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SAFETY RELIEF VALVE SIMMER MARGIN ANALYSIS
FOR
THE MONTICELLO NUCLEAR GENERATING PLANT

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1. INTRODUCTION AND SUMMARY

Steam leakage past the pilot disc of Target Rock safety relief valves (SRVs) may result in severe erosion at the disc/seat interface with extended operation, possibly impacting valve performance. Operating data demonstrates that an increase in the valve simmer margin (difference between normal plant operating pressure and the valve setpoint) will reduce the probability of pilot valve leakage. An investigation of the impact on plant transient and accident response is required when valve setpoints are increased.

A study has been performed for the Monticello Nuclear Generating Plant to optimize the simmer margin while maintaining required safety margins for vessel overpressure protection, fuel peak cladding temperature and loads associated with the actuation of SRVs. The impact of increased valve setpoints on High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems operation has also been considered in this evaluation.

The results of this study demonstrate that the SRV setpoints may safely be increased by 12 psi over the Monticello Cycle 11 SRV setpoints. The setpoint increase is limited by the HPCI and RCIC rated design pressure. The current setpoint pressures are summarized in Table 1-1, and the increased setpoint pressures are summarized in Table 1-2.

Table 1-1

CURRENT SRV SETPOINTS FOR THE MONTICELLO NUCLEAR GENERATING PLANT

SRV Setpoints (psig)	Valves[a]			
	1	2	3	4-8
Steam Pilot Opening	1108	1108	1108	1108
Steam Pilot Closing	1078	1078	1078	1078
Low-low Set Open	1040	1050	1060	--
Low-low Set Close	960	970	980	--

Table 1-2

INCREASED SRV SETPOINTS FOR THE MONTICELLO NUCLEAR GENERATING PLANT

SRV Setpoints (psig)	Valves[a]			
	1	2	3	4-8
Steam Pilot Opening	1120	1120	1120	1120
Steam Pilot Closing	1090	1190	1090	1090
Low-low Set Open	1052	1162	1072	--
Low-low Set Close	972	982	992	--

[a]Reload Licensing Analyses assume nominal SRV setpoint pressure plus 1%, as required by the ASME Boiler and Pressure Vessel code, section III.

2. SAFETY ANALYSIS

The safety analysis for the Monticello Nuclear Generating Plant, Cycle 11, is used as the basis for determining which transients are limiting. The increase in SRV setpoints 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 have been verified for Cycle 11 and will be confirmed for subsequent cycles by using the revised plant parameters in the reload licensing analyses.

The limiting events which were considered in establishing the new setpoints were as follows:

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

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

In addition, the capabilities of the RCIC and HPCI Systems were reevaluated at a dome pressure corresponding to the lowest set valve for the revised SRV setpoints. The effects of a 12-psi increase in SRV setpoints on calculated SRV quencher loads, SRV discharge line loads and torus shell loads resulting from SRV actuation were also considered.

The results of the analysis which demonstrate the acceptability of the increased simmer margin are given below. The analyses were performed using the same input parameters which were used for the Cycle 11 reload analysis with the exception of the SRV setpoints which were increased 12 psig.

2.1 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 Monticello Nuclear

Generating Plant, the pressure relief system includes eight dual function (Target Rock) SRVs. These valves provide the required ASME Code capacity to limit nuclear system overpressurization.

To protect the reactor pressure vessel from the consequences of pressure in excess of the vessel design pressures, the ASME Boiler and Pressure Vessel Code, Section III, specifies the following requirements:

a. Total rated relieving capacity shall be sufficient to prevent a rise in system pressure of more than 10% above design pressure under any system upset conditions. (1375 psig for a vessel with a design pressure of 1250 psig.)

b. The lowest qualified pressure relief device setpoint must be at or below vessel design pressure.

c. The highest safety valve setpoint must not be greater than 105% of vessel design pressure (1313 psig for a 1250-psig vessel).

The proposed setpoints for the eight dual function SRVs are 1120 psig. Thus, Requirements "b" and "c" are satisfied.

Requirement "a" is evaluated by considering the limiting isolation event with indirect scram. All SRVs are assumed to be active. The event which satisfies this specification is the closure of all MSIVs with flux scram. The results of this event are given in Table 2-1 and shown in Figure 2-1. An abrupt pressure and power rise occurs as soon as the reactor is isolated. Neutron flux reaches scram level in about 1.7 seconds, initiating reactor shutdown. The SRVs open to limit the pressure rise at the bottom of the vessel to 1244 psig. This response provides a 131-psi margin to the ASME code limit of 1375 psig. Thus, Requirement "a" is satisfied and adequate overpressure protection is provided by the pressure relief system with the revised valve setpoints.

2.2 PEAK CLADDING TEMPERATURE EVALUATION

Analysis of the design basis LOCA (large break) demonstrates that the pressure decays during the event, and the increase in SRV setpoints will have no effect on the results since the valves never open.

For small breaks the reactor will remain pressurized until the initiation of the automatic pressure relief system (ARS) assuming the single failure of the HPCI). The increase in SRV setpoints could potentially result in a slight increase in inventory loss through the break during this period leading to an increase in the PCT of the high power node.

The peak cladding temperature (PCT) for the limiting small break for Monticello is currently 1675°F.[1] This PCT is well below the limit of 2200°F. The 12 psi increase in the SRV setpoints will not have any effect on the calculated PCT. The inventory loss through the SRV which controls the PCT is dominated by the low-low set (LLS) operation. Changes in the LLS opening setpoint by 12 psi will not affect the PCT as long as the blowdown range is not changed. This is because the blowdown range is the dominant factor for the inventory loss for such a small change in LLS setpoint.

Therefore, the new SRV setpoints are acceptable from the standpoint of peak cladding temperature.

2.3 EFFECT OF HIGHER SRV SETPOINT 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 make-up flow are unavailable (HPCI not available). After isolation, the SRVs 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 SRV setpoints. The proposed SRV setpoints would maintain the reactor pressure below the design operating limit for the RCIC System.

A similar evaluation of the HPCI System characteristics also indicated that the 12-psi increase in SRV setpoints would maintain reactor pressure at or below the design operating limit of the HPCI System.

It is concluded that the increase in SRV setpoints will have no adverse effect on RCIC and HPCI System capability.

2.4 EFFECT OF HIGHER SRV SETPOINTS ON LOADS ASSOCIATED WITH SRV QUENCHER DISCHARGES

Four loads associated with the discharge of SRVs could potentially be affected by the 12-psi increase in SRV setpoints. These four loads are as follows:

a. Maximum water clearing thrust load

b. Torus shell pressure loading

c. Water jet induced loads on the quencher

d. Air bubble induced drag loads on the T-Quencher and submerged portions of the SRV discharge line (SRVDL)

The effects of the increased SRV setpoint pressures on the calculational procedures used to determine these four loads were evaluated by performing a sensitivity study.

The maximum water clearing thrust load and water jet induced loads on the quencher are calculated using the RVFOR computer model.[2] A conservative factor in RVFOR has been identified which more than compensates for the increase in loads associated with the increased SRV setpoints. Therefore, the current design margin for these loads is not reduced when the combined effect of the RVFOR conservatism and the increased SRV setpoints are considered.

The torus shell pressure load and air bubble induced drag loads are calculated using the RVFOR and QBUBS computer models.[3] The RVFOR margin previously discussed does not affect the final results of these load evaluations. A sensitivity study using these models demonstrated that an increase of less than 1% in the calculated torus shell pressure load and air bubble induced drag loads may be expected as a result of the increased SRV setpoints. This small increase in the calculated loads is insignificant compared to numerous conservatisms which have been identified in the load calculation procedures. Specifically, conservatively high assumed SRV flow rates and setpoints lead to a 15% conservative calculation of the loads. In addition, an empirical multiplier incorporated into the QBUBS model is considered to be 65% conservative. The 1% change in calculated torus shell loads and air bubble induced drag loads is insignificant compared to the conservatisms implicit in the models.

It is concluded as a result of this sensitivity study, that no significant increase in SRV quencher discharge related loads is expected to result from the increased SRV setpoint pressures.

2.5 CONTAINMENT RESPONSE

In terms of the containment responses (peak pressure and LOCA pool swell loads), the most severe event is the design basis LOCA. An increase in SRV setpoints has no effect on this event and, therefore, no impact on the above containment responses. A small break accident is the limiting event for peak drywell temperature. An increase in SRV setpoints will result in a slightly lower drywell temperature because saturated steam enthalpy decreases with increasing pressure.

2.6 CRITICAL POWER RATIO

The most severe reactor isolation which has an effect on the Minimum Critical Power Ratio (MCPR) is a feedwater controller failure event at 98% power and 100% flow (Table 3-2, Reference 4). Increasing the SRV setpoints will have no effect on the calculated CPR for this transient. With the current lower SRV setpoints the peak heat flux would occur before the SRVs open. Therefore, raising the setpoints will not change the peak heat flux.

2.7 IMPACT ON OTHER PROGRAMS

Raising the safety relief valve setpoints by 12 psi will have no adverse impact on the Extended Load Line Limit Analysis (ELLLA), ARTS, Appendix R, and low-low set programs. The setpoints are compatible with these programs.

The consequences of events initiated from within the ELLLA region are bounded by the consequences of the same events initiated from the licensing basis condition for Monticello, Cycle 11.[4] The increase in SRV setpoint will not change that condition.

The Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvements (ARTS) program is a comprehensive project with th objective of improving plant operating efficiency and performance.[5] The ARTS program required the implementation of numerous Technical Specification changes. An increase in SRV setpoints will not affect the results of the ARTS program because for the most severe transient the peak surface heat flux occurs before the SRV opens. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) thermal limits remain unchanged.

Appendix R of 10CFR50 concerns the fire protection program for nuclear power facilities, operating prior to January 1979. The proposed increase in SRV setpoints has no effect on the Appendix R requirements for Monticello.

Raising the SRV setpoints will have no impact on the low-low set function. The blowdown range, which remains unchanged, is the dominant factor which affects the time between SRV actuations. The low-low set function will still provide adequate time for clearing the water leg in the SRV discharge line.

Table 2-1

MSIV CLOSURE WITH FLUX SCRAM EVENT
WITH INCREASED SRV SETPOINTS

Event	Power (%)	Flow (%)	Peak Neutron Flux (% of Ref)	Peak Surface Heat Flux (% of Ref)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
MSIV Closure (Flux scram)	100	100	618	125	1228	1244[a]

[a]131 psi margin to ASME Code Limit of 1375 psig.

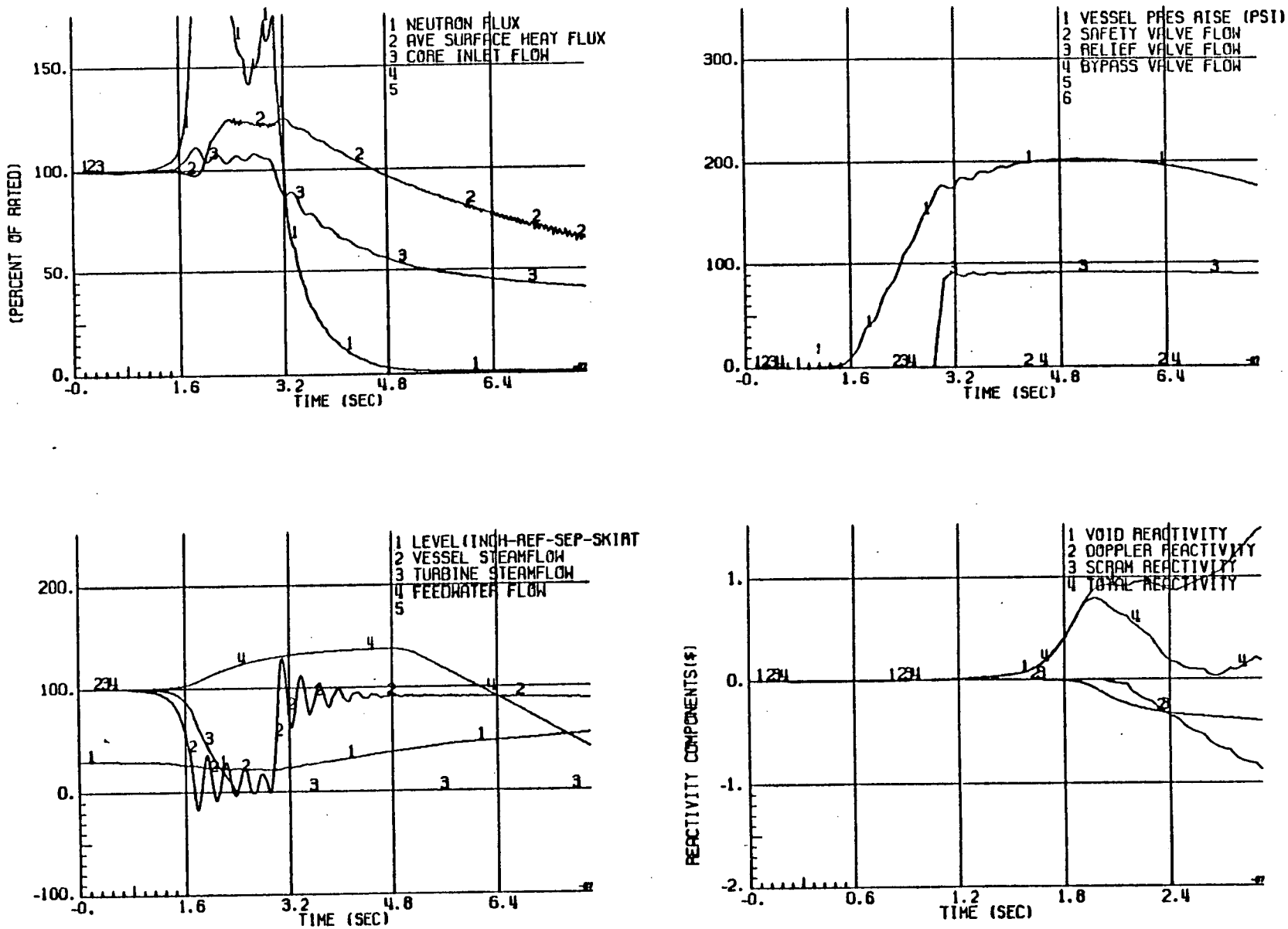

Figure 2-1. Monticello Cycle 11 MSIV Closure, Flux Scram

3. REFERENCES

1. "Evaluation of Mark I S/RV Load Cases C3.1, C3.2, and C3.3 for the Monticello Nuclear Generating Plant," General Electric Co., December 1981 (NEDC-24374).

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3. F. Valandani, "Mark I Containment Program-Analytical Model for Computing Air Bubble and Boundary Pressures Resulting from a SRV Discharge Through a T-Quencher Device," General Electric Co., March 1981 (NEDO-21878-1).

4. "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant, Cycle 11," General Electric Co., March 1984 (NEDC-30515).

5. "General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," General Electric Co., April 1984 (NEDC-30492-P).