Northern States Power Company

Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

NSP

July 18, 2000

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

10 CFR Part 50
Section 50.90

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Request for NRC Concurrence dated July 18, 2000
Comments on NRC Power Uprate Safety Evaluation
and
License Amendment Request dated July 18, 2000
Alternate Shutdown System Operability Requirements

Reference 1: NRC letter to NSP, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Power Uprate Program (TAC No. M96238)," dated September 16, 1998.

Attached is a request for concurrence with comments on the Reference 1 NRC Safety Evaluation (SE) for the Monticello Power Uprate Program, License Amendment 102. Also attached is a license amendment request which proposes a change to the Technical Specifications, Appendix A of the Operating License for the Monticello Nuclear Generating Plant. This request is submitted in accordance with the provisions of 10 CFR Part 50, Section 50.90.

During Northern States Power (NSP) review of the Reference 1 NRC SE for License Amendment 102, several areas were noted which should be clarified. The NSP comments are included in Exhibit A to this letter. The comments are minor in nature and do not modify previous NSP correspondence or commitments. NSP requests NRC concurrence with the comments.

The proposed amendment changes Technical Specification (TS) 3.13.H, Alternate Shutdown System (ASDS), to include specific operability requirements for 12 Residual Heat Removal Service Water (RHRSW) Pump. 12 RHRSW Pump is operated from the ASDS Panel to achieve safe shutdown in the event of a Control Room fire.

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Specification 3.13.H requires system controls on the ASDS panel to be operable whenever that system/component is required to be operable. License Amendment 102 revised the TS 3.5.C operability requirements for Containment Spray/Cooling subsystems such that only one of two RHRSW Pumps is required to consider one train of Containment Spray/Cooling to be operable. Thus, 12 RHRSW Pump is not specifically required to be operable. The proposed amendment specifically requires 12 RHRSW Pump to be operable from the ASDS Panel.

Northern States Power Company (NSP), a Minnesota corporation, requests NRC concurrence with NSP comments on the NRC SE for License Amendment 102, as discussed in Exhibit A. Exhibit B is a mark-up of the Reference 1 SE, showing the incorporated comments. NSP also requests authorization for a change to Appendix A of the Monticello Operating License as shown on the attachments labeled Exhibit C, D, and E. Exhibit C contains a description of the proposed TS change, the reasons for requesting the change, a Safety Evaluation, a Determination of No Significant Hazards Consideration, and an Environmental Assessment. Exhibit D contains the current Technical Specification pages marked up with the proposed change. Exhibit E contains revised Monticello Technical Specification pages.

NSP requests a period of up to 45 days following receipt of this license amendment to implement the changes.

This letter does not contain any new NRC commitments and does not modify any prior commitments. This letter contains no restricted or other defense information. Please contact Doug Neve, Sr. Licensing Engineer, at (763)-295-1353 if you require further information related to this request.

To the best of my knowledge, information and belief, the statements made in this document are true and correct.

by Byron D. Day

Byron D. Day
Plant Manager
Monticello Nuclear Generating Plant

Signed before me on this 18TH day of July, 2000 by Byron D. Day, Plant Manager, Monticello Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company.

Marcus H Voth

Marcus H. Voth
Notary Public - Minnesota
Wright County
My Commission Expires January 31, 2005

MARCUS H. VOTH
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments: next page
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Attachments:

Exhibit A – Northern States Power Comments on License Amendment 102 NRC Safety Evaluation

Exhibit B –NRC Safety Evaluation of Monticello License Amendment 102 Marked Up With NSP Comments

Exhibit C – Evaluation of Proposed Change to the Monticello Technical Specifications

Exhibit D – Current Monticello Technical Specification Pages Marked Up With Proposed Change

Exhibit E – Revised Monticello Technical Specification Pages

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J Silberg, Esq.

Exhibit A

Request for NRC Concurrence Dated July 18, 2000
Northern States Power Comments on License Amendment 102 NRC Safety Evaluation

Background

In response to several Northern States Power (NSP) submittals, Reference 1 issued NRC approval of Amendment 102 to the Monticello Nuclear Generating Plant Facility Operating License. The amendment changes the maximum reactor core thermal power level specified in the Operating License from 1670 megawatts-thermal (MWt) to 1775 MWt. The amendment also approves changes to the Operating License Appendix A Technical Specifications to support uprated power operation. Enclosure 2 to Reference 1 provided the NRC staff safety evaluation (SE) of the amendment.

NSP has reviewed Enclosure 2 to Reference 1 and has several comments. The specific comments and rationale follow. A markup of the NRC SE of Amendment 102, showing the comments discussed below, is included in Exhibit B.

Discussion

The following discussion summarizes the NSP comments on the Reference 1 SE for License Amendment 102. Recommended wording changes are included. None of the comments present new information and they do not affect the bases for NRC conclusions in the SE.

1. Section 2.1.b, "Thermal Limits Assessment," page 4

This paragraph should be revised to read:

> "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."

The Technical Specification SLMCPR was used as acceptance criteria in the thermal limits calculations for the uprate. The reload safety analysis calculates OLMCPR for each reload to ensure that the SLMCPR is not exceeded. Thus, while no change in SLMCPR was required for the uprate, the OLMCPR may vary between cycles.

2. Section 2.5.b.(1), "Long Term Suppression Pool Cooling Temperature Response, Bulk Pool temperature," page 20

Revise the last sentence in the second paragraph to read:

> "This is below the torus attached piping limit of 195°F and the suppression chamber design temperature of 281°F."

The only design temperature limit for the suppression chamber is 281°F; the 195°F limit only applies to piping attached to the torus.

3. Section 2.7, "Instrumentation and Control," Item 6, page 26

Revise item 6, add item 7, and clarify setpoint changes that are a percentage of flow or power as follows:

"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 power, 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%"

It is appropriate to address the APRM scram as a separate item 7, since the equation for determining the set point changed and Technical Specification Section 2.3 is affected.

The change to the last paragraph is made to clarify the setpoints that were changed based on the change in power.

4. Section 3.5, "Radiological Analyses for Design-Basis Accidents," page 50

Revise the third sentence of the second paragraph on page 50 to read:

> "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."

The main steam line downstream of the main steam isolation valves (MSIVs) to the main condenser and associated piping were not designed to seismic Category (Class) I and are not safety related. The Monticello licensing basis does not require assumption of simultaneous seismic event and loss of coolant accidents (LOCA). As discussed in Section 4.0 of Reference 1 (included in Exhibit B, attached), the piping has been evaluated in accordance with BWROG methodology to assure an intact flow path for main steam isolation valve leakage following a LOCA. The proposed change to Section 3.5 reflects the discussion in Section 4.0, which forms our basis for the acceptability of the MSIV leakage path.

5. Section 3.5, "Radiological Analyses for Design-Basis Accidents," page 50

On pages 50 and 51, revise the last sentence in the sixth paragraph to read:

> "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 exhausters have been removed and the openings sealed off to eliminate an unmonitored release point. It is conservative to assume this release point is still available.

6. Section 5.2, "Level 2 Internal Events PRA," page 73

Revise the first and second sentences of the second paragraph to read:

> "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 uprate and baseline LERF values (4.8E-7 and 4.1E-7, respectively) included in the NRC SE do not correspond to the actual calculated values. The values included in the NRC SE were derived by applying 3% to the reported CDF values. The licensing submittals stated that the LERF was approximately 3% of CDF both before and after

uprate. The actual values were not specifically cited. Since the actual values were not formally submitted, it is appropriate to revise the NRC SE as suggested above.

7. Section 5.4, "Quality of PRA," page 74

Revise the first sentence of the second paragraph to read:

> "The CDF reported in the original IPE was 2.6E-5/year."

As stated in GE Licensing topical report NEDC-32546P (Reference 21 to the NRC SE), the original core damage frequency reported in the individual plant evaluation (IPE) was 2.6E-5; the updated estimate of CDF was 1.37E-5. After the uprate, CDF is 1.61E-5.

Additional Discussion

Anticipated Transient Without Scram (ATWS) and operator ability to initiate the Standby Liquid Control (SBLC) System is discussed in Section 2.13 ,"Human Factors," page 45 and Section 5.1, "Individual Plant Evaluation, Internal Events," page 72 of the Reference 1 NRC SE. Page 72 of the SE states:

> "The remaining third of the increase in CDF was attributed to ATWS sequences. A major portion of the ATWS contribution is characterized by a turbine trip with turbine bypass to the main condenser. For most ATWS scenarios, feedwater would continue to operate and energy is released to containment due to the relatively limited turbine bypass capacity at Monticello (about 15% bypass at 1670 MWt) without SBLC injection. For the turbine trip with turbine bypass ATWS scenario, the licensee reported that 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 was estimated to be still high. As part of emergency operating procedure training that is discussed above, the licensee noted that the operators are trained on this particular sequence in the classroom and at the Monticello simulator."

The NRC SE concludes that the reduction in time available to the operator and the change in CDF are acceptable.

This issue was discussed at length at an NSP presentation to the Advisory Committee on Reactor Safeguards (ACRS) on June 2 and 3, 1998. Subsequent to the ACRS presentation and issuance of the NRC SE, and to fulfil a license condition to monitor plant parameters for impact on the PRA models, NSP further investigated the issue. A summary of the results of the investigation are provided below for information. No response is requested from NRC on the discussion below. Amendment 102 was approved on the basis of the previous submittals and the results of our investigation show that the total change in CDF is less than previously thought.

After further investigation NSP determined that power uprate does not change ATWS power level or the time available for operator action during ATWS events. Therefore, the change in CDF was overestimated and the actual change in CDF due to power uprate is about one third less than we previously had thought.

Previously it was thought that the CDF due to ATWS would increase after power uprate because the higher operating power would mean a higher ATWS power, and therefore less time available for the operator to inject standby liquid control -- a change from 21 minutes to 13 minutes. An analysis of the ATWS event based on the power to flow map (power after recirculation pump trip) provides a better representation of plant response. Increased power was achieved by increasing recirculation flow rate along the same recirculation flow control line. Therefore, reactor power during an ATWS with natural circulation will be only very slightly higher after uprate. Thus, our analysis should have concluded that the time for operator action before and after uprate is the same for ATWS events and there is no contribution to change in CDF due to ATWS after uprate.

Conclusion

NSP requests NRC concurrence with the clarifications to the Reference 1 SE issued to support Monticello License Amendment 102. The changes do not affect the previous submittals which form the bases for the NRC SE of Amendment 102. A markup of the NRC SE of Amendment 102 showing the comments is included in Exhibit B.

Reference

1. NRC letter to NSP, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Power Uprate Program (TAC No. M96238)," dated September 16, 1998.

Exhibit B

NRC Safety Evaluation of Monticello License Amendment 102
Marked Up With NSP Comments

Request for NRC Concurrence dated July 18, 2000

Exhibit B consists of a markup of the License Amendment 102 NRC safety evaluation, showing the NSP comments discussed in Exhibit A. The pages included in this exhibit are as listed below:

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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.

operating limit minimum

2.1.b Thermal Limits Assessment

S critical power ratio

safety

Fuel operating limit~~s~~, such as the maximum average planar linear heat generation rate (MAPLHGR) and ~~operating~~ limit minimum critical power ratio (ØLMCPR) for future fuel reloads will continue to be met after power uprate. The methods used for calculation of MAPLHGR and 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 plant Core Operating Limits Report. A representative cycle core is used for the uprate evaluation. These evaluations showed no change is required in the ~~safety limit or~~ ØLMCPR, or the MAPLHGR and LHGR limits for power uprate.

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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 is 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 ~~peak bulk pool design temperature~~ 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. <insert attached>

In addition to the above changes, the licensee will implement new set points for the instrumentation that is listed in the TS as 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:

c ~~(a)~~ APRM flux scram trip setting shall be no greater than 120%

~~(b) Turbine first stage scram bypass~~

a ~~(c)~~ Main steam line high flow ≤ 140% rated

b ~~(d)~~ Automatic bypass of ~~T~~turbine control valve fast closure and turbine stop valve scram is effective below 30% thermal power as indicated by turbine first stage pressure.

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

Insert for page 26:

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 $\leq$0.66W + 70% to $\leq$0.66W + 65.6%.

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

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 the main condenser. 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 ~~are required to be classified as safety related and seismic Category I~~. 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. 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.
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 the row of four sealed off roof exhauster openings ~~exhausters~~ 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 with the proposed power uprate at MNGP there is 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 to be acceptable.

Tables 3.5-1 through 3.6-6 follow.

<No change; info only>

4.0 SEISMIC VERIFICATION OF THE MSIV LEAKAGE PATH

The Monticello power uprate radiological analysis takes credit for deposition and holdup of radioactive iodine in the steam lines downstream of the main steam isolation valves (MSIVs) and in the main condenser. The main condenser and the pathway from the MSIVs were evaluated to assure that they would retain sufficient structural integrity following a safe shutdown earthquake (SSE) to transport the MSIV leakage to the condenser.

Because the original design basis of certain main steam system piping, equipment, and components that comprise the leakage pathway is not in accordance with Seismic Category I requirements, NSP has performed evaluations and seismic verification walkdowns to demonstrate that these main steam system piping, equipment, and components are seismically rugged.

The licensee used methodology suggested in the BWROG [Boiling Water Reactor Owners' Group] Report, NEDC-31858P, Rev. 2, entitled "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Contro' Systems," (Ref. 31), to seismically evaluate this pathway. The licensee's submittal of June 15, 1998 (Ref. 12) discusses the applicability of this methodology for Monticello and summarizes the seismic evaluation that was performed for the piping systems and equipment in the MSIV leakage path for Monticello. The licensee stated that a reliable pressure boundary can be maintained in the pathway for the MSIV leakage to reach the condenser during and after an SSE seismic event.

The BWROG report has not been approved by the staff. However, based on a preliminary review to date, the staff has found the BWROG approach of utilizing the earthquake experience-based methodology to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and main condensers, in addition to supplemental plant-specific seismic evaluations, to be acceptable for this amendment request.

The above methodology relies, in part, on the use of earthquake experience data and similarity principles. In addition, plant-specific analyses of piping and equipment was used in combination with the experience database method. Guidance on the use of experience database method for qualification of piping systems is described in Reference 31, and in the supporting documents cited therein. The Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) described in Reference 32 that was developed for the implementation of Unresolved Safety Issue (USI) A-46, was used to demonstrate the seismic ruggedness of certain existing equipment in the MSIV leakage path.

For Monticello, the primary components in the MSIV leakage path that are relied upon for pressure boundary integrity are the main condenser, main steam lines from the MSIVs to the turbine stop valves and to the turbine bypass valves, and the drain lines to the condenser. The drain lines originate from each of the four main steam lines. These drain lines are located downstream of the MSIVs and connect into a drain header that connects to the condenser. The leakage path utilizes three separate drain lines from the main steam piping to the drain header. These three drain lines include the main steam drain lines, the main steam cross tie drain, and the turbine bypass line drain. Each of these lines can be isolated by motor-operated valves (MOVs). Each MOV has a bypass line with a restricting orifice. Since the MOVs are not

< No changes; info only >

powered by essential power and are normally closed, it is assumed that the leakage will be through the MOV bypass lines via the restricting orifices. This provides a passive pathway for the MSIV leakage to reach the condenser because no valve positioning or operator action is necessary to establish the pathway. Therefore, periodic testing to demonstrate valve operability is not required.

The branch lines which interconnect with the MSIV leakage path are included in the scope of the system piping that is reviewed. These branch lines include the connection from the pathway to locations such as a closed valve that would assure that the MSIV leakage would be confined within the branch lines, and leakage would be transferred to the condenser.

The turbine bypass valves are normally closed and fail closed. Because these types of valves are not well represented in the experience data, the licensee conservatively assumed that the valves would fail open as a result of a postulated seismic event, and leakage would therefore go past the turbine bypass valves to the condenser. This portion of the piping was, therefore, also included in the evaluation.

4.1 Earthquake Ground Motion

This section of the safety evaluation contains a review of the earthquake data to assure that the vibratory ground motion, experienced at each of the facilities with equipment being used as a surrogate for similar equipment at Monticello, did indeed exceed the Monticello SSE. The ideal case, for this type of comparison, is to have actual recordings of the earthquake ground motion made at each of the facilities. The licensee has indicated that it relied on the ground motion estimates in the data base from actual instrument recordings at or near five facility sites and the SQUG Bounding Spectrum from the GIP-2 (Ref. 32) to verify the adequacy of the MSIV leakage path equipment.

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics such as the magnitude, focal mechanism, radiation pattern, stress drop, location of asperities and fault rupture history, and depth and orientation of the fault. It is also a function of the distance of the facility to the fault and the propagation properties of the rocks between them. The geology immediately under the facility site can also have a large effect on the amplitude and frequency content of the ground motion. Two of the more appropriate methods of estimating earthquake ground motion where there are no nearby recordings involve the use of (1) calibrated numerical modeling of the fault rupture and wave propagation process, and (2) empirical attenuation relationships obtained from the statistical analysis of large sets of earthquake data.

The licensee has stated that the Monticello condenser design is similar to, or bounded by, data for Moss Landing Units 6 and 7 which experienced the Loma Prieta 1989 earthquake, and Ormond Beach Units 1 and 2 which experienced the Point Mugu 1973 earthquake. They also indicate that the earthquake experience data that is directly being used for comparison to the Monticello piping is obtained from the following site-earthquake pairs.

El Centro Steam Plant - Imperial Valley 1979 earthquake.
Valley Steam Plant - San Fernando 1971 earthquake.

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 <u>Level 2 Internal Events PRA</u>

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. approximately 3% of CDF, the same percentage as for the base line.

For the bounding uprated power level at 112 percent, the licensee determined that the large early release frequency (LERF) was ~~estimated to be about 4.8E-7/Year. The "uprated" LERF represents an increase of about 7E-8/Year from the baseline LERF of about 4.1E-7/Year.~~ 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 the requested power increase by 6.3 percent to be acceptable as it meets the criteria of DG-1061. Since the CDF for the uprated power level increased slightly compared to the base line, the uprate LERF also increased slightly.

5.3 <u>Internal Fire, Seismic, and Other External Events PRA</u>

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 <u>Quality of PRA</u>

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 ~~1.64~~ 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.

Exhibit C

Evaluation of Proposed Change to the Monticello Technical Specifications

License Amendment Request Dated July 18, 2000
Alternate Shutdown System Operability Requirements

Pursuant to 10 CFR Part 50, Section 50.90, Northern States Power Company hereby proposes the following change to Appendix A to Facility Operating License DPR-22, "Technical Specifications" for Monticello Nuclear Generating Plant.

Background and Reasons for Change

In the event of a fire, 10CFR50 Appendix R requires that the ability to achieve and maintain safe shutdown be provided. Two areas of the Monticello plant, the Control Room and Cable Spreading Room, contain many cables from both trains of safe shutdown equipment. To comply with Appendix R, an alternate shutdown system (ASDS) was installed to provide the required protection for a fire in either of these areas. The ASDS provides control of the minimum necessary Division II systems from the ASDS panel, once the transfer switches are activated, to achieve safe shutdown. Among other systems, controls are provided for the 12 Residual Heat Removal Service Water (RHRSW) Pump to remove heat from the Torus. Section 10.3 of the MNGP USAR (Reference 1) provides additional information on the ASDS.

NRC Generic Letter (GL) 81-12, "Fire Protection Rule," (Reference 2) requested licensees to propose technical specifications to provide limiting conditions for operation (LCOs) of alternate shutdown equipment not already covered by existing technical specifications. Amendment 61 to the Monticello Operating License (References 3 and 4) established Technical Specification 3.13.H, "Alternate Shutdown System." Current Specification 3.13.H requires that:

> *The system controls on the ASDS panel shall be operable whenever that system/component is required to be operable.*

In the case of the RHRSW pumps, operability requirements are contained in Specification 3.5.C "Containment Spray/Cooling System." Amendment 102 to the Monticello Operating License (Reference 5) revised Specification 3.5.C to state that a train of Containment Spray/Cooling is considered operable when one of the two supporting RHRSW pumps (per train) is operable. Specification 3.5.C does not specify a particular pump required to be operable. Thus, there is no existing specification which specifies that 12 RHRSW Pump must be operable.

Proposed Change

A change to Appendix A of the Monticello Technical Specifications, Specification 3.13.H "Alternate Shutdown System," is proposed as follows. A change to Specification 3.13.H.1 is proposed to require 12 RHRSW Pump to be operable from the ASDS panel whenever there is irradiated fuel in the reactor vessel and reactor temperature is above 212°F. Similarly, a change to Specification 3.13.H.2 is proposed to address

inoperability of 12 RHRS Pump in the action statement. A markup of these changes is shown in Exhibit C to this letter. Exhibit D shows the final proposed version.

Safety Evaluation

MNGP Technical Specification 3.13.H, "Alternate Shutdown System," currently requires operability of ASDS system controls. In the context of specification 3.13.H, the scope of system controls is limited to those particular controls which are dedicated to the ASDS function and does not include the controlled component such as pump motors which have many diverse functions in addition to ASDS functions. This is appropriate presuming that the component/systems controlled by ASDS circuits have operability requirements under different technical specifications. Thus, only the ASDS controls would need to be addressed in Specification 3.13.H. Since the specific operability requirements for 12 RHRSW Pump were eliminated from Specification 3.5.C in Amendment 102, it is appropriate to establish operability requirements in 3.13.H.

For the components/systems having operability requirements defined in other technical specifications, only the ASDS controls are addressed in Specification 3.13.H.1. For 12 RHRSW Pump, the proposed requirement would encompass the remaining sub-components which allow the pump to be considered operable from the ASDS (e.g., motor, pump, etc.) . Hence the terminology "operable from the ASDS."

The conditions under which 12 RHRSW Pump is required to be operable as proposed in Specification 3.13.H.1 are consistent with the conditions for all of the RHRSW Pumps in Specification 3.5.C. Specifically, the proposed change and existing Specification 3.5.C require operability whenever there is irradiated fuel in the reactor vessel and reactor temperature is above 212°F.

The proposed change to Specification 3.13.H.2 provides an action statement for 12 RHRSW Pump that is consistent with the action statement for other components controlled from the ASDS panel. If 12 RHRSW Pump is not operable from the ASDS, the same actions are taken as other components controlled from the ASDS. Actions would also be taken to ensure that Specification 3.5.C is met with respect to the accident mitigation aspects of RHR SW pump operability.

No Significant Hazards Consideration:

An amendment to Appendix A of the Monticello operating to provide alternate shutdown system (ASDS) operability requirements for a specific Residual Heat Removal Service Water (RHRSW) Pump. The proposed amendment has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10 CFR Part 50, section 50.91 using standards provided in section 50.92. This analysis is provided below:

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The 12 RHRSW Pump is not an accident (fire) initiator. During a fire in the Control Room or Cable Spreading Room, the ASDS panel provides alternate shutdown capability. The proposed amendment provides operability requirements to ensure 12

RHRSW Pump is available when alternate shutdown is required so that safe shutdown can be achieved and maintained in accordance with existing procedures. The proposed operability requirements are consistent with previous ASDS requirements for 12 RHRSW Pump and other equipment required for alternate shutdown. Dose to the public and the Control Room operators are not affected by the proposed change.

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not significantly increase the probability or the consequences of an accident previously evaluated.

<u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

The proposed amendment is within current Technical Specification requirements for other equipment required for alternate shutdown and ensures that 12 RHRSW Pump will be available for alternate shutdown when required. The allowed ASDS outage time for 12 RHR SW Pump is consistent with that allowed for other alternate shutdown equipment. The proposed amendment maintains margins of safety. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

Environmental Assessment

Northern States Power has evaluated the proposed change and determined that:

1. The change does not involve a significant hazards consideration.
2. The change does not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite, or
3. The change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(b), and an environmental assessment of the proposed change is not required.

References

1. MNGP USAR, Section 10.3, "Plant Service Systems," Revision 17.
2. NRC Generic Letter (GL) 81-12, "Fire Protection Rule," dated February 20, 1981.
3. NRC letter to NSP, "Amendment No. 61 to Facility Operating License No. DPR-22: (TAC No. 61302)," dated March 29, 1989.
4. NRC letter to NSP, "Errata for Amendment No. 61 to Facility Operating License No. DPR-22: (TAC No. 61320)," dated July 14, 1989.
5. NRC letter to NSP, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Power Uprate Program (TAC No. M96238)," dated September 16, 1998.

Exhibit D

Current Monticello Technical Specification Pages Marked Up
With Proposed Change

License Amendment Request Dated
July 18, 2000

Exhibit D consists of current Technical Specification page marked up with the proposed change. The page included in this exhibit is as listed below:

<u>Page</u>

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12 RHR Service Water Pump shall be operable from the ASDS panel whenever there is irradiated fuel in the vessel and reactor water temperature is greater than 212°F.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

H. Alternate Shutdown System

1. The system controls on the ASDS panel shall be operable whenever that system/component is required to be operable.

2. If system controls required to be operable by ~~Specification~~ 3.13.H.1 or 12 RHR Service Water Pump are made or found inoperable, restore ~~the inoperable system control to operable~~ operability within 7 days, or perform one of the following;

   a. Provide equivalent shutdown capability and within 60 days restore the inoperable system controls to operable; or

   b. Establish a continuous fire watch in the cable spreading room and the back-panel area of the control room and within 60 days restore the inoperable system controls to operable; or

   c. Verify the operability of the fire detectors in the cable spreading room and the back-panel area of the control room and establish a hourly fire watch patrol and within 60 days restore the inoperable system controls to operable; or

   d. Place the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be operable within 24 hours.

3. The alternate shutdown system panel master transfer switch shall be locked in the normal position except when in use, being tested or being maintained.

4.0 SURVEILLANCE REQUIREMENTS

H. Alternate Shutdown System

1. Switches on the alternate shutdown system panel shall be functionally tested once per operating cycle.

2. The alternate shutdown system panel master transfer switch shall be verified to alarm in the control room when unlocked once per operating cycle.

Exhibit E

Revised Monticello Technical Specification Pages

License Amendment Request Dated
July 18, 2000

Exhibit E consists of revised Technical Specification pages that incorporate the proposed change. The pages included in this exhibit are as listed below:

<u>Pages</u>

227c

3.0 LIMITING CONDITIONS FOR OPERATION

H. Alternate Shutdown System

1. The system controls on the ASDS panel shall be operable whenever that system/component is required to be operable. 12 RHR Service Water Pump shall be operable from the ASDS panel whenever there is irradiated fuel in the vessel and reactor water temperature is greater than 212°F.

2. If system controls or 12 RHR Service Water Pump required to be operable by Specification 3.13.H.1 are made or found inoperable, restore operability within 7 days, or perform one of the following;

   a. Provide equivalent shutdown capability and within 60 days restore the inoperable system controls to operable; or

   b. Establish a continuous fire watch in the cable spreading room and the back-panel area of the control room and within 60 days restore the inoperable system controls to operable; or

   c. Verify the operability of the fire detectors in the cable spreading room and the back-panel area of the control room and establish a hourly fire watch patrol and within 60 days restore the inoperable system controls to operable; or

   d. Place the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be operable within 24 hours.

3. The alternate shutdown system panel master transfer switch shall be locked in the normal position except when in use, being tested or being maintained.

4.0 SURVEILLANCE REQUIREMENTS

H. Alternate Shutdown System

1. Switches on the alternate shutdown system panel shall be functionally tested once per operating cycle.

2. The alternate shutdown system panel master transfer switch shall be verified to alarm in the control room when unlocked once per operating cycle.