OCT 2 1973

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer, Director of
Nuclear Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Change No. 10
License No. DPR-22

Gentlemen:

We have reviewed your request (NSP letter dated September 13, 1973) to change the Monticello technical specification set points for the four spring-loaded safety valves on the steam lines at the reactor vessel from two at 1210 psig and two at 1220 psig to four set at 1240 psig and to require four safety valves where three of four installed were required in the past. According to your letter, the increase in safety valve set points to 1240 psig will increase the calculated fuel cycle 2 exposure threshold, considering the revised scram reactivity curve and the modified relief valve response times, by allowing higher system pressures following turbine trip without exceeding the 25 psi GE design margin between peak pressure and safety valve set point. A reanalysis to determine the new exposure threshold for Monticello fuel cycle 2 is in progress. However, until we have received and evaluated your analysis for the remainder of cycle 2, the fuel exposure threshold will be maintained at 1200 MWD/STU for cycle 2 and reactor power will continue to be limited by the fixed control rod inventory established when 1200 MWD/STU exposure level was attained. This is the fuel exposure threshold calculated prior to resetting safety valves upward to 1240 psig and modifying relief valves to improve response times.

We have concluded that the increase in safety valve set points from 1210-1220 to 1240 psig is within acceptable limits to prevent damaging pressure transients and, therefore, do not present significant hazards considerations. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed.

PO

OCT 2 1973

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Provisional Operating License No. DPR-22 are hereby changed by replacing existing pages 16, 20, 21, 23, 24, 25, 26, 85, 118, 119, and 134 with the enclosed revised pages bearing the same numbers.

Our Safety Evaluation is included for your information.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Revised pages as stated above
2. Safety Evaluation

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Bases Continued:

2.1 During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail (4,5,6,7). In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a delay in the control rod action to reduce neutron flux to less than the scram setting within 0.95 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed anytime a neutron flux scram setting of the APRM's is exceeded for longer than 0.95 seconds.

Analysis within the nominal uncertainty range of all appropriate significant parameters, show that if the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 0.95 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected.

The computer provided with Monticello has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

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(4) FSAR Volume I, Section III-2.2.3
(5) FSAR Volume III, Sections XIV-5
(6) Supplement on Transient Analyses submitted by NSP to the AEC February 13, 1973
(7) Letter from NSP to the AEC, "Planned Reactor Operation from 2,000 MWD/T to end of cycle 2", dated August 21, 1973

Bases Continued:

2.3 For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 18% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

The operator will set the APRM neutron flux trip setting no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 18.

B. APRM Control Rod Block Trips - Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at a given recirculation flow rate, and thus protects against exceeding a MCHFR of 1.0. This rod block set point, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable set point provides substantial margin from fuel damage, assuming steady state operation at the set point, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship, therefore,

Bases Continued:

2.3 the worst case MCHFR during steady state operation is at 110% of rated power. Peaking factors as specified in Section 3.2 of the FSAR were considered. The total peaking factor was 3.08. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than 3.08 exist. This assures a rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve (keeping the slope constant); thus, the entire curve will be shifted downward.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual set point can be as much as 6 inches lower due to the deviations discussed on Page 12.

D. Reactor Low Low Water Level ECCS Initiation Trip Point - The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram set point, and the ECCS initiation set point. To lower the set point for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

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. Reactor Coolant System Safety/Relief Valves Initiation shall be as follows:

4 valves at $\leq$ 1080 psig.

C. Reactor Coolant System Safety Valves Nominal Settings shall be as follows:

4 Valves at $\leq$ 1240 psig.

Bases:

2.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured in the vessel steam space is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value was derived from the design pressures of the reactor pressure vessel, coolant piping, and recirculation pump casing. The respective design pressures are 1250 psig at 575°F, 1148 psig at 562°F, and 1400 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and the USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10 percent over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure (120% x 1148 = 1378 psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 1.5 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 29,400 psi, still safely below the yield strength.

The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

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 1214 psig. The safety 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 1308 psig in this case. 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 valves offer yet another protective feature for the reactor coolant system pressure safety limit. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety 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 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 valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1308 psig. See FSAR Section 4.4.3 and supplemental information submitted February 13, 1973. Evaluations presented indicate that a total of eight valves (4 safety valves and 4 dual purpose safety/relief valves) set at the specified pressures maintain the peak pressure during the transient within the code of allowable and safety limit pressure.

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 1080 psig or lower. However, the actual set point can be as much as 11 psi above the 1080 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 consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10% of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated and reported in the Dresden II SAR Amendments 17 and 19. These evaluations, equally applicable to Monticello, show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. This requires the negative reactivity insertion in any local region of the core and in the over-all core to be equivalent to at least one dollar within 0.75 second. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity locally and in the overall core, respectively, within 0.75 second. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram as presented in the FSAR. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.8. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods.

(b) When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least once per shift and analyzed for conductivity and chloride ion content.

4. If Specification 3.6.C.1, 3.6.C.2, and 3.6.C.3 are not met, normal orderly shutdown shall be initiated.

D. Coolant Leakage

Any time irradiated fuel is in the reactor vessel, and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor placed in the cold shutdown condition within 24 hours.

D. Coolant Leakage

Reactor coolant system leakage into the drywell shall be checked and recorded at least once per day.

E. Safety and Relief Valves

1. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than 345°F, four safety valves and the safety valve func-

E. Safety and Relief Valves

1. A minimum of two safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. All four valves shall be checked or replaced

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
tion of four safety/relief valves shall be operable. The solenoid activated relief function of the safety/relief valves shall be operable as required by Specification 3.5.E.	every two refueling outages. The nominal popping point of the four safety valves shall be set at $\leq$ 1240 psig.
2. If specification 3.6.E.1 is not met, initiate an orderly shutdown and have coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.	2. a. A minimum of two safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. All four valves shall be checked or replaced every two refueling outages. The popping point of the safety/relief valves shall be set as follows:

Number of Valves	Set Point (psig)
4	$\leq$ 1080

b. At least one 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

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming 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.

The performance of the reactor coolant leakage detection system, including an evaluation of the speed and sensitivity of detection, will be evaluated during the first 18 months of plant operating, and the conclusions of this evaluation will be reported to the AEC. Modifications, if required, will be performed during the first refueling outage after AEC review. In addition, other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety and Relief Valves

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. A tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of the set pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the set pressure, the reactor coolant pressure safety limit of 1375 psig is not exceeded. Safety/relief valves are used to minimize activation of the safety valves. The operator will set the pressure settings at or below the settings listed. However, the actual setpoints can vary as listed in the basis of Specification 2.4.

The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a MSIV closure with the reactor at 1670 MWt. The analysis assumes no MSIV closure scram, but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assumed to total 83.9% (47% relief and 36.9% safety) of the full power steam generator rate. This capacity corresponds to assuming that four safety/relief valves (47%) and four safety valves (36.9%) operated.

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

Northern States Power Company (NSP), by letter dated September 13, 1973, has proposed to change the Technical Specifications of Provisional Operating License No. DPR-22 to permit operation of the Monticello Nuclear Power Plant with the four safety valve set points at 1240 psig instead of two at 1210 and two at 1220 psig and to require four safety valves where three of four installed valves were required previously. We have reviewed the proposed Technical Specifications changes and the safety analysis provided as attachments to the NSP letter.

According to the Final Safety Analysis Report (FSAR), three of four relief valves and two of four safety valves (ref. 1) provided sufficient capacity to guard against excessive pressure due to turbine trip without bypass, conservatively assuming reactor scram from a high flux signal instead of from the turbine valve trip signal. NSP in a later assessment (ref. 2) of relief and safety valve performance changed the basis for steam safety valve capacity determinations to simultaneous closure of all MSIVs assuming delayed reactor scram due to high neutron flux signal because this transient is more severe. For this transient, the peak steam pressure was calculated to be 1283 psig using the scram reactivity curve corresponding to an exposure threshold of 2250 MWD/STU (ref. 3). We accepted the revised basis for calculating safety valve requirements and changed the Technical Specifications (ref. 4) to show the revised pressure peak assuming three relief and two safety valves operated as designed following MSIV closure with delayed reactor scram due to high neutron flux.

Slower relief valve opening times (ref. 5) caused a reduction in the exposure threshold from 2250 to 2000 MWD/STU and prompted examination of the advantages that could be gained by setting safety valves at 1240 psig to allow an increase in transient peak pressure while maintaining the 25 psi GE design margin to the safety valve set point (ref. 6 and 7).

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According to the reanalysis of safety valve performance attached to the September 13, 1973 NSP letter, the overpressure peak for closure of all four MSIVs, assuming delayed reactor scram from the high flux signal and a new end-of-cycle (EOC) scram reactivity curve, the maximum vessel pressure (at the bottom of the pressure vessel) is 1308 psig or 67 psi below the maximum overpressure design limit of 1375 psig. However, the basis for the calculation was changed to require that all four safety/relief valves and four safety valves open. In the previous analysis, only three safety/relief and two safety valves were required. Therefore, the severity of the transients using the fuel exposure threshold at 2000 MWD/STU and EOC are not directly comparable. Inquiry brought the telephone response by NSP that the following combinations of safety/relief and safety valves had been evaluated at 100% power with 0.8 second relief valve response times and delayed flux scram after simultaneous closure (within 3 seconds) of all MSIVs:

1. 4 safety/relief valves and 0 safety valves
2. 3 safety/relief valves and 2 safety valves
3. 2 safety/relief valves and 4 safety valves

and the margin to 1375 psig design limit remains greater than 25 psi. The margin to the pressure design limit has, therefore, been reduced from 92 psi to approximately 25 psi under similar circumstances. We have concluded that this margin, with allowance for reliability considerations, is acceptable and the safety valves may, therefore, be set at 1240 psig instead of 1210 and 1220 psig. We note that both valve types, i.e., the pilot-operated safety/relief valve and the spring-loaded safety valve are pressure actuated (self-actuated) and are not dependent on any other source of power to prevent overpressure.

We understand that sensitivity calculations are currently being performed by NSP to determine the peak transient pressure effect of increasing the safety/relief valve set pressure to 1090 psig (from 1080) so that allowance can be made for set point drift or variations. Pending completion of this study and the analysis for the remainder of fuel cycle 2, however, Monticello operations should continue to be conservatively restricted by requiring the same core control rod inventory attained at 1200 MWD/STU specified by NSP prior to the September 29, 1973 shutdown. (Shutdown to modify relief valve response time and the increase in the safety valve set points.)

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On the basis of our evaluation, we have concluded that the increase in safety valve set point and the requirement for all four safety valves to be in service does not present an unreviewed safety consideration or significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor with the safety valve set points increased by 20 psi for two valves and 30 psi for the remaining two safety valves. The Technical Specifications should therefore be changed as proposed.

/s/

James J. Shea
Operating Reactors Branch #2
Directorate of Licensing

/s/ John I. Riesland for

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: OCT 2 1973

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REFERENCES

1. FSAR - page 4-4.4

"The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a turbine trip initiated with the reactor at 1670 MWt. The analysis assumes no steam bypass system flow, no turbine valve trip scram but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assured to total 50% (35% relief and 15% safety of the full power steam generator rate). This capacity corresponds to assuming that three of the four relief/safety valves (35.4%) and two of the four safety valves (18.5%) operated."

2. NSP letter to AEC dated February 13, 1973, transmitting "Results of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics". A significant change in the shape of the scram reactivity curve could occur by the end of fuel cycle 2 (see Figure 1 - the new analysis curve is sometimes referred to as curve B).

Page 4 - "It should be noted that the original FSAR analysis used for the safety valve sizing transient was the turbine trip without bypass (identical to instantaneous loss of condenser vacuum transient) with flux scram. However, it was determined with later plants that the main steam line isolation with flux scram could be more severe." Hence this analysis is used in checking safety valve adequacy.

Page 5 - Relief Valve Adequacy Transient

"A scram signal is initiated at the same time a turbine trip occurs by position switches on the turbine stop valves. This transient causes a rapid pressure increase in the reactor pressure vessel. Primary system relief valves are provided to remove sufficient energy from the reactor to prevent safety valves from lifting." Using improved control rod scram times (Figure 2) and four relief valves (three required previously) the peak pressure in the steam line at the safety valve location was calculated to be 1183 psig and since the lowest safety valve set point is 1210 psig, the GE design margin between peak pressure and the safety valve set point of 25 psi is maintained.

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Page 5 and 6 - Safety Valve Adequacy

Figure 4 shows the transient resulting from closure of all 4 MSIVs within 3 seconds wherein 3 of the 4 relief/safety valves open (32% of main steam generation rate) and only 2 of the 4 safety valves (18% of main steam generation rate). Neutron flux reaches the scram level at about 1.8 seconds, initiating reactor shutdown. The assumed safety valve capacity (Target Rock plus spring safety capacities) keeps the peak vessel pressure 92 psi below the peak allowable ASME overpressure of 1375 psig. "Therefore, the relief valves plus spring safety valves provide adequate protection against excessive overpressurization of the nuclear system process barrier with a large margin because of the reduced capacities assumed for this analysis."

3. NSP letter to AEC dated June 1, 1973 - Request to change the Technical Specifications to require four operable relief valves instead of three, and slightly shorter control rod scram times in accordance with the analysis presented in the attachment to NSP letter dated February 13, 1973 (reference 2 above). "Preliminary calculations show that the new analyses present the most limiting conditions expected during the first 2250 MWD/STU exposure increment of cycle two."

4. AEC approval letter (Change No. 8) dated July 2, 1973, to require four relief valves instead of three as previously required and slightly faster scram times than previously specified in accordance with NSP change request dated June 1, 1973 (reference 3 above) for reactor operation at rated power out to 2250 MWD/STU.

   "We are continuing our evaluation of the shape changes in the scram reactivity curve and the necessity for more restrictive technical specifications but agree that the technical specification changes you have proposed should be made now."

5. NSP letter to AEC dated August 1, 1973 - "Observed Relief Valve Opening Times Different than those Assumed in the Transient Analysis".

   General Electric reports that results of Target Rock relief valve performance tests show a delay in initial opening time of about 0.8 second rather than 0.2 second as reported in the Monticello FSAR.

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6. NSP letter to AEC dated August 21, 1973 - "Planned Reactor Operation from 2000 MWD/T to the End of Cycle 2".

   Page 2 - "Relief valve modifications will reduce peak vessel pressure following transients for the end of cycle 2 as well as subsequent cycles. Safety valve setting increases will maintain or improve the margin between vessel pressure and valve set points" (following turbine trip without steam bypass).

7. AEC Memo to File dated September 13, 1973.

   We will consider a change to the Technical Specifications to increase safety valve set point from 1210-1220 psig to 1240 psig. Our final conclusions, in this regard, are dependent on additional analysis for the period beyond 2000 MWD/T to be provided by NSP.

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