NOV 27 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. 11
License No. DPR-22

Gentlemen:

Your letters of January 18, March 2, April 11, and October 4, 1973, submitted additional information concerning your request dated September 22, 1972, for proposed changes to the Technical Specifications of the Monticello Nuclear Generating Plant to revise the maximum reactivity that could be added by the dropout of any one control blade. Your September 22, 1972 letter also proposed additional surveillance requirements for the rod worth minimizer.

During our review of your proposed changes, we informed your staff that certain modifications were necessary in the bases for the maximum reactivity added specification and in the limiting condition for operation of the rod worth minimizer. The modification to the bases has been made. The modification to the limiting condition of operation for rod worth minimizer operability has been made with an effective date of May 1, 1974, to allow time for implementation of measures necessary to achieve acceptable rod worth minimizer reliability and operability.

We have concluded that the proposed changes to the Technical Specifications, as modified, do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered.

A copy of our Safety Evaluation of this change is enclosed.

Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License No. DPR-22 are hereby changed by replacing pages 77, 78, 82, and 84 with the revised pages appended hereto.

Sincerely,

Original Signed by
D. J. Skovholt

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

Enclosures:
1. Safety Evaluation
2. Revised pages

cc w/enclosures:
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See next page for additional cc

 NOV 27 1973

cc w/enclosures and cy of NSP
ltrs dtd 9/22/72, 1/18/73,
3/2/73, 4/11/73, and 10/4/73:
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CDeBevec, L:ORB #2

Discussed the AEC changes to NSP proposal as reflected in the Tech Spec with NSP representatives on 11/26/73. They concurred reluctantly in the RWM operability requirement. JJShea

DR 11-7-73

OFFICE ▶	L:ORB #2	L:ORB #2	L:ORB #2	L:RS	L:OR	
SURNAME ▶	X7403 JJShea:sjh	RWReid RMDiggs	DLZiemann	VStello	DJSkovholt	
DATE ▶	11/1/73	11/1/73	11/6/73	11/7/73	11/?/73	

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

ROD DROP ACCIDENT

In a letter dated September 22, 1972, Northern States Power Company submitted a request for changes to the Technical Specifications for the Monticello reactor concerning the rod drop accident. In response to our requests, Northern States Power Company submitted additional information in letters dated January 18, March 2, April 11, and October 4, 1973. In addition, a meeting was held on May 17, 1973, with representatives of Northern States Power Company and the General Electric Company to review the calculational models and to discuss the input assumptions to be used in the change to the rod drop accident technical specifications. The change is based on new calculational models developed by the General Electric Company, presented in references 1, 2, and 3, and by a change in the assessment of the accident and scram reactivity shape. These changes result in a reduction in maximum allowable in-sequence control rod reactivity worth from 2.5% to 1.3% delta k/k, and increase the assurance that a control rod is not in an out-of-sequence position during low power operation.

The rod drop accident is one of the design basis accidents for boiling water reactors. For calculational purposes it is assumed that a control rod blade separates from its drive, lodges in the core with the drive withdrawn, and drops at the time which causes the most serious power excursion due to rapid reactivity insertion. The consequences of this accident are evaluated by determining the energy input to the fuel assuming

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(1) Paone, C. J., Stirn, R. C., and Wooley, J. A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.
(2) Stirn, R. C., Paone, C. J., and Young, R. M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972.
(3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores", Supplement 2 - NEDO-10527, January 1973.

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that the reactivity worth of the dropped rod is the maximum which could occur. The maximum acceptable energy in the fuel is limited such that, in the event of fuel cladding failure, the energy input into the coolant will not result in a pressure pulse which might damage the core geometry or the reactor pressure vessel.

The analytical methods used by the General Electric Company (GE) to evaluate the consequences of the rod drop accident have been reviewed by the staff and independent calculations have been performed by Brookhaven National Laboratory which show reasonable agreement with GE results. Based on these reviews, it is concluded that the analytical methods used by GE are acceptable.

Application of the GE analytical methods to operating reactors requires that the input parameters conservatively represent the reactor core over a broad range of operating conditions. The proposed changes to the Technical Specifications include, in the Bases, a set of boundary conditions which are used to calculate the maximum allowable reactivity worth of a control rod. It is not expected that these boundary conditions will be exceeded for reactor cores of current design. The boundary conditions include a maximum inter-assembly local power peaking factor, an end-of-cycle delayed neutron fraction, a beginning of life Doppler reactivity feedback, the technical specification control rod scram insertion rate, a control rod drop velocity of 3.11 ft/sec, and specified accident and scram reactivity shape functions. The rod drop velocity of 3.11 ft/sec is based on tests with a "worst case" rod built with maximum clearances and features known to contribute the high rod drop velocities. The difference between the mean rod drop velocity and the 99.9% confidence limit for a group of production rods was added to the mean velocity obtained for the "worst case" control rod. We have included in the Bases the value 0.005 end-of-cycle delayed neutron fraction to further define the boundary assumptions that were used in the calculations. In addition, we have added a statement to the Bases that each reload core must be analyzed to show conformance to the bounding assumptions. The peak fuel enthalpy resulting from an in-sequence rod drop accident within the above boundary conditions is calculated not to exceed 280 cal/gm, which is acceptably below the peak fuel enthalpy at which prompt fuel dispersal would occur based on the SPERT tests. Based on the above, the resultant maximum allowable in-sequence rod worth of 1.3% delta k/k is acceptable.

Separate consideration is being given to the potentially adverse effect of compaction of boron carbide in the control rods on the rod drop accident in the event of inverted poison tubes. The evaluation of the effect of possible inverted poison tubes on the allowable in-sequence rod worth is currently in progress and if determined necessary, appropriate changes to the allowable control rod reactivity worth will be made.

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If a control rod is withdrawn out of sequence, a rod worth of greater than 1.3% delta k/k could result. In the event of rod drop accident associated with such an out-of-sequence rod, the peak fuel enthalpy could exceed 280 cal/gm. The rod worth minimizer (RWM) is designed as an operator aid to prevent an out-of-sequence rod withdrawal. Current Technical Specifications allow the RWM to be bypassed if it is inoperable during a reactor startup provided that a second operator is assigned to monitor the rod withdrawal sequence. To increase the control on RWM availability during reactor startups, the technical specification is being changed to require that the RWM be operable for the withdrawal of a significant number of control rods. The effective date of the change in technical specifications concerning RWM operability is being deferred for six months to allow any necessary upgrading of the RWM to be accomplished.

Based on the above, we conclude that the proposed changes do not involve significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered.

/s/

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

/s/

Robert W. Reid
Operating Reactors Branch #2
Directorate of Licensing

/s/

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

Date: NOV 27 1973

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3.0 LIMITING CONDITIONS FOR OPERATION

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met.

3. (a) Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical.

4.0 SURVEILLANCE REQUIREMENTS

(b) when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernbile, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response.

2. The control rod drive housing support syste shall be inspected after reassembly and the results of the inspection recorded.

3. (a) To consider the rod worth minimizer operable, the following steps must be performed:

   (i) The control rod withdrawal sequ for the rod worth minimizer computer shall be verified as correct.

   (ii) The rod worth minimizer compute on-line diagnostic test shall b successfully completed.

   (iii) Proper annunciation of the sele error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

3.0 LIMITING CONDITIONS FOR OPERATION

(b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. After May 1, 1974, the second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. Whenever the Engineer, Nuclear, determines that a limiting control rod pattern exists, withdrawal of designated control rods shall be permitted only when the RWM system is operable.

4.0 SURVEILLANCE REQUIREMENTS

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

(b) If the rod worth minimizer is inoperal while the reactor is in the startup or run mode below 10% rated thermal powe and the second independent operato or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawa of each rod group.

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.

5. Whenever the Engineer, Nuclear, determines that a limiting control rod pattern exists, an instrument functional test of the RWM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

Bases Continued 3.3 and 4.3:

A. Reactivity Limitations

1. Reactivity margin - core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk in the most reactive condition during the operating cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the FSAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. For the first fuel cycle, R was calculated to be 0.012 Δk. A new value of R must be determined for each fuel cycle.

The 0.25% Δk in the expression R + 0.25% Δk is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% Δk in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the strongest rod fully withdrawn but at least a R + 0.25% Δk margin beyond this.

2. Reactivity margin - stuck control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved

Exhibit B (Continued)

Bases Continued 3.3 and 4.3

Section 6.5.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted since the reactor would remain sub-critical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta k supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.(3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta k limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Recent improvements in analytical capability have allowed more refined analysis of the control rod drop accident. These techniques have been described in a topical report and two supplements.(1)(2)(3) By using the analytical models described in these reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

(1) Paone, C J, Stirm R C and Wooley, J A, "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.
(2) Stirn, R C, Paone, C J, and Young, R M, "Rod Drop Accident Analysis for Large BWR's," Supplement 1 - NEDO-10527, July 1972.
(3) Stirn, R C, Paone, C J, and Haun, J M, "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

Exhibit B (Continued)

Bases Continued 3.3 and 4.3

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in reference 4. Each core reload will be analyzed to show conformance to the limiting parameters.

a. A startup inter-assembly local power peaking factor of 1.30 or less. (5)

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity as a result of a rod drop and in a conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

(4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3.(a)" transmitted by letter from L. O. Mayer (NSP) to J. F. O'Leary (USAEC) dated October 4, 1973.

(5) To include the power spike effect caused by gaps between fuel pellets.

Exhibit B (Continued)

Bases Continued 3.3 and 4.3

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses twice that previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Reference Section 7-9 FSAR. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup after May 1, 1974.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The