Docket File

JUL 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. 8
License No. DPR-22

Gentlemen:

We have reviewed your request (NSP letter dated June 1, 1973) to change the Monticello Technical Specification requirements for relief valves and control rod scram times. Based on the report "Result of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle 1 Core Dynamic Characteristics" provided to the Directorate of Licensing by NSP letter dated February 13, 1973, and the statement in your June 1, 1973 letter that preliminary calculations show that the End-of-Cycle 1 analysis presents the most limiting conditions expected during the first 2250 MWD/STU exposure increment of cycle 2, you have proposed that the required number of relief valves be increased from 3 to 4 and that the control rod scram time be changed to require faster response and insertion times over the first 2 seconds of the 5-second scram time interval. With primary system pressure relieved through 4 relief valves instead of 3 and the faster control rod scram initiation you have requested, the maximum core coolant pressure according to the EOC 1 analysis following an assumed turbine trip and failure of the automatic turbine bypass valve to open will remain at least 25 psi below the lowest of the 4 safety valve set points, thereby satisfying a General Electric requirement that the pressure margin to the safety valve set point be 25 psi.

Rg

We further understand from telephone conversations with NSP representatives that analysis will continue for the period beyond 2250 MWD/STU to the end of cycle 2 and cycle 3 or until the equilibrium fuel cycle conditions are attained and that additional Technical Specification changes may be necessary for the period beyond 2250 MWD/STU to maintain relief and safety valve core coolant system overpressure safety margins. To assure that we have sufficient time to complete our review of your analysis for operation beyond an incremental fuel exposure of 2250 MWD/STU during cycle 2, it is requested that your analysis and any proposed changes in technical specifications be submitted at least 30 days prior to the date at which an exposure increment during cycle 2 of 2250 MWD/STU would be achieved.

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 Specifications changes you have proposed should be made now. All four relief valves must be in service where in the past is was possible to operate the reactor with one of the four relief valves out of service. Analysis of the overpressure transient following turbine trip without bypass will be based on faster Technical Specifications control rod scram times than previously required but conservatively slower than actual measured control rod scram times.

We have concluded that the changes you have proposed by your June 1, 1973 letter as modified by your letter of June 20, 1973, are required to maintain acceptable core coolant system overpressure safety margins and do not present significant hazards considerations, and 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 to an exposure increment in the second fuel cycle of 2250 MWD/STU.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License No. DPR-22 are hereby changed to replace existing pages 22, 24, 25, 26, 39, 79, 85, 85a, 86, 112, 119, and 134 with revised pages bearing the same numbers. The revised pages also include minor typographical corrections.

Sincerely,

Original Signed by:
Dennis L. Ziemann

for

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

Enclosure and cc: See next page

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JUL 2 1973

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Revised pages as stated al e

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DATE ▶	6/28/73	6/28/73	6/29/73	7/2/73		

Bases Continued:

2.3 The operator will set the low low water level ECCS initiation trip setting $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 18.

E. Turbine Control Valve Fast Closure Scram - The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Reference Sections 14.5.1.1 and 14.5.1.2 FSAR.

F. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure, only a slight increase in surface heat flux occurs and thus adequate margin exists. The primary system relief valves open to limit the pressure rise, then reclose as pressure decreases. For this condition the peak surface heat flux is less than 110% of its rated power value and MCHFR remains above 1.8. Referenc FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973.

G. Main Steam Line Isolation Valve Closure Scram - The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 850 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 18.

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.

The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip from rated power 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 1183 psig. The safety valves are sized assuming no direct scram during

Bases:

2.2 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 1283 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 1283 psig. See FSAR Section 4.4.3 and supplemental information submitted February 13, 1973. Evaluations presented indicate that a total of five valves (2 safety valves and 3 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 22. 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 22.

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

Bases Continued:

3.1 condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds ten times normal full power background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive release of radioactive materials. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation to the main condenser off-gas line provided the instantaneous limit specified in Specification 3.8 is exceeded for a 15-minute period.

The main steamline isolation valve closure scram is set to scram when the isolation valves are $\leq$10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Reference Section 14.5 .1.3.1 FSAR and supplemental information submitted February 13, 1973.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.7.1 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	5.00

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	5.300

C. Scram Insertion Times

During each operation cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

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 power transient is that resulting from closure of the MSIV's with failure of the valve closure scram but an indirect scram from high flux. 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.

Bases Continued 3.3 and 4.3:

This is adequate and conservative when compared with the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of the time interval results from the sensor and circuit delays; at this point the pilot scram solenoid deenergized. Approximately 120 milliseconds later control rod motion is estimated to begin. However, to be conservative, control rod motion is not assumed to start until 200 milliseconds later. This value was included in the transient analyses and is included in the allowable scram insertion times of Specifications 3.3.C.1 and 3.3.C.2.

The scram times for all control rods will be determined at the time of each refueling outage. The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram since if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. The frequency of exercising the control rods under the conditions of two or more control rods out of service provides even further assurance of the reliability of the remaining control rods.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds six,the allowable number of inoperable rods.

D. Control Rod Accumulators

The basis for this specification was not described in the FSAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$ -- other repeating rod sequences with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Rod inventory predictions can be normalized to actual initial steady state rod patterns to minimize calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one per cent in reactivity.

margin, the RCIC system (a non-safeguard system) has been required to be operable during this time, since the RCIC system is capable of supplying significant water makeup to the reactor (400 gpm).

E. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray system or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. Either of the two core spray systems or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact. Three of the four relief/safety valves are included in the automatic pressure relief system. Of these three, only two are required to provide sufficient capacity for the automatic pressure relief system. See Section 4.4 and 6.2.5.3 FSAR.

F. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

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.

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.

every two refueling outages. The nominal popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig)
2	$\leq$ 1210
2	$\leq$ 1220

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

Bases Continued 3.6 and 4.6:

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 measureable. 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 operation, 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 ±1% of design 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 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%) and two of the four safety valves (18%) operated. For additional margin three safety and four safety/relief valves are required to be operable.