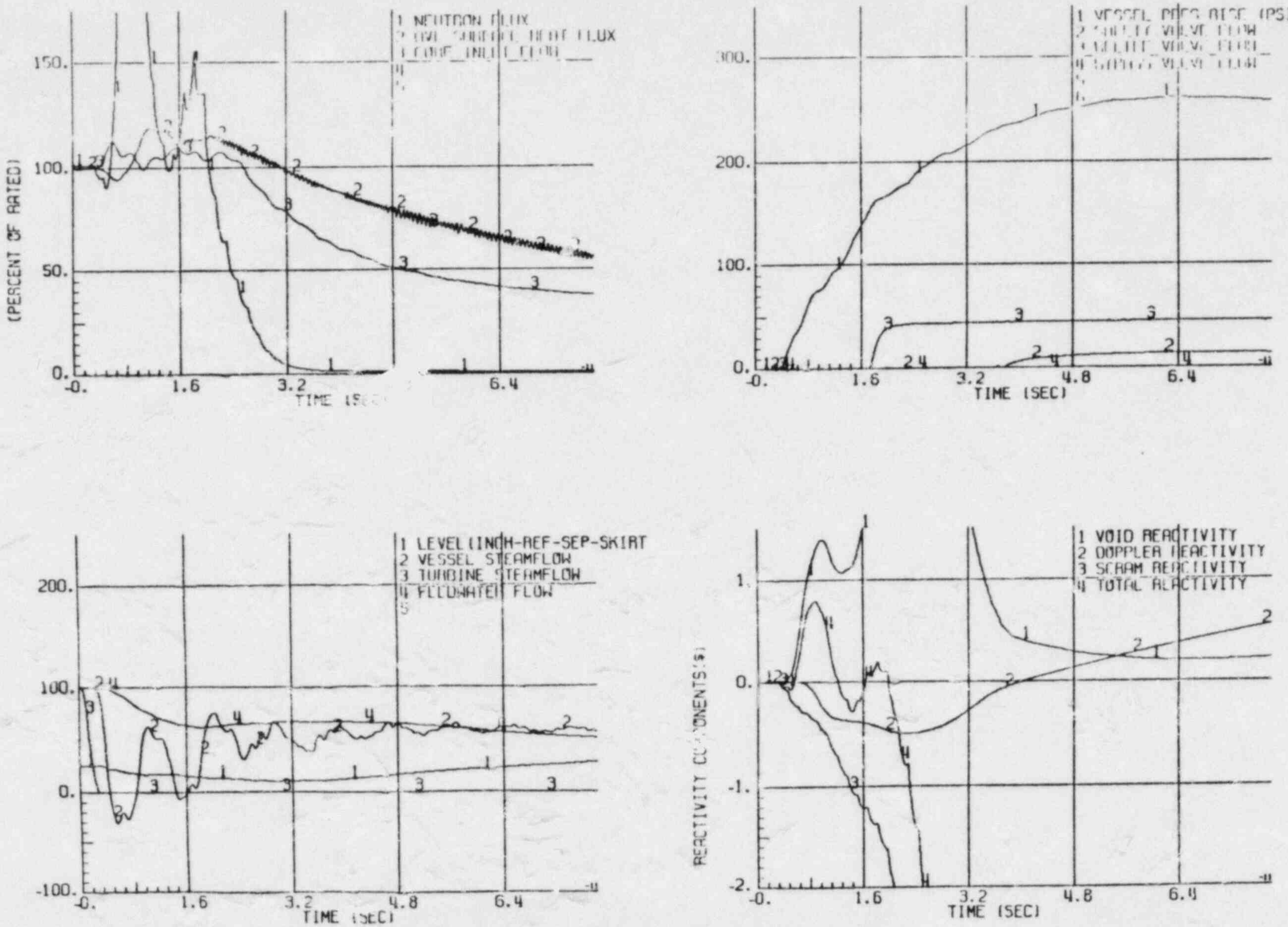

Figure 1. Generator Load Rejection Without Bypass, EOC 6, 100%/100% Power Flow Condition with Four SRV at 1125 psig + 1% Set Point and 2 SSV at 1240 psig + 1% Set Point

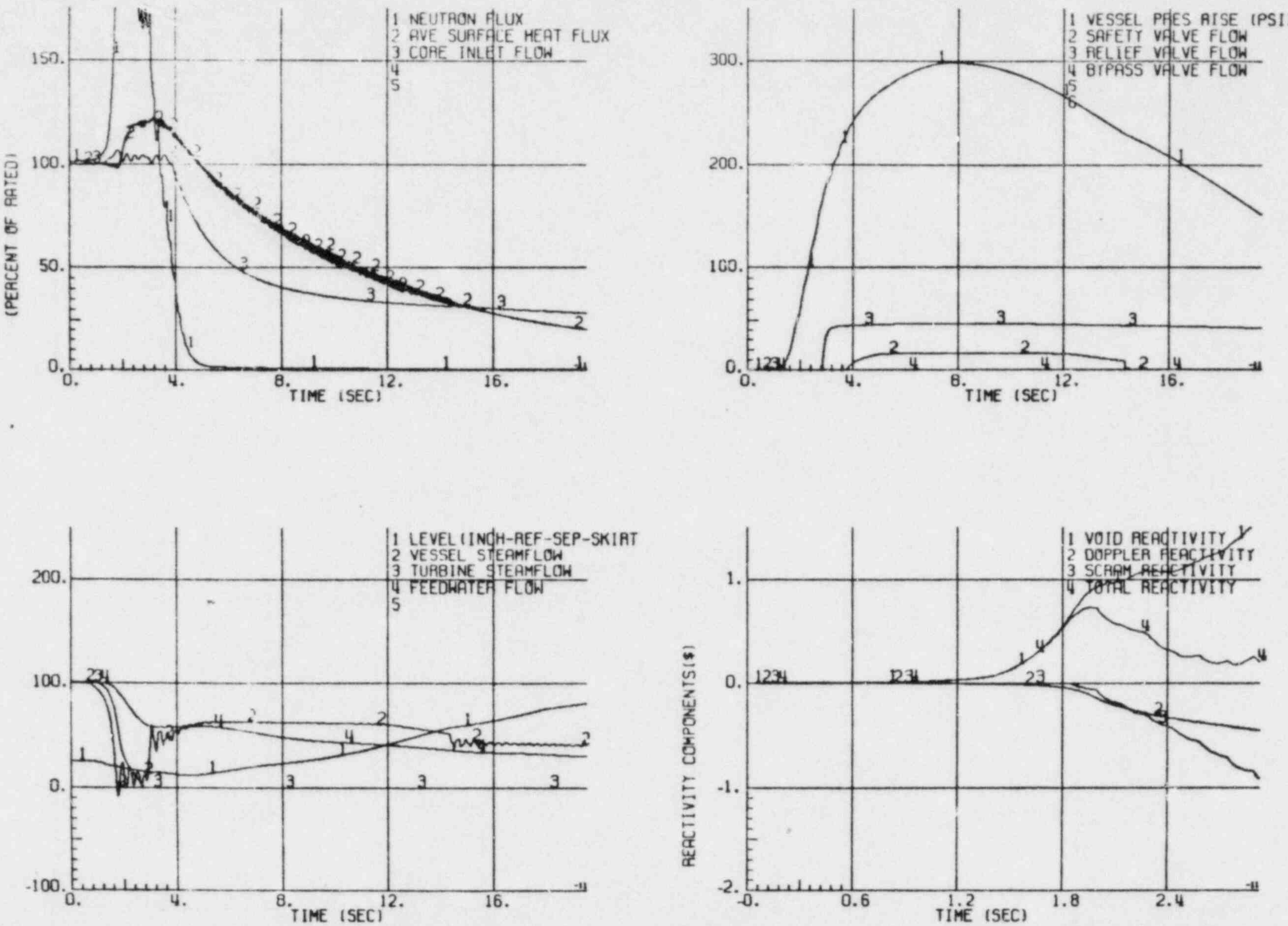

Figure 2. MSIV Flux Scram, EOC 6, 100%/100% Power Flow Condition with Four SRV at 1125 psig + 1% Set Point and Two Spring Safety Valves at 1240 psig + 1% Set Point. Peak vessel pressure is 1335 psig at 7.9 seconds into the event.

3. REFERENCES

1. "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1, Reload 5," General Electric Company, August 1981 (Y1003J01A28).

2. "Pilgrim Nuclear Power Station, Final Safety Analysis Report," Boston Edison Company.

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