April 14, 1997

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - REQUEST FOR ADDITIONAL INFORMATION ON LICENSE AMENDMENT REQUEST DATED JULY 26, 1996 ENTITLED "SUPPORTING THE MONTICELLO NUCLEAR GENERATING PLANT POWER RERATE PROGRAM" (TAC NO. M96238)

Dear Mr. Anderson:

By letter dated July 26, 1996, Northern States Power Company (NSP) submitted a license amendment request entitled "Supporting the Monticello Nuclear Generating Plant (MNGP) Power Rerate Program." The requested amendment proposes an increase in the MNGP Operating License maximum power level to 1775 megawatts thermal, as well as supporting changes to the MNGP Technical Specifications. This change reflects an increase of 6.3% above the currently licensed power level of 1670 megawatts thermal. In the letter, NSP requested completion of the staff review by December 1, 1997.

Subsequently, in a letter dated January 20, 1997, NSP requested that the staff's review schedule be deferred for about 8 months such that the staff review is completed by July 31, 1998. NSP's letter indicated that this delay in schedule will allow NSP to focus its resources on additional issues recently identified during an NRC inspection.

Based on a preliminary review of the subject submittal dated July 26, 1996, the staff has determined that additional information is necessary to complete its review. The enclosed request for additional information (RAI) provides details of the required material. Please advise NRC of NSP's schedule for responding to the enclosed RAI.

Sincerely,

Original signed by



Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No: 50-263
Enclosure: As stated
cc w/encl: See next page
DISTRIBUTION:

| | |
|---|---|
| Docket File | OGC |
| PUBLIC | ACRS |
| PD31 RF | JJacobson, DRP/RIII |
| JRoe | EAdensam (EGA1) |

DFO1/1

DOCUMENT NAME: G:\WPDOCS\MONTICEL\MON96238.RAI
To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

| OFFICE | PD31:PM | E | PD31:LA | E | PD31:PD | |
|---|---|---|---|---|---|---|
| NAME | TJKim:db | | CJamerson | | JHannon | |
| DATE | 4/11/97 | | 4/11/97 | | 4/14/97 | |

OFFICIAL RECORD COPY

cc:

J. E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
ATTN: Site Licensing
Northern States Power Company
2807 West County Road 75
Monticello, Minnesota 55362-9637

Robert Nelson, President
Minnesota Environmental Control
Citizens Association (MECCA)
1051 South McKnight Road
St. Paul, Minnesota 55119

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, Minnesota 55119

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

Darla Groshens, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, Minnesota 55313

Kris Sanda, Commissioner
Department of Public Service
121 Seventh Place East
Suite 200
St. Paul, Minnesota 55101-2145

Adonis A. Neblett
Assistant Attorney General
Office of the Attorney General
445 Minnesota Street
Suite 900
St. Paul, Minnesota 55101-2127

REQUEST FOR ADDITIONAL INFORMATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO LICENSE AMENDMENT REQUEST DATED JULY 26, 1996
SUPPORTING THE MONTICELLO NUCLEAR GENERATING PLANT
POWER RERATE PROGRAM
DOCKET NO. 50-263

Electrical Systems:

1. Information provided in Exhibit A (page A-24) indicates no change for normal conditions for temperature, pressure, and humidity inside containment for the power rerate conditions while Exhibit E (section 10.2.1.1) indicates a slight increase. Provide clarification.

2. Information provided in Exhibit A (page A-24) indicates no change for accident (design-basis accident/loss-of-coolant accident) (DBA/LOCA) conditions for temperature, pressure, and humidity inside containment for the power rerate conditions while Exhibit E (section 10.2.1.1) indicates a slight increase. Provide clarification.

3. Slight increases in the current accident (DBA/LOCA) and normal conditions for temperature, pressure, and humidity for power rerate are considered insignificant as stated in Exhibit E (section 10.2.1.1).

   a. Define the slight increases for temperature, pressure, and humidity.

   b. Explain why the slight increases are considered insignificant.

      (1) Has each piece of equipment been evaluated to ensure it is still qualified?

      (2) Explain why equipment remains qualified.

   c. Section 10.1, Exhibit E, states that these increases are well within the margins in the existing environmental qualification (EQ) envelopes.

      (1) Do these increases cut into test margins or do they cut into the margin between qualification levels and actual predicted profiles?

      (2) Define how margins are being cut.

4. Provide an EQ package for one piece or type of electric equipment that is within the scope of 10 CFR 50.49 which demonstrates (1) continued qualification for the rerate environment and (2) the process for establishing qualification for the increased temperature, pressure, humidity, and radiation levels for power rerate.

Enclosure

5. On page A-58, it is implied that minor modifications (required to assure the continued qualification of electrical equipment outside the scope of 10 CFR 50.49) are not considered unreviewed safety questions and thus will be implemented under the provisions of 10 CFR 50.59 during implementation of power rerate. 10 CFR 50.59 requires, in part, that a proposed change be deemed to involve an unreviewed safety question if the probability of malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. The increase in system temperature, pressures, or power requirements identified to exist on page A-24, Exhibit A, (no matter how minor) can be interpreted to increase the probability of malfunction of heat sensitive electrical equipment important to safety. Thus, any modification to electrical equipment due to increased temperature, pressures, or power requirements associated with MNGP power rerate could be considered an unreviewed safety question. Explain why (or how) these future modifications (which have yet to be identified) should (or will) not be considered unreviewed safety questions.

6. Section 6 of Exhibit E indicates that a Northern States Power (NSP) grid stability analysis has been performed at 1775 MWt to verify no significant effects on grid stability and reliability. Explain why there are "no significant effects on grid stability and reliability."

7. Provide results of analysis which demonstrates that sufficient power will remain available and connected to safety systems from the offsite system (transmission network) immediately following reactor trip caused by LOCA when operating MNGP at 1775 MWt for all expected modes of operation of the transmission network.

8. Provide results of analysis (or other justification) which demonstrates that there has been no reduction in margin (due to power rerate) between trip setpoints for loss of voltage or degraded grid voltage protective schemes installed on safety buses and transient voltage on safety buses that are expected following reactor trip due to a LOCA.

9. Technical specifications will allow plant operation with the 1R and 2R transformers operable while the 1AR transformer is out of service. It is not clear that the 1R and 2R transformers each have sufficient capacity and capability to supply safety-related loads for this mode of operation. It is also not clear if operability requirements need to be established for the automatic load shedding feature on the 1R transformer (or for the administrative procedures for limiting load on the 1R transformer) for this mode of operation. Provide technical specification changes that preclude this mode of operation or provide a system description, the results of analysis that demonstrate compliance with design-basis requirements, and proposed limiting conditions for operation (if applicable) for this mode of operation.

10. Technical specifications will allow plant operation with the 1R and 1AR transformers operable while the 2R transformer is out of service. Provide technical specification changes that preclude this mode of operation or provide a system description, the results of analysis that

demonstrate compliance with design-basis requirements, and proposed limiting conditions for operation (if applicable) for this mode of operation.

11. Page 2 of 3 of Updated Safety Analysis Report (USAR) 8.2 Revision 12 states that the 1R transformer is of adequate size to provide the plant's full auxiliary load requirements. Exhibits A and E of the rerate submittal indicate that the MNGP design has been modified (and will be further modified as part of rerate) such that the capacity of the 1R transformer is something less than the 100 percent capacity required by the licensing basis for MNGP documented in the USAR. Provide the description and analysis for this modification.

Materials Engineering:

12. Provide an assessment of how the proposed thermal uprate will affect the end of life (EOL) upper shelf energy analysis for Vessel Plate No. I-15 (Heat No. C2220-2), and the equivalent margins analyses for Vessel Plates I-16, I-17, and I-14 (Heat Nos. A0946-1, C2193-1, and C2220-1) and the beltline vessel welds (no heats given). Include appropriate calculations, figures, or references demonstrating continued compliance with the requirements of 10 CFR Part 50, Appendix G, under the proposed increased power conditions and also updated values for the 1/4T fluence and the upper shelf energies for beltline materials of the MNGP reactor pressure vessel (RPV) at EOL.

13. Provide an assessment of how the uprated conditions will affect the scope or schedule of the surveillance capsule withdrawal program (10 CFR Part 50, Appendix H Program) for the MNGP RPV.

14. Provide a more detailed evaluation of the effects caused by extended power uprate on reactor internals (i.e., expand on your submitted determination of the effects).

15. Provide a more detailed evaluation of the effect caused by extended power uprate on components exposed to single- and two-phase fluid flow.

Mechanical Engineering:

16. In reference to Section 2.5.1 of Exhibit E, provide an evaluation of the control rod drive mechanism with regard to the stress and fatigue usage as a result of the 6.3 percent power uprate. Also, provide the allowable code limits for the critical components evaluated, and the code and code edition used for the evaluation. If different from the code of record, justify and reconcile the differences.

17. In regard to Section 3.3.2, provide the maximum calculated stress at the critical locations of the reactor internal components, the allowable code limits, and the code and code edition used in the evaluation for the power uprate. If different from the code of record, provide justification.

18. In Section 3.3.2.2, an assessment of flow-induced vibration of the reactor internal components due to power uprate is performed to address the increase in steam product in the core, the increase in the core pressure drop, and the increase in the recirculation pump speed. In that assessment, the vibration levels were estimated by extrapolating the recorded vibration data at Monticello and by using the operating experience of similar plants. Provide a sample evaluation and the basis for using the operating experience of similar plants.

19. In reference to Sections 3.3 and 3.3.2, provide the methodology, assumptions, and loading combinations used for evaluating the reactor vessel and internal components with regard to the stresses and fatigue usage for the power uprate. Were the analytical computer codes used in the evaluation different from those used in the original licensing-basis analysis? If so, identify the new codes used and provide justification for using the new codes and state how the codes were qualified for such applications.

20. In reference to Section 3.6, provide the methodology and assumptions used for evaluating the reactor coolant piping systems for the power uprate. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the code and code edition used in the evaluation for the power uprate. If different from the code of record, justify and reconcile the differences.

21. Discuss the analytical methodology and assumptions used in evaluating pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchors at the power uprate conditions. Were the analytical computer codes used in the evaluation different from those used in the original licensing-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

22. The power uprate fatigue cumulative usage factors (CUFs) (shown in Table 3-4) for the reactor vessel are given in three locations: at the refueling bellows skirt, the closure region bolts, and the recirculation inlet nozzles. Provide CUFs for the limiting components of the reactor coolant piping systems. Discuss how the calculated CUFs for the reactor vessel and piping components compare to the CUFs resulting from the actual loading cycles based on the data recorded during plant operation.

23. Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs)

will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed.

24. In reference to Section 3.13, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchors. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

25. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the code of record, and code edition used for the power uprate conditions. If different from the code of record, justify and reconcile the differences.

26. Referring to Sections 3.6 and 4.1.2, provide the evaluation of piping systems attached to the torus shell, vent penetrations, pumps, and valves that may be affected by the LOCA dynamic loads (pool swell, condensation oscillation, and chugging) considered in the evaluation for the power uprate.

27. In reference to Section 3.8.2, provide a detailed discussion of the effects of the steam flow increase, identified in Table 1-2, on the design-basis analysis of the main steam piping due to main steam isolation valve (MSIV) closure and turbine stop valve (TSV) closure loads. Also, provide an evaluation of MSIV structural integrity and functionality due to the increase in the hydraulic pressure for the higher flow rate following the power uprate, as discussed in Section 4.7 of GE's Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate (NEDC-32523P) (proprietary information - not publicly available).

28. Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate.

29. In reference to Section 7.4, provide the evaluation of the feedwater heater for the power uprate with regard to vibration, stress, and fatigue usage.

30. In Exhibit D, the statement is made that modification to piping or equipment supports for some plant systems due to load changes involves approximately 12 pipe supports. Provide examples of pipe supports requiring modification and discuss the nature of these modifications.

Human Factors:

31. Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?

32. Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate. Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate. Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the power uprate will significantly affect the operator's ability to complete manual actions in the times required.

33. Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?

34. Discuss any changes the power uprate will have on the Safety Parameter Display System (SPDS).

35. Describe any changes the power uprate will have on the operator training program and the plant simulator.

Instrumentation and Controls:

36. For power uprates, the GE setpoint methodology discussed in GE topical report NEDE-31366 has been used to determine instrument setpoints. Therefore, this methodology should be referenced in the basis section of the technical specifications.

37. The submittal does not address the effect of power uprate on instrumentation range/span. Also, Section 5.2.1, Control Systems Evaluation, states that, "process control valves and instrumentation have been evaluated for range and adjustment capability for use at the expected rerated condition. Any required changes will be performed prior to operation at the rerate...." However, the submittal does not identify any such instrumentation and control valves. Provide this information for staff review.

38. Table 5-1 provides changes in the analytical limit for setpoints for the current and power uprate condition. The justification for these changes is based on the assumption that they do not increase the probability and consequences of postulated accidents, or reduce significantly the margin of safety. In order for the staff to arrive at the same conclusion,

information is needed on instrument setpoints and allowable values in addition to the analytical limit for the instrumentation identified in Table 5-1 at both the current and uprate power conditions.

Radiation Protection:

39. Section 8.5.2 of Exhibit E states that "NSP has established successful cobalt reduction, zinc injection, and hydrogen water chemistry programs. These programs and other dose reduction programs will adequately compensate for the possible increases in individual doses due to power rerate." Provide additional information concerning these dose reduction programs, state when these programs were implemented at Monticello, and describe what effect they have had on reducing overall doses at Monticello. Compare the estimated annual reduction in overall doses resulting from the implementation of these dose reduction programs with the estimated annual increase in doses at Monticello due to the proposed power rerate.

40. Section 8.4.2 of Exhibit E states that the power rerate may result in a net increase in the activated corrosion product production due to the increase in activation rate in the reactor region combined with the decrease in filter efficiency of the condensate demineralizers (due to the feedwater flow increase). Describe the magnitude of the estimated increase in activated corrosion products in the reactor piping and describe how this will affect dose rates in the vicinity of this piping. Describe any plans (such as increasing the amount of zinc injection to the reactor coolant system) that you may have to reduce the increased amounts of activated corrosion products in the piping caused by the proposed power rerate.

41. Section II.H.3.b of Exhibit A states that "Reference to containment spray is to be deleted from this discussion." Provide your reasons for not taking credit for containment spray and state whether deleting reference to the containment spray in this section constitutes a change in your accident dose analysis.

42. Exhibit D (p. D-1) states that one of the hardware changes for power rerate will be to modify the Control Room Emergency Filtration Train system "to reduce control room ventilation filter bypass leakage to establish consistency with control room dose calculation inputs." Discuss what you mean by establishing "consistency with control room dose calculational inputs." In Table 9.4 of Exhibit E you state that the estimated thyroid dose in the control room following a LOCA at the rerate power of 1880 MWt would be 13 rem. State what the estimated LOCA thyroid dose in the control room would be (at 1880 MWt) if the control room ventilation filter bypass leakage were not reduced. Provide both the current and the reduced control room bypass leakage figures in cubic feet per minute.

43. Exhibit A (p. A-21) states that, based on a radiological analysis for the proposed rerate, you will improve the efficiencies of the control room emergency filtration system filter and the standby gas treatment system filter. State the current and proposed filter efficiencies and discuss your timetable for making these changes.

44. The table on page A-20 of Exhibit A lists the calculated potential offsite doses at the exclusion area boundary (EAB) and low-population zone (LPZ) from the following design-basis accidents: loss-of-coolant, refueling, control rod drop, and steamline break. These doses were calculated by the AEC staff and are contained in the staff safety evaluation dated March 18, 1970.

    a. For the same four accidents described above, provide a listing of the postulated doses (both whole body and thyroid) at the EAB, LPZ, and control room that were calculated by the licensee during the initial licensing of the plant.

    On page A-21 of Exhibit A, you state that the inputs and evaluation methods for the MNGP power rerate differ from those used in the current licensing basis evaluation contained in the USAR and in the AEC safety evaluation. You state that you have established dose multipliers that should be used to multiply the doses contained in your original licensing basis evaluation to obtain the doses calculated for the MNGP power rerate.

    b. Show how you applied these dose multipliers (listed in Table 14.7-22 of the USAR) to the doses calculated using your current licensing basis evaluation to arrive at the revised accident doses for the proposed power rerate (listed in Table 9-4 (Appendix E) of the rerate licensing amendment request).

45. Describe those plant changes (both operational and hardware changes) made to accommodate the proposed power rerate that will have an effect on the calculated EAB, LPZ, and control room doses following any one of the following design-basis accidents: loss-of-cooling, refueling, control rod drop, and steamline break.

Probabilistic Risk Assessment:

46. On page 10-8, the last paragraph states "rerate analysis did require two SRVs [safety-relief valves] to open to avoid reactor overpressure whereas only one SRV is adequate for the 100% power level case." How was this change reflected, if any, in the risk analysis and how (much) did it contribute toward the estimated increase in plant core damage frequency (CDF)?

47. On page 10-9, 2nd paragraph under the section, "Time Available for Operator Action," states that "the most important post-initiator human errors were recalculated using the method described in NUREG/CR-4772 ["Accident Sequence Evaluation Program Human Reliability Analysis

Procedure"] for deriving nominal human error probability estimates (ASEP [Accident Sequence Evaluation Program] method). Please describe the increase in human error rates of the most impacted operator actions due to the power rerate by providing their "current" human error rates as well as the "new" human error rates that were estimated using the above method.

For example, on page 10-15, the first paragraph states "the time for the operator to initiate SBLC is reduced from approximately 21 minutes to 13 minutes. In spite of the reduction in time to perform this action, the likelihood of the operator correctly performing this action is still high." Please provide the change in human error rates associated with the change in requirement to initiate SBLC from 21 minutes to 13 minutes and show how this change impacts the analysis results.

As another example, on page 10-14, the fourth paragraph states "a large portion of the CDF due to high pressure core damage sequences result from internal flood initiator events." On page 10-15, the fourth paragraph states "this is due to the decrease in the time available for the operator to blowdown the vessel before the core becomes uncovered." How is two thirds (approximately 1.6E-6/Yr) of the CDF increase attributed to operators' ability to respond to these sequences?

48. On page 10-11, Section 10.5.3.2, "Internal Events PRA - Level 2 (Containment Analysis)", the first paragraph states that the requantified results of the Level 2 portion of the PRA update was not available at the time of this analysis. Please provide the quantitative results of the Level 2 analysis. If available, please also provide the quantitative results of the risk analysis for external events.

49. On page 10-14, the second paragraph states "Human Reliability Analysis (HRA)....This portion of the PRA involves some of the largest uncertainty in failure probability estimates...." In view of such uncertainty in the HRA and since the CDF increase of 2.4E-6/Yr (or 17.5 percent increase from the Monticello baseline CDF) is not considered insignificant, the staff needs to review the uncertainty analyses to understand how uncertainties were addressed, both quantitatively and qualitatively, in the decisionmaking process.

Containment Systems:

50. It is indicated in USAR Table 5.2-4 that maximum drywell pressure is 42.0 psig rounded off to the nearest psi. In NEDC-32546P ("Power Rerate Safety Analysis Report for Monticello Nuclear Generating Plant," proprietary information - not publicly available) Table 4-1 it is stated that peak drywell pressure is 41.0 psig at 102 percent of 1670 MWt using Mark I long-term program (LTP) method and 39.0 psig using Mark I LTP method with break flow from more detailed RPV Model. Please discuss the reasons for the difference between the USAR and MARK I numbers. Also discuss the reasons why the pressure goes up by only 1 psig to 40 psig when power is raised from 1670 MWt to 1880 MWt using the same method.

Please confirm that if pressure is rounded off, it is rounded to the next higher number. Please indicate key input parameters besides power related that are different from the USAR and the effects on peak pressure.

51. It is indicated in 4.1.1.1 for emergency core cooling system (ECCS) net positive suction head (NPSH) that the decrease in NPSH due to the increase in the long-term bulk suppression pool temperature at uprated power will be offset by the suppression pool airspace pressure increase. Please provide the specific numbers.

52. Please provide the confirmatory calculations validating the results from the analyses using the SHEX computer code.

Reactor Systems:

53. In Section 3.2 of Exhibit E, did the overpressure analysis assume 102 percent of rerated power, 105 percent rerated steam flow, and an SRV opening tolerance of 3 percent?

54. In Section 3.5 of Exhibit E, the licensee should commit to performing the vibration monitoring of the reactor recirculation system (RRS) as stated in the GE generic report in Section 5.5.1.3 and the review of the plant operating data as specified in Section 5.6.2 to confirm that the RRS will accommodate the uprated flow conditions.