November 24, 2000

Mr. Michael F. Hammer
Site General Manager
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - CLARIFICATION OF THE SAFETY EVALUATION RELATED TO AMENDMENT NO. 102 (TAC NO. MA9604)

Dear Mr. Hammer:

The NRC approved Northern States Power Company's (NSP's) request for power uprate of the Monticello Nuclear Generating Plant by issuing Amendment No. 102, dated September 16, 1998, to Facility Operating License No. DPR-22. Following its review of the safety evaluation enclosed with Amendment No. 102, NSP noted several areas which it requested be clarified. The NSP comments were documented in a letter to the NRC from Mr. B. Day dated July 18, 2000. NSP stated that the comments were minor in nature and did not modify previous NSP correspondence or commitments, and requested NRC concurrence with the comments. The staff has reviewed Mr. Day's letter and agrees with NSP's characterization and resolution of the issues. Consequently, please revise the safety evaluation for Amendment No. 102 by removing original pages 4, 20, 26, 50, 51, 73, and 74 and inserting the replacement pages from the enclosure to this letter. None of the comments made by NSP changed the conclusions of the staff expressed in the safety evaluation accompanying Amendment No. 102.

If you have any questions, please call me at 301-415-2296.

Sincerely,

/RA/

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure: Revised safety evaluation pages

cc w/encl: See next page

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REVISIONS TO SAFETY EVALUATION FOR LICENSE AMENDMENT NO. 102

Replace the following pages of the safety evaluation for License Amendment No. 102, dated September 16, 1998, with the attached revised pages.

REMOVE	INSERT
4	4
20	20
26	26
-	26a
50	50
51	51
73	73
74	74

ENCLOSURE

the core will be changed to achieve increased core power while limiting the absolute power in any individual bundle. Increased fuel enrichments or higher batch fractions may be used to provide additional operating flexibility.

Thermal-hydraulic design and operating limits assure an acceptably low probability of boiling transition occurring in the core anytime, even for the most severe postulated operational transients. Limits are also placed on fuel average planar linear heat generation rates to meet both peak cladding temperature (PCT) limits for the limiting loss-of-coolant-accident (LOCA) and fuel mechanical design bases. Subsequent core reloads at power uprate will also take into account these limits to assure acceptable margins between the licensing limits and their corresponding operating values. At power uprate conditions, all fuel and core design limits will continue to be met by control rod pattern adjustments. New fuel designs are not needed for power uprate to assure adequate safety. However, different fuel enrichment distributions may be used to provide additional operating flexibility and maintain cycle length.

2.1.b Thermal Limits Assessment

Fuel operating limits, such as the maximum average planer linear heat generation rate (MAPLHGR) and safety limit minimum critical power ratio (SLMCPR) for future reloads will continue to be met after power uprate. The methods used for calculation of MAPLHGR and operating limit minimum critical power ratio (OLMCPR) limits will not be changed because of power uprate, although the actual thermal limits may vary between cycles. Cycle specific thermal limits will be included in the Core Operating Limits Report. A representative cycle core is used for the uprate evaluation. These evaluations showed no change is required in the SLMCPR or the MAPLHGR and LHGR limits for power uprate.

2.1.c Reactivity Characteristics

All minimum shutdown margin requirements that apply to cold (212° F or less) conditions, will be maintained without change. Operation at higher power could reduce the excess reactivity during the cycle. This loss of reactivity is not expected to significantly degrade the ability to manage the power distribution through the cycle to achieve the uprated power level. The lower reactivity will result in an earlier all-rods-out condition. Any reduction in operational shutdown margins may need to be accommodated through core design. The technical specification requirements for shutdown margin will continue to be met.

(1) Power/Flow Operating Map

The power uprate flow map is shown in Figure 2-1 of NEDC-32546P (Ref. 21). Changes to the power/flow operating map are consistent with the generic descriptions given in Sections 5.2 and C.2.3 of NEDC-32424P (Ref. 16). The maximum thermal operating power and maximum core flow shown on Figure 2-1 correspond to the uprated power and the analyzed core flow range when rescaled so the uprated power is equal to 100 percent rated. Power uprate raises the upper portion of the core operating map (reactor power versus core flow) along the current rod/flow control lines. These lines have not changed but have been renamed to reflect the redefinition of rated thermal power. Full power operation under the maximum extended operating domain (MEOD), which was previously achieved at a minimum value of

associated loads. The staff has previously accepted the use of the LAMB code to model the RPV break flow in containment analyses for power uprate.

The licensee indicated that the SHEX code was used to model the long-term post-LOCA containment pressure and temperature response. The results of the benchmark analyses of the SHEX code to the HXSIZ code (the code used in the current licensing-basis analyses) at power levels of 1670 MWt and 1880 MWt were provided by the licensee. The benchmark analyses were performed using the May-Witt decay heat model and the ANS 5.1 nominal decay heat model. Using the May-Witt decay heat model, the peak suppression pool temperature was predicted to be 207.2 °F with the SHEX code and 207.6 °F with the M3CPT/HXSIZ code. Using the ANS 5.1 nominal decay heat model, the peak suppression pool temperature was predicted to be 193.6 °F with the SHEX code and 194.0 °F with the M3CPT/HXSIZ code. The results of the analyses demonstrated that the peak suppression pool temperature predicted with the SHEX code are within 1 °F of the peak pool temperature predicted with the M3CPT/HXSIX code. Based on the review of the benchmark analyses results, the staff finds the use of the SHEX code acceptable for MNGP power uprate analyses.

2.5.b <u>Long-term Suppression Pool Cooling Temperature Response</u>

(1) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA. A bounding analysis was performed at 102 percent of 1880 MWt using the SHEX code and the ANS 5.1 nominal decay heat model. The staff has determined that a 2σ adder (95 percent confidence interval) is necessary for the use of the ANS 5.1-1979 nominal decay heat model to account for the uncertainty. In a letter dated May 5, 1998 (Ref. 9), the licensee provided a comparative study between the generic 1880 MWt shutdown decay heat profile used for containment analyses and the MNGP-specific shutdown power profile for 1775 MWt with a 2σ adder. The comparative study shows that the nominal integrated energy at 1880 MWt bounds the integrated energy at 1775 MWt with the 2σ adder for the first 30 days post-LOCA. Therefore, it is reasonable to conclude that the generic 1880 MWt decay heat profile used in the power uprate containment analyses bounds the MNGP-specific 1775 MWt decay heat profile with the 2σ adder. Based on the above, the staff finds the bounding 1880 MWt nominal decay heat model acceptable for the proposed power uprate to 1775 MWt.

The licensee indicated that the long-term containment analysis was performed with the most limiting set of assumptions including the assumption of availability of containment cooling equipment (i.e., 1 RHR pump, 1 RHR service water pump, and 1 RHR exchanger) and the assumption of the maximum ultimate heat sink temperature. The use of the containment sprays was not assumed in this analysis. The analysis shows that, using the SHEX code and the ANS 5.1-1979 decay heat model as described above, the power uprate would increase the peak pool temperature by 8 °F, resulting in a DBA-LOCA peak suppression pool temperature of 194 °F. This is below the torus attached piping limit of 195 °F and the suppression chamber design temperature of 281 °F.

The licensee stated that the increased suppression pool temperature and pressure were analyzed for the potential impact on the NPSH for the ECCS pumps that draw water from the

4. TS Table 3.2.1, Function 3a, Reactor Cleanup System, Low Reactor Water Level -

   Trip setting has been changed from ≥ 10.6" above the top of the active fuel to ≥ 7" annulus.

5. TS Table 3.1.1, Function 7, Reactor Low Water Level -

   Trip setting has been changed from ≥ 7 in. (6) to ≥ 7 in. (annulus) and note 6 was deleted. Note 6 states that 7" of water level instrumentation is 10'6" above the top of the active fuel at rated power.

6. TS Table 3.2.3, Function 3, Rod Block -

   For two loop operation, trip setting has been changed from ≤0.66W + 58% to ≤0.66W + 53.6%.

   For single loop operation, trip setting has been changed from ≤0.58(W-5.4) + 50% to ≤0.66(W-5.4) + 53.6%.

7. TS Section 2.3.A.1.a and 2.3.A.1.b APRM Scram -

   For two loop operation, trip setting has been changed from ≤0.66W + 70% to ≤0.66W + 65.6%.

   For single loop operation, trip setting has been changed from ≤0.58(W-5.4) + 62% to ≤0.66(W-5.4) + 65.6%.

In addition to the above changes, the licensee will implement new set points for the instrumentation that is listed in the TS as a percentage of flow or pressure, as the actual set point of these instruments will change although the percentage has not been changed. The licensee has identified this instrumentation as follows:

(a) Main steam line high flow ≤140% rated
(b) Automatic bypass of turbine control valve fast closure and turbine stop valve scram is effective below 30% thermal power as indicated by turbine first stage pressure.
(c) APRM flux scram trip setting shall be no greater than 120%.

The licensee has also revised the associated TS Bases to incorporate the changes to the TS. In addition to these changes, the licensee has made some editorial and administrative changes to the TS to incorporate values based on the new thermal power level.

The licensee's submittal of July 26, 1996 (Ref. 1), identified that GE Licensing Topical Report NEDC-31336, “General Electric Instrumentation Setpoint Methodology,” dated October 1986, was used for the instrument set point calculations. The staff has previously accepted the NEDC-31336 for instrument set point calculations in a safety evaluation dated February 9, 1993, and found it acceptable for establishing new set points in power uprate applications.

By letters dated April 14, 1997, and February 11, 1998, the staff requested additional information regarding set point margins for the new thermal power level. The licensee in its letters dated September 5, 1997 (Ref. 2), and March 6, 1998 (Ref. 4), provided the requested information. The proposed set point changes resulting from the power uprate are intended to maintain the existing margins between operating conditions and the reactor trip set points and

In its evaluation of the radiological consequences due to the MSIV leakage following a postulated LOCA, the staff allowed a credit for iodine holdup for decay and iodine deposition for plate-out in the main steam lines, the steam drain lines, and main condensers. This is a deviation from the SRP. The licensee also claimed a similar credit in its analyses using the methodologies and models developed by GE.

Section III(c) and VI of Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after an SSE. Thus, the main steam line, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. Consequently, these components have been evaluated as described in Section 4.0 to assure that they would retain sufficient structural integrity following a safe shutdown earthquake to transport main steam isolation valve leakage to the condenser. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after occurrence of an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

For the purpose of providing a credit for iodine holdup and plate-out, the staff requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products.

The licensee provided additional information regarding the seismic verification of the MSIV leakage path in a separate submittal (Ref. 12) in response to the staff's request. The licensee concluded that the MNGP design provides reasonable assurance that the main steam piping from the outboard isolation valve up to the turbine stop valve, the main steam drain lines up to the condenser, and the main condenser will remain structurally intact; therefore, they can act as a holdup volume for fission products during and following an SSE. The staff’s review of this area is documented in Section 4.0 of this safety evaluation.

The licensee submitted the site meteorological data and calculated atmospheric dispersion factors (X/Q values) (Ref. 30). The licensee stated that these meteorological data, analysis, and X/Q values are also applicable to the power uprate radiological consequence analysis. In the submittal, the licensee stated that it has used the methodology described in Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” for determining the site boundary and low population zone X/Q values and for calculating the X/Q values for the release from the offgas stack to the control room intake. The licensee used the methodology described in NUREG/CR-5055 (Ref. 28) for calculating the ground level release control room intake X/Q values. The staff has not accepted the methodology in NUREG/CR-5055 that has been revised into the ARCON96 methodology described in NUREG/CR-6331, Rev. 1 (Ref. 29).

The staff independently calculated X/Q values for the site boundary and low population zone using the methodology described in Regulatory Guide 1.145 and for the control room air intake using the ARCON96 methodology. The staff has found the licensee’s X/Q calculations for the offgas stack and turbine building releases to be adequately conservative for this assessment. For the postulated turbine building release, fission products are conservatively assumed to be

released at a point located in the center of four sealed off roof exhauster openings on the turbine building roof closest to the control room air intake.

The staff finds that the differences in the control room X/Q values calculated by the licensee and staff are within the uncertainty ranges of mixing of fission products with air in the turbine building prior to release to the environment. The staff has not provided and the licensee has not claimed any mixing credit in the turbine building. The staff used the X/Q values calculated by the licensee in the staff dose assessment for this power uprate analysis. The resulting radiological consequence analyses are provided in Table 3.5-1 and the major parameters and assumptions used by the staff are provided in Table 3.5-2 through Table 3.5-6.

The staff concludes that the proposed power uprate at MNGP will provide reasonable assurance that the radiological consequences of bounding DBAs will not exceed dose acceptance criteria specified in the SRP, 10 CFR Part 100, and GDC 19 of Appendix A to 10 CFR Part 50. This conclusion is based on the staff's review of the radiological consequence analyses submitted by the licensee and the staff's independent confirmatory analyses. Therefore, the staff finds that the proposed power uprate is acceptable.

Tables 3.5-1 through 3.6-6 follow.

which includes training on the Monticello plant simulator. In addition, multiple indications and alarms in the control room were cited as providing assistance in following the procedures.

Based on the reported analysis and results, the staff agrees that the resulting change in CDF (internal events) is mainly due to increase in human error rates which reflect decreased time available for accident mitigating operator actions. The staff believes that although virtually no significant change in initiating event frequencies, success criteria and component failure rates are predicted at this time, it remains to be seen whether these attributes will indeed be unaffected by uprated power level operation in the future. However, based on the information available at present, the staff believes that the reported increase in CDF (internal events) is small. Therefore, the staff considers the change in CDF for internal events due to the requested power increase by 6.3 percent to be acceptable.

5.2 Level 2 Internal Events PRA

The licensee reported that from the baseline (100-percent power level) level 2 PRA results, the potential for a large early release is small, on the order of 3 percent of the total CDF. As with other Mark I containments, large early releases for Monticello are dominated by ATWS and interfacing LOCA sequences.

For the bounding uprated power level at 112 percent, the licensee determined that the large early release frequency (LERF) was approximately 3% of CDF, the same percentage as for the baseline. Since the CDF for the uprated power level increased slightly compared to the baseline, the uprate LERF also increased slightly. The changes in the Level 2 quantification resulted from the changes made to the Level 1 accident sequence analysis due to reduced time available for operator recovery action. The ATWS sequences dominate the increase in large early releases due to the shorter time available to the operator to initiate standby liquid control. The major contributors to large early releases remained the same as in the baseline analysis and include ATWS, hydrogen combustion, and interfacing LOCA sequences. As in the base case analysis, the majority of the Level 2 accident sequences either do not result in containment failure, are vented or released through a pool, or are estimated to occur many hours into the accident. Based on the small increase in LERF, the staff considers the change in LERF due to requested power increase by 6.3 percent to be acceptable as it meets the criteria of DG-1061.

5.3 Internal Fire, Seismic, and Other External Events PRA

The CDF contribution from internal fires increased from 8.34E-6/Year to 8.8E-6/Year. This was attributed solely to the increase in human error rates because the time available to perform various accident mitigating tasks decreases with uprate. The decrease in time available with uprate is due to higher core decay heat increasing the steaming rate and thus leading to an earlier core uncovery. A majority of the change in CDF occurs due to scenarios involving core damage occurring at high pressure. This is attributed to the decrease in the time available for the operator to blow down the vessel before the core becomes uncovered. The remaining CDF increase involves sequences related to long-term containment heat removal, and a reduction in time to repair failed decay heat removal equipment (from 27 hours to 24 hours).

The licensee reported that there were no changes to the plant's capability to cope with a seismic event due to the power uprate. In addition, the potential for and capability of the plant to withstand "other" external event initiators were found to be essentially unaffected by the power uprate. A sensitivity study performed for tornado missiles showed that the difference in available operator response times resulted in a negligible change in CDF.

Based on these reported changes in CDF due to internal fire, seismic event, and other potential external initiators, the staff considers the CDF change to be small. Therefore, the staff considers the CDF change for these events due to the 6.3 percent increase in power to be acceptable.

5.4 Quality of PRA

The licensee's original IPE was submitted to the NRC in 1992 and the staff's safety evaluation accepting the submittal was issued by the staff in 1994. As stated in the safety evaluation, the staff found that (1) the IPE was complete with respect to the information requested in Generic Letter 88-20 and associated supplement 1, (2) the analytic approach was technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding, (3) the licensee employed a viable means to verify that the IPE models reflect the current plant design and operation at time of submittal to the NRC, (4) the IPE had been peer reviewed, (5) the licensee participated in the IPE process, (6) the IPE specifically evaluated the decay heat removal function for vulnerabilities, and (7) the licensee responded appropriately to the Containment Performance Improvement program recommendations. Based on these findings, the staff concluded that the licensee met the intent of Generic Letter 88-20 ("Individual Plant Examination on Severe Accident Vulnerability").

The CDF reported in the original IPE was 2.6E-5/Year. The latest updated PRA (baseline) estimated the CDF at 1.37E-5/Year. The licensee attributed this decrease in CDF to changes in the model as well as improvements that have been made since the IPE. These changes include (1) diesel generator 13 backfeed through emergency bus 15 to supply battery chargers, (2) installation of the hard pipe vent which provides an additional means for containment heat removal, (3) improvements to SRV pneumatics (including power supplies), (4) diesel fire pump as an additional source of low pressure makeup water, (5) addition of air compressor 14 which is not dependent on service water, (6) success criteria for service water changing from 2 pumps to 1 pump, and (7) updated internal floods analysis.

The licensee reported that the internal events PRA used for the power uprate evaluation is based on a more current version of the PRA than the version used for the IPE. Although the licensee did not provide a full documentation of their PRA, a review of their submittal pertaining to PRA for power uprate as well as information contained in the original IPE submittal and the safety evaluation provided sufficient indication to the staff that the licensee's PRA and their analysis for power uprate are adequate to support the power uprate request.

cc:

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