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SUBJECT:
FORWARDING LIC NO DPR-22 APPL FOR AMEND: APPENDIX A TECH SPEC PROPOSED CHANGE CONCERNING REVISION TO THE PERMISSIBLE SETPOINT OF THE EIGHT SAFETY/RELIEF VALVES INSTALLED AT SUBJECT FACILITY TO 1108 PSIG... NOTARIZED 08/16/78... W/ATT NEDO-24133 [redacted]

PLANT NAME: MONTICELLO
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NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

August 16, 1978

Rec'd 08/18/78

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request Dated August 16, 1978

Attached are three originals and 37 conformed copies of a request for a change to the Provisional Operating License, Appendix A Technical Specifications, for the Monticello Nuclear Generating Plant. This change will raise the permissible setpoint of the eight safety/relief valves installed at Monticello to 1108 psig. The resulting improvement in valve simmer margin is expected to significantly improve the reliability of these valves.

Attached are forty copies of Exhibits A, B and C. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibit C is a safety analysis supporting the changes. We ask that this request be reviewed and approved prior to our Fall refueling outage which is presently scheduled to commence on October 1, 1978.

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/DMM/ak

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

Attachments

A001 3/40

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT — Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

(License Amendment Request Dated August 16, 1978)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A, Exhibit B, and Exhibit C. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibit C is a safety evaluation supporting the changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production & System Operation

On this 16th day of August, 1978, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

LICENSE AMENDMENT REQUEST
DATED August 16, 1978

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

Pursuant to 10CFR50, the holders of Provisional Operating License DPR-22 hereby propose the following changes to the Appendix A Technical Specifications.

PROPOSED CHANGES

1. Increase the allowable setpoint for the safety/relief valves from 1080 psig to 1108 psig.

2. Revise the minimum Operating MCPR Limit for both 8x8 and 8x8R fuel to 1.33.

3. Revise the Bases to be consistent with the changes proposed in (1) and (2) above.

REASON FOR CHANGES

For a number of years Northern States Power Company has been directly involved in improving the reliability of the eight safety/relief valves installed at the Monticello Nuclear Generating Plant. A number of equipment modifications and procedural changes have been made to greatly reduce the probability that a valve will fail to open when required. One continuing problem, however, is the tendency for these valves to leak excessively during normal operation. At other facilities, spurious opening or failure to reclose have been problems. While excessive leakage or spurious opening is not a serious safety problem, it does reduce availability by requiring the plant to go to cold shutdown for safety/relief valve replacement or repairs.

General Electric has determined that leakage and the probability of a spontaneous valve opening or failure to reclose is strongly influenced by the safety/relief valve simmer margin. Simmer margin is defined as the pressure difference between the valve setpoint and the normal system operating pressure. General Electric recommends increasing valve simmer margin to the maximum permitted by safety analyses. We propose in this License Amendment Request to increase the simmer margin by 28 psi and believe that this increase will significantly improve the performance of our valves.

We are able to increase safety/relief valve setpoint because of the large amount of excess relief capacity installed at Monticello. Monticello was originally designed with four safety valves discharging directly into the drywell atmosphere and four safety/relief valves with exhaust piping to the suppression pool. In 1974, all safety valves were replaced with safety/relief valves of the same type as those originally installed. This significantly increased the installed relief capacity in two ways (references 1 and 2). First, each safety/relief valve provided a greater flow rate than the spring safety valve it replaced. Second, because the setpoints of the safety/relief valves are substantially lower than safety valve setpoints, the modified system provided an earlier negative void reactivity feedback which aided in reducing transient pressure during limiting pressurization events. The NRC Staff gave credit for the valve capacity but required additional time to evaluate analytical models before allowing credit for the latter phenomenon. An interim Technical Specification was issued (page 5 of the safety evaluation attached to reference 2) which required seven safety/relief valves to be operable even though the transient analyses showed that only six valves were necessary. The practice over the four intervening years has been to license similar BWR's using the same analytical models for the number of operable safety/relief valves assumed in the transient analyses. The Monticello Technical Specifications were never revised to remove the interim conservatism imposed by the Staff. Rather than seek a reduction from seven to six operable safety/ relief valves at this time, we prefer to take credit for the seventh safety/relief valve which permits a 28 psi increase in the setpoints of all valves while maintaining an acceptable transient peak vessel pressure.

Reload 6 is scheduled for the 1978 Autumn refueling outage. In addition to the safety evaluation based on the currently authorized safety/relief valve setpoint of 1080 psig (reference 4), additional transient analyses have been performed to justify an increase in setpoint to 1108 psig. The results of this analysis are presented in Exhibit C.

As noted in Exhibit C, in the analysis of the turbine trip without bypass, it was found that the change in critical power ratio caused by the increased safety/relief valve setpoint is insignificant (0.002 delta CPR). This change affects the roundoff to two significant decimal places (the conventional roundoff adopted), therefore the MCPR Operating Limit is increased by 0.01 over the limit reported in reference (4).

SAFETY EVALUATION

The safety evaluation for Reload 6 was submitted for NRC review on August 10, 1978 (reference 4). Increases in the maximum allowable safety/relief valve setpoint only affect those events which result in valve self-actuation. The limiting events which have been reanalyzed are the most severe pressurization transient (turbine trip with failure of the bypass valve), vessel overpressure protection analysis (closure of all main steam isolation valves with indirect scram from high neutron flux), and the loss-of-coolant accident (small break). In addition, the capability of the RCIC and HPCI systems were evaluated for the higher safety/relief valve setpoints. Refer to Exhibit C for the results of these analyses. A safety/relief valve setpoint of 1108 psig for all eight valves is clearly acceptable.

A stress analysis of all four main steam lines and all eight safety/relief valve discharge lines was completed and submitted to the NRC for review when additional safety/relief valves were added (reference 1). Increasing the valve setpoint from 1080 to 1108 psig will result in a steam flow increase at setpoint pressure of less than three percent. Conservative assumptions were used in deriving the transient loads for the stress analysis reported in reference (1) making it valid for the increased valve setpoints. We will re-evaluate the torus discharge piping (including newly installed T-quenchers) for the 1108 psig setpoint using recent Monticello T-quencher test data.

All safety/relief valve discharge lines, main steam piping loads, and T-quenchers will be re-evaluated for the Mark I Containment Long Term Program loads and for the effects of the increased setpoints. This analysis will be initiated later this year when the discharge line loads model is available from General Electric. This reanalysis will allow for further increases in safety/relief valve simmer margin which may be justified in the future.

References

1. Letter from L O Mayer, NSP, to J F O'Leary, USAEC, dated January 23, 1974, "Permanent Plant Changes to Accommodate Equilibrium Core Scram Reactivity Characteristics," with Errata dated March 19, 1974.

2. Letter from K R Goller, USAEC, to L O Mayer, NSP, dated May 14, 1974, Amendment No. 3 to DPR-22.

3. Letter from L O Mayer, NSP, to Director, NRR, USNRC "License Amendment Request dated March 21, 1978."

4. Letter from L O Mayer, NSP, to Director, NRR, USNRC, dated August 10, 1978, "Supplement No. 1 to License Amendment Request March 21, 1978."

EXHIBIT B

LICENSE AMENDMENT REQUEST
DATED AUGUST 16, 1978

This exhibit consists of the following pages revised to incorporate all of the proposed Technical Specification changes:

23
25
26
119
134
189D*
189G*

*Revisions proposed in Supplement No. 1 to License Amendment Request dated March 21, 1978 are also shown on these pages. This supplement was submitted for NRC review on August 10, 1978.

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

2.4 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. The self-actuation function of at least seven Reactor Coolant System safety relief valves shall be operable. Valves shall be set as follows:

8 valves at $\leq$ 1108 psig.

Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is limited to 1207 psig. The safety/relief valves are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux) and the pressure at the bottom of the vessel is limited to 1248 psig in this case. The analysis assumed that only seven of the eight valves are operable and that they open at 1% over their setpoint with a 0.4 second delay. Reactor pressure is continuously monitored in the control room during operation on a 1500 psig full scale pressure recorder.

Bases:

2.4 The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1248 psig. Only seven of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1108 psig or lower. However, the actual set point can be as much as 11.1 psi above the 1108 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves 1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F. a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable. b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.	E. Safety/Relief Valves 1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal setpoint of all operational safety/relief valves shall be 1108 psig. b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage. c. The integrity of the safety/relief valve bellows shall be continuously monitored. d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such a leakage was comning from the primary 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. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

E. Safety/Relief Valves

Testing of all required safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1108 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the main steam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 Mwt. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 83.2% of the full power steam generation rate.

C. Minimum Critical Power Ratio (MCPR)

During power operation, the Operating MCPR Limit shall be $\geq$ 1.33 for 8x8 fuel and $\geq$ 1.33 for 8x8R fuel at rated power and flow. If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. For core flows other than rated the Operating MCPR Limit shall be the above applicable MCPR value times K_f where K_f is as shown in Figure 3.11.3.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR limit.

Bases Continued

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.18 for all fuel types. In addition, the ECCS analysis presented in Reference 6 assumed an initial MCPR of 1.24 for reduced flow conditions. The Operating MCPR Limit of 1.33 for 8x8 fuel and 1.33 for 8x8R fuel is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

3. Communication: VA Moore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Loss-of-coolant Accident Analysis Report for the Monticello Nuclear Generating Plant," NEDO-24050, September 1977, L O Mayer (NSP) to V Stello (USNRC), September 15, 1977.

5. "General Electric BWR Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's," R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.

EXHIBIT C

LICENSE AMENDMENT REQUEST

DATED MARCH 16, 1978

This exhibit consists of General Electric Report NEDO-24133-1 entitled, "Supplement 1 Monticello Reload 6 - Simmer Margin Evaluation." This report supplements the safety analysis for reload 6 contained in NEDO-24133. NEDO-24133 was submitted for NRC review on August 10, 1978.

NEDO-24133-1 provides the results of additional analyses which demonstrate the acceptability of increasing the maximum allowable safety/relief valve setpoint to 1108 psig.

NEDO-24133-1
Supplement 1
Class I
July 1978

SUPPLEMENT 1 FOR MONTICELLO RELOAD 6
SIMMER MARGIN EVALUATION

P. H. Henrikson
Licensing Engineer

Approved: R. O. Brugge

R.O. Brugge, Manager
Operating Licenses II

NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

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SUPPLEMENT 1 FOR MONTICELLO RELOAD 6
SIMMER MARGIN EVALUATION

1. INTRODUCTION AND SUMMARY

One event that has a significant impact on boiling water reactor (BWR) availability is the spurious opening or failure to reclose of the dual function safety/relief valves. As described in Reference 1, the event from a safety standpoint has a relatively minor effect on the reactor core and reactor coolant pressure boundary. The event does result in a significant maintenance outage since the reactor must be shutdown, depressurized, and the valve repaired or replaced before the plant can be restarted and continue with power operation.

The cause of the majority of these spurious openings or failures to reclose of safety/relief valves is excessive leakage around the setpoint pilot valve. Other causes of valve failures have been identified and corrective action has been taken. Operating data demonstrate that an increase in valve simmer margin (the differential pressure between the valve setpoint and normal system operating pressure at the valve) will reduce the probability of valve failure due to pilot leakage. A study was performed for Reload 6 of Monticello Nuclear Generating Plant, to determine if the simmer margin of the safety/relief valve could be increased without imposing additional restrictions on plant operation.

This supplement provides the results of these evaluations. As demonstrated in Section 2, the operating limits derived in NEDO-24133 are still valid for a 28 psi increase in safety/relief valve setpoint. Therefore, additional safety/relief valve reliability can be obtained for the next operating cycle without the imposition of any new limits on the plant.

2. SAFETY ANALYSIS

2.1 Introduction

The safety analysis for Reload 6 is provided in NEDO-24133. The raising of the safety/relief valve setpoints only affects those events which result in valve

operation to limit system pressure. The limiting events which require reanalysis are the most severe pressurization transient (turbine trip with failure of the bypass valve), vessel overpressure protection analysis (closure of all main steamline isolation valve - flux scram) and loss-of-coolant accident (small break). In addition, the capability of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems were re-evaluated for the higher safety/relief valve setpoints. 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 used in NEDO-24133 with the exception of safety/relief valve setpoint and capacity. The nominal safety/relief valve setpoint assumed was 1108 psig +1% using seven safety/relief valves. The capacity of the safety/relief valves at their setpoint was 83.2% of rated steam flow. The increase in safety/relief valve capacity is due to the increase in mass flow rate as a result of the higher pressure at the valve setpoint, and seven safety/relief valves are used instead of six.

2.2 Turbine trip With Failure of the Bypass Valves

This transient produces the most severe reactor isolation. The primary characteristic of this transient is a pressure increase due to the obstruction of steam flow by the turbine stop valves. 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 position switches on the turbine stop valves and by a void increase after the safety/relief valves have automatically opened on high pressure. The results of these analyses are given in Table 1 and shown in Figure 1.

The change in critical power ratio caused by the change in setpoint is insignificant (0.002 ΔCPR). However, this change was enough to affect the roundoff of the third significant figure, so that the ΔCPR for turbine trip without bypass with increased simmer margin is 0.26. Therefore, the MCPR Operating Limit with increased simmer margin is 1.33 for both 8x8 and 8x8R fuel.

2.3 Vessel Overpressure Protection Analysis

The pressure relief system must prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude an uncontrolled release of fission products.

The Monticello pressure relief system includes eight dual function safety/relief valves located on the main steamlines within the drywell between the reactor vessel and the first isolation valve. These valves provide the capacity to limit nuclear system overpressurization (analysis assumes 7 S/RV's).

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:

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

(b) The lowest qualified safety/relief valve setpoint must be at or below vessel design pressure.

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

Monticello's safety/relief valves will be set to self-actuate at a nominal setpoint of 1108 psig, thereby satisfying (b) and (c) above. Requirement (a) is evaluated by considering the most severe isolation event with indirect scram. The code does not require failure of reactor protective systems; however, General Electric provides a conservative analysis for licensing purposes which take credit only for reactor protective signals which are indirectly derived.

The event which satisfies this specification is the closure of all main steamline isolation valves with indirect (flux) scram. The initial conditions assumed are those specified in Section 6 of NEDO-24133. Figure 2 graphically illustrates the event. An abrupt pressure and power rise occurs as soon as the reactor is isolated. Neutron flux reaches scram level at about 1.7 seconds, initiating

reactor shutdown. The safety/relief valves open to limit the pressure rise to 1248 psig at the bottom of the vessel. This response provides a 127 psi margin to the vessel code limit of 1375 psig. Thus, requirement (a) is satisfied and adequate overpressure protection is provided by the pressure relief system.

2.4 Loss-of-Coolant Accident Analysis

Analysis of the design basis loss-of-coolant accident demonstrates that the pressure decays during the event, and the change in safety/relief valve setpoints 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 change in safety/relief valve setpoint will result in a slight increase in inventory loss of the break during this period.

ECCS analysis predicts a PCT of approximately 1760°F, which is 40°F higher than that for the case of the old SRV setpoint, for the most limiting small recirculation line break of 0.07 ft^2. This small increase in PCT is due primarily to the fact that the higher SRV setpoint results in higher vessel pressure which increases inventory loss and delays ECC systems initiation slightly.

2.5 HPCI and RCIC Capability

One of the design requirements for the HPCI and RCIC systems is that they be capable of providing design flow at the lowest safety/relief valve setpoint. These systems still meet the design requirement with the increase in lowest safety/relief valve setpoint to 1108 psig, the nominal setpoint.

Table 1

EVENT DATA SUMMARY (EOC7)

Event	Power (%)	Core Flow (%)	Peak Neutron Flux (% of Ref)	Peak Surface Heat Flux (% of Ref)	Peak Steamline Pressure (psig)	Peak Vessel Pressure (psig)
Turbine Trip w/o Bypass - Trip Scram	100	100	312	115	1168	1207
MSIV Closure, Flux Scram	100	100	602	127	1199	1248

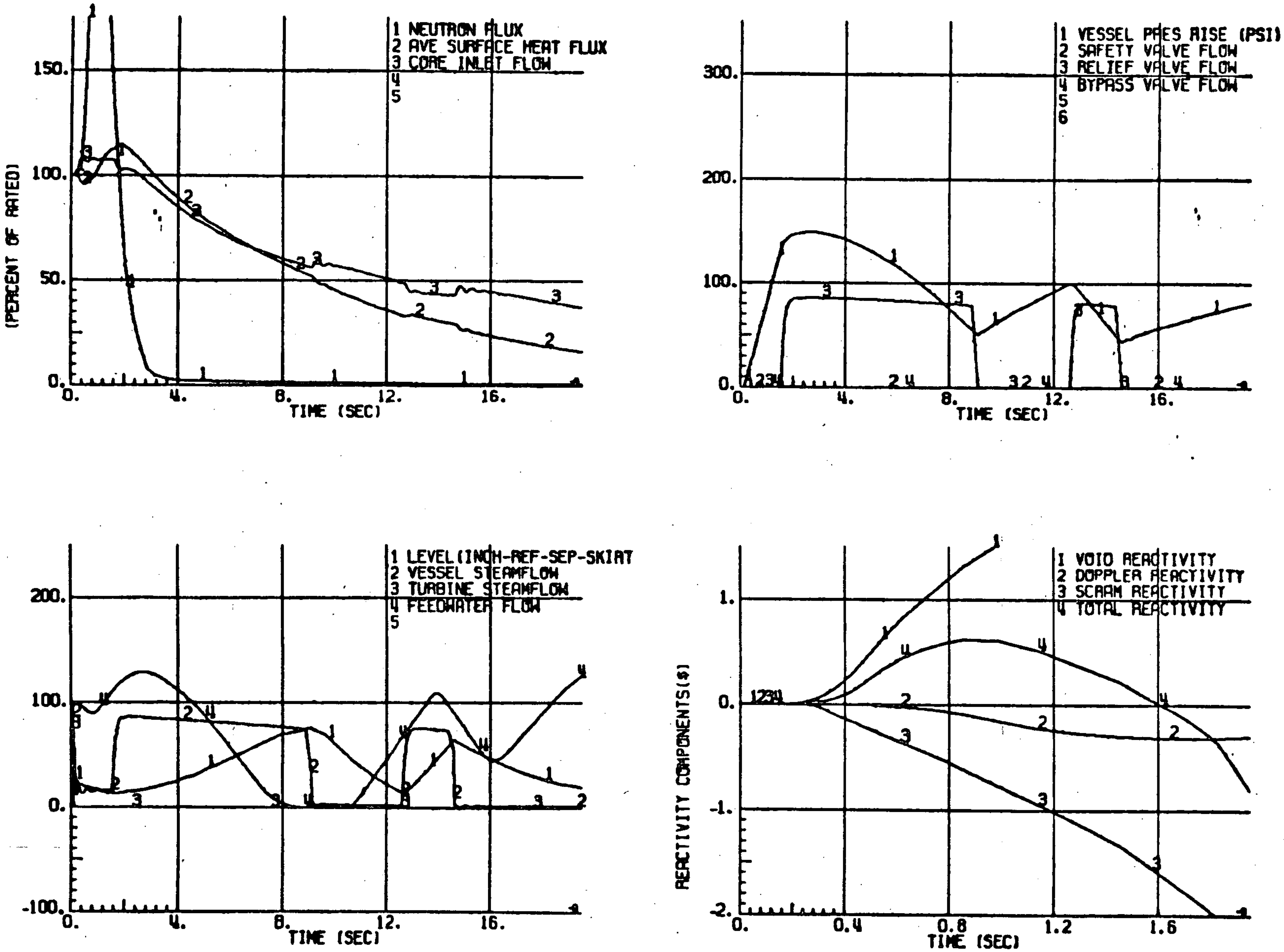

Figure 1. Monticello EOC7 Turbine Trip Without Bypass, Trip Scram, 100.0% Power

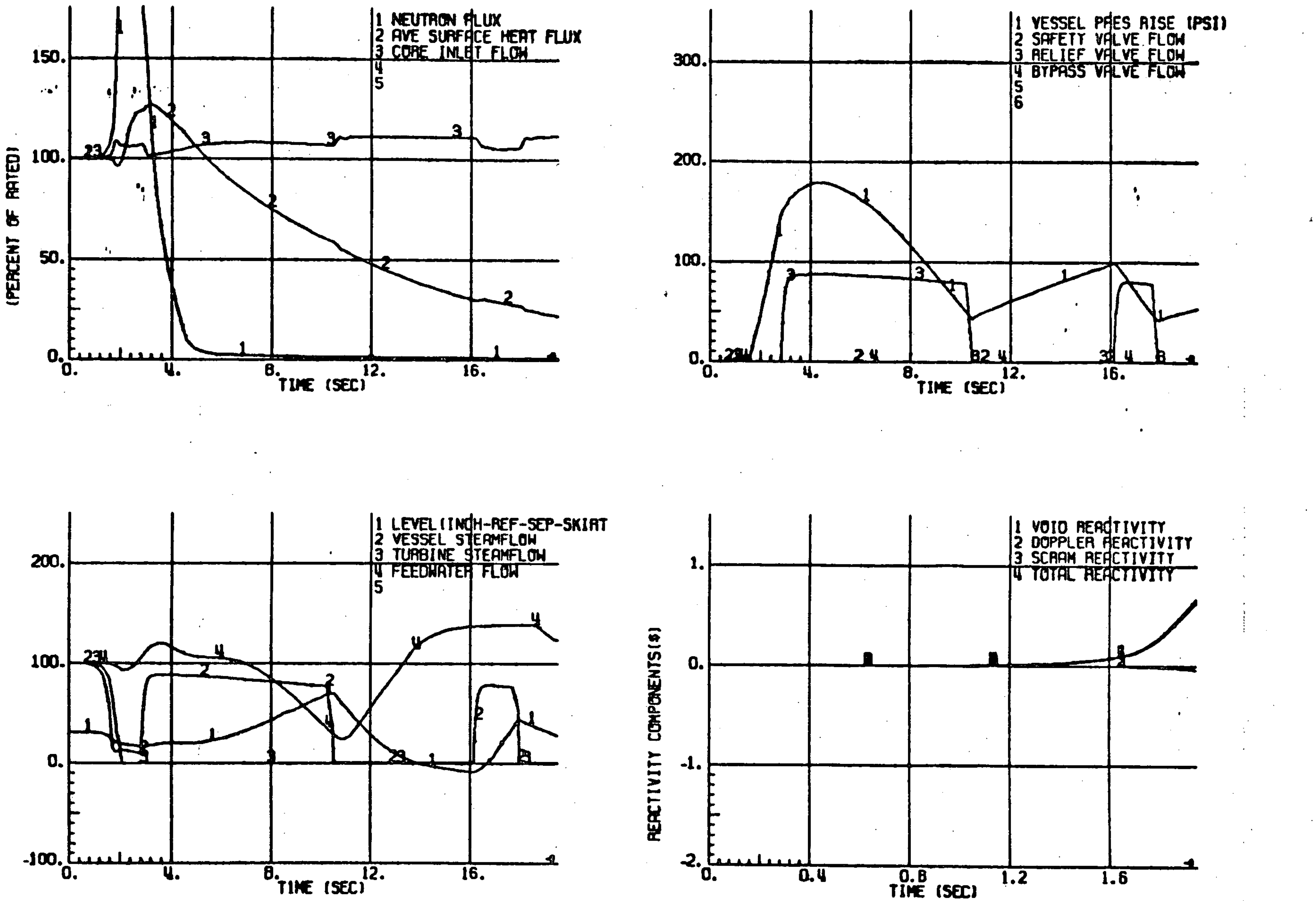

Figure 2. Monticello EOC7 MSIV Closure, Flux Scram