NSP

Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

March 26, 1998

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

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

NSP Response to Supplemental Request For Additional Information
Concerning the Monticello Nuclear Generating Plant
Power Rerate Program (TAC No. M96238)

Ref. 1 Letter from NRC to R.O. Anderson, NSP, "Monticello Nuclear Generating Plant - Request for Additional Information on License Amendment Request Entitled `Supporting the Monticello Nuclear Generating Plant (MNGP) Power Rerate Program' (TAC No. M96238)," March 6, 1998

Ref. 2 Letter from NRC to R.O. Anderson, NSP, "Monticello Nuclear Generating Plant - Request for Additional Information on License Amendment Request Entitled `Supporting the Monticello Nuclear Generating Plant (MNGP) Power Rerate Program' (TAC No. M96238)," February 11, 1998

By letter dated March 6, 1998 (Ref. 1), the NRC staff provided a Request for Additional Information (RAI) to complete its review of NSP's license amendment request for the Monticello Nuclear Generating Plant (MNGP) Power Rerate Program. This request supplements a previous rerate RAI dated February 11, 1998 (Ref. 2). NSP's response to Ref. 1 is provided in Attachment 3 to this letter. NSP's response to question 6 of Ref. 2 is provided as Attachment 4. Two calculations provided as part of Attachment 4 are considered to be NSP proprietary. An affidavit to withhold this information from public disclosure is provided as Attachment 2. Conference calls were held between the staff and NSP on March 10 and March 17, 1998. Some additional rerate questions were asked by the staff. The responses to these questions are provided in Attachment 5.

Certain responses include calculations. These calculations are current as of the date of this letter. Future revisions to these calculations, if any, will be available onsite for staff review.

Please contact Joel Beres, Monticello Licensing, at (612) 295-1436 if additional information is required.

M F Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

A001

change PDR   Ltr 1   Enc 1 w/o PROP.

c: Regional Administrator - III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
State of Minnesota, Attn: Kris Sanda
J. Silberg, Esq.

Attachments

Attachment 1	Affidavit to the US Nuclear Regulatory Commission
Attachment 2	NSP Proprietary Information Affidavit
Attachment 3	NSP Response to Staff Power Rerate RAI dated March 6, 1998
Attachment 4	NSP Response to Question 6 of RAI dated February 11, 1998
Attachment 5	Rerate Questions from Conference Calls Between Staff and NSP on March 10, and March 17, 1998
Attachment 6	Gothic Verification Calculation
Attachment 7	Moisture Separator System Engineering Evaluation
Attachment 8	Power Rerate Work Scope

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT DOCKET NO. 50-263

Response to March 6, 1998 Request for Additional Information (RAI) on Monticello Power Rerate License Amendment (TAC No. M96238)

Northern States Power Company, a Minnesota corporation, by letter dated March 26, 1998 provides its response for the Monticello Nuclear Generating Plant to a US Nuclear Regulatory Commission (NRC) letter dated March 6, 1998, with the subject "Monticello Nuclear Generating Plant - Request for Additional Information on License Amendment Request Entitled 'Supporting the Monticello Nuclear Generating Plant (MNGP) Power Rerate Program' (TAC No. M96238)." This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By M F Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

On this 26th day of March 1998 before me a notary public in and for said County, personally appeared Michael F. Hammer, 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, and that to the best of his knowledge, information, and belief the statements made in it are true.

Samuel I. Shirey

Samuel I. Shirey
Notary Public - Minnesota
Sherburne County

SAMUEL I. SHIREY
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2000

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT DOCKET NO. 50-263

Request to Withhold Proprietary Information
from Public Disclosure

Northern States Power Company, a Minnesota corporation, hereby requests that two calculations, CA 97-176 Rev. 2 and CA 98-105 Rev. 0, provided to the NRC within attachment 4 to the letter titled, "NSP Response to Supplemental Request For Additional Information Concerning the Monticello Nuclear Generating Plant Power Rerate Program (TAC No. M96238)," be withheld from public disclosure due to their proprietary nature. The details of this request are provided in the following affidavit:

AFFIDAVIT

I, Michael F. Hammer, being duly sworn, depose and state as follows:

(1) I am Plant Manager, Monticello Nuclear Power Plant, Northern States Power Company ("NSP") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld consists of two calculations CA 97-176 Rev. 2, "Equivalent Temperature Evaluation for EQ Temperature in Containment," and CA 98-105 Rev. 0, "Qualified Life Evaluation for Containment EQ Components." The drawing provides key design information and technical details concerning plans for power rerate equipment qualification. The calculations have the words "NSP Proprietary Information" on each page.

(3) In making this application for withholding of proprietary information of which it is the owner, NSP relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR9.17(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission 975F2d871 (DC Cir. 1992), Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by NSP's competitors without license from NSP constitutes a competitive economic advantage over other companies;

b. information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future NSP funded development plans and programs of potential commercial value to NSP;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. through (4)c., above.

(5) The information sought to be withheld was submitted to NRC in confidence. The information is of a sort customarily held in confidence by NSP, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by NSP, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

(6) Public disclosure of the information sought to be withheld is likely to cause harm to NSP's competitive position and foreclose or reduce the availability of profit-making opportunities. The research, development, engineering, and analytical costs comprise a substantial investment of time and money by NSP. The value of this information to NSP would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive NSP of the opportunity to exercise its competitive advantage to seek an adequate return on its investment in developing this repair method.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By M F Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

On this 26th day of March 1998 before me a notary public in and for said County, personally appeared Michael F. Hammer, 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, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true.

Samuel I. Shirey

Samuel I. Shirey
Notary Public - Minnesota
Sherburne County

SAMUEL I. SHIREY
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2000

Attachment 3

NSP Response to Staff Power Rerate RAI dated March 6, 1998

1. *Section 4.5.4 states that the increase in the radioactive source term caused by operating at higher power levels results in a potential increase in the control room operator dose. Explain what the "potential" increase is, and how it corresponds to the existing allowable dose to the control room operators.*

NSP Response

The use of "potential" within the context of Section 4.5.4 was to signify that the possibility of a dose increase existed, and that given this possibility additional analysis is required to quantify the effects of rerate. It was not intended to provide a conclusion on the extent of the change. The nature and extent of the change and the associated analyses and conclusions are provided below. The increases in dose attributable to operating at higher power levels varies from no change for the main steam line break accident to a change proportional to the increase in reactor thermal power for the loss of coolant accident (LOCA) and other design basis accidents. Refer to Tables 8.1-1 through 8.1-4 of GENE-B3100594-1 included in NSP's submittal dated March 6, 1998. The potential increase in the control room dose referred to in Section 4.5.4 is primarily due to the use of up-to-date Nuclear Regulatory Commission (NRC) approved methodology in the treatment of secondary containment bypass leakage.

Background

The original LOCA radiological evaluations are contained in the Monticello Final Safety Analysis Report (Reference 1.1) and in the Atomic Energy Commission Staff Safety Evaluation Report issued for the initial licensing of Monticello (Reference 1.2). These analyses assumed no secondary containment bypass leakage. All accident-related radioactive material was assumed to leak from primary containment into secondary containment at the maximum allowable rate. This leakage was then processed by the Standby Gas Treatment (SBGT) System and released from the top of the 100-meter offgas stack.

Secondary containment bypass leakage is not filtered by the SBGT System and is released at ground level without the benefit of dilution and dispersion provided by the offgas stack. Main steam isolation valve (MSIV) leakage is the major contributor to secondary containment bypass leakage. Since bypass leakage was not considered in the original plant design, a filtered ventilation treatment system was not required for the control room for the initial licensing of MNGP.

Control Room Habitability Improvements

Following the accident at Three Mile Island (TMI) Unit 2 in 1979, the NRC required all nuclear power plants to review the post-accident habitability of control rooms (Reference 1.3). The NRC required that potential bypass leakage sources be taken into account in evaluating the radiological doses that control room operators would receive following a LOCA. The control room habitability requirement, which was delineated as TMI Action Plan Item III.D.3, was satisfied at Monticello by installation of the Emergency Filtration Train (EFT) System. The system was designed by the Monticello architect/engineer - Bechtel. Bechtel also performed the original control room habitability radiological analysis that included the effects of bypass leakage.

The Bechtel analysis included an assumption that 25% of the primary containment leakage bypasses secondary containment. This bypass leakage was assumed to be dominated by main steam isolation valve (MSIV) leakage that enters the main condenser. Twenty-five percent is large enough to account for MSIV leakage at the Technical Specification limit with some margin for other potential bypass leakage paths. This analysis was submitted to the NRC (Reference 1.4) and found acceptable (Reference 1.5).

In 1984 Bechtel was requested to revise the control room LOCA radiological analysis to permit the use of a lower Technical Specification limit for EFT charcoal adsorber efficiency. This analysis assumed iodine removal rates (λ_n) of 4.22 hr^{-1} and 7.33 hr^{-1} for the main steam line and condenser, respectively. Taking credit for iodine removal mechanisms in the main steam line and condenser significantly reduces calculated thyroid doses. This revised analysis was submitted to the NRC on August 17, 1984 (Reference 1.6) and later approved (Reference 1.7). The NRC Safety Evaluation Report that accompanied Reference 1.7 contained an independent control room LOCA dose assessment prepared by the NRC Staff. The NRC assessment differed from the 1984 Bechtel analysis in that no secondary containment bypass leakage was assumed. All leakage was assumed to be processed by the SBGT System and released from the stack.

Bechtel updated the Monticello design basis radiological analyses in 1992 at NSP's request. The new calculations assumed 25% secondary containment bypass, as did the earlier calculations, and also applied iodine plateout factors in the main steam lines and condenser.

Power Rerate Program Dose Calculations

In early 1996, GE updated the design basis accident radiological analyses for the Monticello power rerate program. These analyses were performed at 1880 MWt. Current NRC guidance and the latest GE models were used for these calculations. The initial GE offsite and control room LOCA dose results were higher than would be expected simply due to the higher reactor power level. Review of these calculations indicated that GE did not use the iodine plateout factors assumed for the main steam line and condenser in the Bechtel models.

Informal discussions with the NRC indicated that iodine plateout factors, while applicable to the control rod drop accident evaluation (Reference 1.8), should not be automatically assumed for LOCA dose analyses. The staff indicated that iodine removal mechanisms should be modeled on a plant-specific basis if credit is to be taken for iodine removal.

GE reanalyzed the LOCA accident using the MSIV bypass leakage iodine removal model developed for the BWR Owners Group (Reference 1.9). This model has been approved by the staff for certain plants. The new calculations were included with the power rerate license amendment application (Reference 1.10). Use of the

GE bypass leakage model resulted in an increase in calculated control room thyroid doses over those calculated previously using other methodology. Doses, however, remain well below the guidelines of 10 CFR Part 50, Appendix A, GDC 19.

References

1.1. Monticello Final Safety Analysis Report (FSAR), including amendments 9 through amendment 28 dated 7/21/1970.

1.2. Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, In the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970 (Safety Evaluation Report for Monticello Provisional Operating License)

1.3. NRC Letter dated May 7, 1980, "Five Additional TMI-2 Related Requirements to Operating Reactors," TMI Action Plan Item III.D.3.

1.4. NSP Letter dated January 30, 1981, from L. O. Mayer to Director of NRR, USNRC, "Information Related to Post TMI Requirements."

1.5. NRC Letter dated April 6, 1983, "NUREG-0737, Item III.D.3.4, Control Room Habitability," (Safety Evaluation Report)

1.6. NSP License Amendment Request dated April 3, 1984, including supplemental information submitted to NRC on August 17, 1984, (contains revised control room habitability analysis), August 30, 1985, November 27, 1985, February 19, 1987, June 6, 1988, and July 5, 1988).

1.7. NRC letter dated May 30, 1989, Amendment No. 65 to Facility Operating License No. DPR-22 (TAC No. 56977).

1.8 NRC Standard Review Plan Section 15.4.9, "Spectrum of Rod Drop Accidents (BWR) Appendix A - Radiological Consequences of a Rod Drop Accident

1.9 GE Nuclear Energy Report NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," September, 1993

1.10 NSP License Amendment Request dated July 26, 1996

2. *Section 4.5.4 states that the analysis inputs reflect changes to the emergency filtration train (EFT) system performance to compensate for the analysis method changes. Describe the "changes" to the EFT.*

NSP Response

The changes included an increase in the filter efficiency that is described in NSP's response to Question 43 of our submittal dated September 5, 1997.

In addition, blanking plates have been installed on the air intake ducts. The original design of the EFT system included two outside air supply ducts equipped with automatic isolation dampers that closed on an EFT actuation. Leakage through these outside dampers represents a potential source of unfiltered air to the control room following an accident. The original design of this equipment made it difficult to maintain the leak tightness necessary to prevent the entry of significant amounts of unfiltered outside air. Blanking plates were installed in the existing ductwork to eliminate this potential leakage source. NSP is evaluating alternate solutions, such as bubble tight dampers, for future installation that would permit air intake while meeting leakage requirements.

Use of the GE bypass leakage model requires seismic verification of a leakage collection path from the outboard MSIVs to the main condenser. Walkdowns and analyses of the main condenser and main steam drain lines were completed, and modifications were made to correct discrepancies identified during the seismic evaluation. Some analytical seismic evaluations are not complete. These evaluations were previously discussed with the staff. See Exhibit H of NSP's submittal dated December 4, 1997 for the commitment regarding seismic qualification of the leakage collection path.

3. *Section 6.3 states that the spent fuel pool (SFP) heat loads will slightly increase resulting from plant operations at the proposed power level. Provide the following information:*

a. *Provide/compare the heat loads and corresponding peak calculated SFP temperatures (for plant operations at current power level and at proposed rerate power level) during planned refueling and unplanned full core offload. Single failure of the SFP cooling system does not need to be assumed for the unplanned full core offload.*

NSP Response

Table 6-2 of Exhibit E of NSP's submittal dated December 4, 1997 provides heat loads and corresponding peak calculated fuel pool temperatures for current plant conditions and for rerate conditions during planned refueling (Normal Heat Load From Refueling and Bulk Fuel Pool Temperature with 90 °F River Temperature and Normal Heat Load From Refueling).

For an unplanned full core offload, often referred to as the "emergency heat load," the following heat load values apply.

Present (1670 MWt):	17.8×10^6 BTU/hr
Rerate (1775 MWt):	18.8×10^6 BTU/hr
106% of Rerate (1880 MWt):	20.0×10^6 BTU/hr

The above heat loads were determined by calculation (CA 96-127, Addendum 1). The loads include an additive factor to account for decay heat uncertainty. This calculation was also used as the source of the normal refueling heat load data.

The analysis of the emergency heat load condition includes an assumption that fuel pool cooling is provided by the RHR system, rather than the Fuel Pool Cooling system. In this situation, the analysis assumes that the temperature of the fuel pool is 140 °F and then calculates the RHR flow required to maintain this temperature assuming the river water (the ultimate heat sink) is at 90 °F. The RHR flows required in this situation are presented in Table 6-2 as RHR Flow Required to Maintain 140 °F for [1670, 1775, 1880] MWt. These flows are all within the capability of the RHR system.

b. *Is full core offload the general practice for planned refuelings?*

NSP Response

The standard Monticello practice is to use an in-core shuffle to refuel the reactor. NSP has performed full core offloads on a few occasions in response to specific outage conditions such as recirculation piping replacement and decontamination. Full core offloads may be performed in the future.

c. *How many SFP cooling system trains will be available/operable prior to a planned refueling outage or an unplanned full core offload?*

NSP Response

The evaluation results shown in Table 6-2 of Exhibit E for the fuel pool are based on the use of one fuel pool cooling pump with flow through both heat exchangers. This is applicable to all normal heat load cases shown. Use of both pumps would decrease the temperatures shown. In accordance with MNGP procedures, both divisions of the Fuel Pool Cooling System are required to be available or at least one alternative method of fuel pool cooling must be available.

4. *In the unlikely event that there Is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling) and the boil-off rate (based on the heat load for the unplanned full core offload scenario). Also, discuss sources and capacity of makeup water and the methods/systems (indicating system seismic design category) used to provide the makeup water.*

NSP Response

The following table presents time-to-boil and boiloff rates for the emergency heat loads, assuming an initial fuel pool temperature of 120 °F which coincides with the high temperature alarm.

Power (MWt)	Decay Heat MBTU/hr	Time to Boil (hr)	Boiloff Rate (gpm)
1670	17.8	11.5	38.2
1775	18.8	10.9	40.4
1880	20.0	10.3	43.0

Fuel pool makeup sources are listed below.

RHR Service Water via RHR System (capacity >1744 gpm, seismic class 1)
RHR System (capacity > 1744 GPM, seismic class 1)
Filter/Demin Backwash Connection (capacity >100 gpm, seismic class 2)
Fire Hose Station* (capacity >100 gpm, seismic class 2)
Condensate Service Water Hose Station (capacity approximately 40 gpm per hose station, seismic class 2)
Demin Water Hose Station (capacity approximately 20 gpm per hose station, seismic class 2)

* The water supply is from the electric fire pumps or from the diesel driven fire pump if AC power is lost.

5. *What is the maximum temperature that the fuel racks are designed for, and what temperature would the racks actually experience with power rerate?*

NSP Response

Operation at power rerate does not involve an increase in the bulk temperature that the racks would be exposed to above that for the present power level. The fuel racks were designed and supplied by GE. The structural design of the fuel racks at MNGP is such that the racks are free to expand at elevated temperatures.

The maximum temperature that the racks would experience during normal rerate conditions is 125 °F. This is the same as the current normal maximum temperature because the fuel pool cooling system is capable of removing more heat than is generated by the spent fuel under normal conditions, for both current and rerate power levels. The maximum temperature that the racks would experience during emergency rerate conditions is 140 °F. This is the same as the current emergency maximum temperature because emergency RHR flow to the fuel pool is adjusted as necessary to maintain the fuel pool at or below this temperature, for both current and rerate power levels.

6. *Section 6.4.1.1.2 states that the heat loads for the shutdown cooling, suppression pool cooling, and emergency fuel pool cooling modes of operation are increased by power rerate. Provide the analysis to support this statement and include the maximum spent fuel pool temperature that results from power rerate.*

NSP Response

I. Shutdown Cooling Mode

Shutdown cooling capacity is discussed in USAR Section 10.2.4. USAR Section 10.2.4 includes the following statements.

> Once the reactor water has been cooled to about 281 oF by the main condenser, the reactor shutdown cooling system must be capable of cooling the reactor water down to 125 °F within 24 hours after reactor shutdown and maintaining it at this temperature by removing fission-product decay heat absorbed by the reactor water. To achieve this objective, the RHR heat exchanger was designed as presented in Table 6.2-2.
>
> During a reactor primary system shutdown and cooldown when shutdown cooling subsystem is initially placed in operation, decay heat levels can be high and operation of both RHR heat exchangers may be required to remove the heat rapidly to permit shutdown operation as soon as practical.

USAR Table 6.2-2 describes a design load case that defines heat exchanger capacity. This was typical of several different cases that were provided by the original equipment manufacturer for various system flow rates and process temperatures. Use of equation A-2 of NEDO-20533-1, General Electric Mar^k III Pressure Suppression Containment System Analytical Model Supplement 1, September 1975 allows the development of a K value to evaluate heat exchanger performance at other process fluid temperature values. The K value is defined in units of BTU/sec-°F. For the shutdown cooling evaluation, a K value of 192.3 BTU/sec-°F was calculated. This K value is based on the suppression pool cooling mode, and it is conservative for the higher temperatures in the shutdown cooling evaluation.

The heat load associated with the shutdown cooling evaluation is equivalent to the removal of decay heat plus the latent heat in the primary system. Decay heat loads are proportional to initial reactor thermal power. The 1775 MWt decay heat is 6.3% greater than 1670 MWt while the latent heat is identical, since the reactor operating temperature and pressure is identical for both cases. Maximum capacity of the RHR heat exchanger can vary depending on the process conditions associated with the scenario such as initial river temperature and reactor coolant temperature at the time of shutdown cooling initiation.

An interlock prevents shutdown cooling initiation until reactor dome pressure is about 35 psig (at 281 °F), under the current MNGP Technical Specifications.

This will increase under the proposed rerate changes to the Technical Specifications to 75 psig (at 320 °F). The maximum capacity of the RHR heat exchanger for each case is defined by the following equation.

$$Q = K * (T_h - T_C)$$

Where
Q is heat transfer rate in BTU/sec.
K is the heat exchanger capacity for the flow rates used or 192.3 BTU/sec-°F.
T_h is the fluid temperature of the reactor water entering the heat exchanger, 281 °F with current Technical Specifications and 320 °F after rerate implementation.
T_C is the fluid temperature of RHRSW, or the maximum ultimate heat sink (river) temperature of 90 °F.

At 1670 MWt with a reactor temperature of 281 °F, Q = 36,729 BTU/sec. At 1775 MWt with a reactor temperature of 320 °F, Q = 44,229 BTU/sec.

It should be recognized that these heat exchanger capacities would not normally be used since they would result in exceeding the 100 °F/hour cooldown rate allowed for the primary system. At these temperatures heat exchanger flow would be throttled to reduce heat removal rate to stay within allowable cooldown rates.

At 1880 MWt it will take less than 24 hours to cool down the reactor to meet the requirements of USAR Section 10.2.4. This takes approximately 22 hours at a power level of 1775 MWt and 20 hours at a power level of 1670 MWt with all other inputs being identical.

II. Suppression Pool Cooling Mode

Suppression pool cooling heat loads are defined in a similar fashion. The limiting case for suppression pool temperature response is a LOCA with loss-of-offsite power and a single failure of an emergency diesel generator. This results in a load case where only one RHR pump and one RHRSW pump are assumed to be operating in the suppression pool cooling mode. The appropriate K value to use for RHR heat exchanger capacity has been calculated to be 143.1 BTU/sec- °F for this case.

Per Table 4-1 of Exhibit E of NSP's license amendment dated December 4, 1997, the peak suppression pool temperatures for a power level of 1670 MWt are 184 °F. This will increase to a temperature of 194 °F for operation at 1775 MWt with conservative margin provided for decay heat uncertainty.

RHR heat exchanger capacity in the suppression pool cooling mode is determined by the following equation.

$$Q = K * (T_h - T_C)$$

Where
Q is heat transfer rate in BTU/sec.
K is the heat exchanger capacity for the flow rates used or 143.1 BTU/sec-°F.
T_h is the fluid temperature of the suppression pool water entering the heat exchanger.
T_C is the fluid temperature of RHRSW, or the maximum ultimate heat sink (river) temperature of 90 °F.

At 1670 MWt with a suppression pool temperature of 184 °F, Q = 13,451 BTU/sec.
At 1775 MWt with a suppression pool temperature of 194 °F, Q = 14,882 BTU/sec.

This case defines the limiting containment response evaluated for the DBA LOCA. In order to verify that actual heat exchanger performance meets the requirements assumed for this case, a periodic test is performed to verify that the heat transfer rates assumed in the containment analysis for the suppression pool cooling mode can be met based on actual heat exchanger performance.

III. Emergency Fuel Pool Cooling Mode

As shown in Table 6-2 of Exhibit E of NSP's December 4, 1997 submittal, the maximum spent fuel pool temperature of 140 °F is assumed in the evaluation of an unplanned full core offload. See the response to question 3.a above. The RHR flows shown are also provided in Table 6-2 of Exhibit E. These flows are required to remove the heat loads shown and are based on a fuel pool temperature of 140 °F and a maximum ultimate heat sink (river) temperature of 90 °F. The RHR flows shown in Table 6-2 of Exhibit E were calculated using an NSP-developed computer program, HXPERF. The validity of the calculations performed with HXPERF was demonstrated in Monticello Calculation/Analysis 94-020 and 94-020 Rev. 1, "RHR/RHR Service Water Heat Exchanger Performance."

7. *In Section 6.6, for HVAC [heating, ventilation, and air conditioning] that serves areas with condensate and feedwater lines or with lines that contain suppression pool water, provide the evaluation for the effect of the power rerate on the HVAC systems.*

<u>NSP Response</u>

The evaluation of the effect of power rerate on HVAC Systems is documented in the task report titled, Heating, Ventilation and Air Conditioning System Engineering Evaluation, Task 16, Revision 1. This evaluation is summarized below.

Since operation at rerate conditions does not reduce any process flow temperatures, heating is not impacted. The ability to maintain minimum temperatures in any area of the plant during normal operation is not adversely impacted. Since operation at rerate conditions does not change the flow path or differential pressure requirements for which the system is designed, the ability of the system to minimize the spread of contamination is not adversely impacted. The impact of rerate on the cooling requirements of the HVAC systems is discussed below.

Drywell Ventilation
The feedwater system process temperature will increase 6 °F in the drywell. No other system is exposed to a significant temperature increase. Total drywell heat loads increase 0.1%. This is an insignificant change that is well within system capabilities based on past operating experience. The maximum drywell temperature is used to establish the initial containment non-condensable gas mass assumed in ECCS pump NPSH calculations and in the calculation of the maximum containment response to a DBA LOCA. In order to insure adequate ECCS pump NPSH, an alarm has been established to warn the plant operators if drywell weighted average temperatures reach the value of 135 °F used in the NPSH analysis as an initial condition for the event.

Steam Chase
The original design temperatures assumed for steam and feedwater flow in this area bound the values that will result from operation under rerate conditions. Since steam line pressures and temperatures will not change with rerate, and feedwater temperatures only increase by 6°F, the change in heat load to this area under rerate conditions is insignificant. Steam chase temperatures are currently recorded daily with corrective action required if temperatures reach 130°F. The action temperature is typically reached only on hot summer days. There is sufficient excess capacity in the coolers to maintain steam chase temperatures within equipment operating temperatures during power rerate operation.

ECCS Pump Rooms
The room heatup evaluation is summarized in Table 6-3 of Exhibit E of NSP's rerate submittal dated December 4, 1997. NSP calculation CA-97-157, which documented this evaluation for the 1880 MWt power case, was provided to the NRC by submittal dated July 16, 1997, "Request for Information Regarding MNGP License Amendment Dated June 19, 1997 (TAC No. 97781)."

Suppression Pool Area
An evaluation of reactor building ambient temperature conditions during the period after a LOCA was not part of the original design or licensing requirements for the site. Further evaluation of the impact of post-LOCA temperatures on the building was recognized as an enhancement of the

assessment of the site to be able to withstand a postulated accident. Conservative assumptions have been used in evaluations to date based on engineering judgment. Verification of the conservatism of these assumptions is covered by calculation described in License Commitment No. 6 in Exhibit H of NSP's rerate submittal dated December 4, 1997.

Condensate and Feedwater Areas
While condensate and feedwater line temperatures do increase, the percentage of increase is 2% or less. The evaluations of affected areas show that the maximum design temperatures assumed in the original plant design will not be exceeded at rerate conditions. This conclusion will be confirmed by monitoring temperatures at rerate conditions. The monitoring program will include areas around the condensate pumps, reactor feedwater pumps, turbine deck and the main condenser room.

8. *For Section 7.0. provide the analyses for the following:*

a. *To assure that the turbines can pass the higher steam flow rates with adequate design and pressure control.*

NSP Response

The turbine at Monticello was replaced in part because of anticipated operation at rerate conditions. The high pressure turbine steam flow path was replaced based on the potential for efficiency improvements that were possible due to improvements in design that have occurred since original plant startup in 1971. The new turbine steam flow path was sized for operation at rerate steam flow rates. This work was completed in the 1996 refueling outage.

Startup testing after the 1996 refueling outage showed that the new steam flow path was undersized. This occurred due to errors in the vendor's design and manufacturing process. The errors in the sizing design of the nuclear steam turbine flow paths have been corrected. The analysis work was performed by GE's Turbine Division. Modifications are required to the high pressure turbine steam flow path that will increase unit capacity to allow operation under rerate steam flow rates. The changes will consist of replacing the first 3 rows of stationary diaphragms and the first row of buckets on the high pressure turbine rotor. These changes will be done during the 1998 refueling outage. The new turbine flow path was designed to operate under rerate conditions at the original turbine inlet design pressure of 965 psia with a nominal 3% margin on control valve capacity.

b. *To evaluate the performance of the moisture-separator reheater systems under higher operating conditions.*

NSP Response

The evaluation of the moisture separators is provided in the Moisture Separator System Engineering Evaluation, Task 18.5, report that is provided as Attachment

7. All actions associated with the report are complete except for the monitoring of drain system stability which is part of the rerate startup testing program. The original Appendix A and B for this report are replaced in this submittal by the associated NSP calculations CA-96-009 and CA-96-013 respectively.

c. *To ensure that the main generator and its auxiliaries are capable of producing higher electrical output while still meeting design requirements (Include an evaluation of the effect of the increased generator loads on the TBCCW (turbine building component cooling water system).*

NSP Response

The turbine and the generator were evaluated by the original equipment manufacturer, GE. Modifications to the turbine are discussed in the response to Question 8.a above. Modifications to the generator auxiliaries are discussed briefly in Exhibit E, Section 7.1 of NSP's rerate submittal dated December 4, 1997. The required changes included a modification to increase the capacity of the stator water cooling system and a modification to increase the capacity of the isophase bus cooling system. These modifications were performed during the 1996 refueling outage.

With the completion of these modifications, the design rating of the generator was increased from 632,000 kVA with a power factor of 0.90 to 664,418 kVA at a power factor of 0.95. This is based on an expected gross output of 631,197 kW after implementation of rerate. Additional generating capacity from improved turbine efficiency was expected to result from the turbine replacement discussed in Question 8.a above. Results of turbine performance testing to date, however, show that it is likely that peak gross generation will be in the range of 620,000-625,000 kW which results in additional margin to the generator rating. The new power factor rating was factored into grid stability evaluations and has been reviewed by the Mid-Continent Area Power Pool (MAPP).

Cooling for turbine building loads at MNGP is provided by the normal service water system. The service water system is expected to have increased cooling water flow rate requirements as a result of operation at rerate conditions. Increased loads are principally due to the additional generator cooling requirements. These loads are discussed below.

Stator Cooling
The service water flow rates needed to remove design heat loads from the stator are expected to increase by about 10% at rerate conditions. A test was conducted on July 20, 1995 that showed that normal stator water cooling flow in summer operation is 540 gpm at 1670 MWt. Flow to the unit was increased to approximately 900 gpm during the test. The expected flow requirement under rerate conditions is 600 gpm; thus substantial margin exists to required flow rates under rerate conditions.

Isophase Bus Cooling
A test was conducted on July 20, 1995 that showed that normal isophase bus cooling flow in summer operation is 68 gpm at 1670 MWt. The required flow rate for the original isophase bus cooler was 33.5 gpm at 632,000 kVA. The new isophase bus cooling unit installed in the 1996 refueling outage requires 45.5 gpm to cool the isophase bus at 664,418 kVA. The operators normally operate this system at valve wide open for convenience. The 68 gpm seen in the July 20, 1995 test is typical of summer operation and shows adequate margin for rerate operating conditions.

Generator Hydrogen Cooling
Evaluation of the service water supply to this system shows that the current flow requirements are adequate to cool the unit under rerate operating conditions.

The service water system adequacy at rerate operating conditions will be confirmed by monitoring the system and the system loads during the rerate startup test program.

d. *For the effect of power rerate on the feedwater pumps, feedwater heater drains, and the feedwater control valves.*

NSP Response

The feedwater pumps have a rated flow of 7,820 gpm and rated head of 2,750 ft at the pump's design point. For 1775 MWt, the total feedwater flow obtained from the reactor heat balance is 7,235 Klb/hr. This corresponds to 7,897 gpm per reactor feedwater pump. From the feedwater pump performance curves, the reactor feedwater pumps will provide a head of 2,750 ft at a flow of 7,897 gpm. Therefore, for 1775 MWt rerate conditions, the pump will be operating very close to its design rated flow and head, which corresponds to its most efficient operating point. It should be noted that the pump is capable of significantly higher flows.

The heater drain valves were evaluated by comparing the required C_v as calculated for rerate conditions to the design C_v at valve wide open conditions. The result of this calculation is shown in Table 8-1. Considerable flow margin will exist under rerate operating conditions, therefore the valves are capable of handling increased flows without modification or replacement.

Table 8-1

Heater	Valve I.D.	Required C_v	Design C_v	Approx. % Open
15B	CV-1058	52	105	80-90
14B	CV-1056	246	510	60-70
13B	CV-1054	302	855	55-65
12B	CV-1052	741	1650	60-70

The A and B strings of feedwater heaters can be considered symmetrical such that the B string is representative of both. The above table shows that the B loop has adequate drain valve capacity.

The effect of power rerate on the feedwater control, or regulating, valves is described in the response to Question 10 below.

9. *Section 7.4 states that the feedwater system design pressure and temperature requirements have been evaluated for 1775 MWt [megawatts thermal] and are adequate for power rerate. Explain what is meant by "adequate."*

NSP Response

The use of "adequate" in Section 7.4 was intended to mean that operation would be within code allowables.

The feedwater system piping and supports have been evaluated for rerate design pressures and temperatures. This equipment meets applicable code requirements specified in Section 12 of the USAR. See responses to Question 10 of the March 6, 1998 rerate submittal along with Questions 24 and 25 of the September 5, 1997 submittal. Information on system pressure vessels is provided in the response to question 11 below.

10. *Section 7.4 states that the feedwater regulating valves were originally designed for greater than warranted flow conditions and are therefore adequate. What are the original design flows of the feedwater regulating valves, and how do they compare with the flow conditions that result from power rerate?*

NSP Response

The feedwater regulating valves are Copes Vulcan 14 - 900 PSI USA Std globe valves. The maximum flow through a valve is 9,000 gpm. The flow at 1775 MWt rerate power will be 7,897 gpm.

NSP evaluated the runout flow for various plant parameters. With the current valve setting (C_v = 1350), the calculated runout flow was determined based on expected rerate reactor operating conditions of 1025 psia. This is the normal pressure used in plant design for nominal system conditions which corresponds to operation at the design turbine inlet pressure of 965 psia. With a condensate demineralizer differential pressure of 30 psid (mid-range of normal values), the runout is 113.6% for 1670 MWt and 106% for 1775 MWt. With the current setting, the valves will provide adequate runout flow at rerate conditions.

11. *Section 7.4 states that the ASME Section VIII feedwater heaters will be analyzed and verified to be acceptable for the slightly higher feedwater heater temperatures and pressures for the 1775 MWt power rerate What is the acceptability for power rerate based on for the feedwater heaters now? What Is the effect of the higher flow rates on the heaters? When are the feedwater heaters expected to be recertified for the higher temperature and pressures?*

What modifications are planned to ensure that adequate feedwater flow margin at 1775 MWt exists?

NSP Response

An analysis of the feedwater heaters for operation at 1775 MWt has been completed. The analysis was performed in accordance with ASME Section VIII Division I. The effects of changes in temperature, pressure, and flow were considered. All feedwater heaters were shown to meet applicable ASME Section VIII Division I requirements for operation at 1775 MWt. Some of the feedwater heaters require re-rating which involves application of new code nameplates. This re-rating is scheduled for the 1998 refueling outage. Analysis has shown adequate feedwater flow margin exists at runout conditions at 1775 MWt without modification. See Question 29 of NSP's September 5, 1997 rerate submittal for additional information.

12. *Section 10.1 states that the slight changes in feedwater, condensate, and RWCU [reactor water cleanup] temperatures result in an insignificant increase in the mass and energy release rates following high energy line breaks. Explain how insignificant the increase is, and how it correlates to the margin in the existing EQ [environmental qualification] envelopes.*

The temperature and enthalpy changes in the feedwater, condensate, and RWCU process streams are shown in Table 12-1 below.

Table 12-1

	1670 MWt Temperature	1775 MWt Temperature	1670 MWt Enthalpy	1775 MWt Enthalpy
Outlet Hi Press Heaters	376.3 °F	382.7 °F	350.9 H	357.8 H
Outlet FW Pumps	304.5 °F	305.6 °F	276.3 H	277.5 H
Outlet Low Press Heaters	302.1 °F	303.0 °F	271.6 H	272.8 H
Inlet Low Press Heaters	93.6 °F	91.7 °F	61.6 H	59.7 H
RWCU Inlet	530 °F	523. °F	524.6 H	523.7 H

Feedwater and Condensate Line Breaks

The analysis of the feedwater High Energy Line Break (HELB) and the condensate HELB in the feed pump area showed that all areas containing safe shutdown equipment remain mild (less than 120 °F). The feedwater HELB at the feed pumps was analyzed at 1880 MWt conditions and bounds the condensate HELB for these areas. The temperatures and pressures for these areas remain unchanged from the 1670 MWt to rerate conditions, except for the motor control centers B-42 A & B and B-43 A & B area which increased in temperature by 0.7 °F to 109.7 °F.

The analysis of the feedwater HELB and the condensate HELB in the condenser area showed that all the areas containing safe shutdown equipment remain mild (less than 120 °F). The feedwater HELB in the condenser area was analyzed at 1880 MWt conditions and bounds the condensate HELB for these areas. The temperatures and pressures for these areas remain unchanged from the 1670 MWt to the rerate conditions, except for the motor control centers B-33 A & B area which increased in temperature by 0.4 °F to 109.4 °F.

The feedwater HELB on the turbine deck was analyzed at 1880 MWt conditions. The analysis of the feedwater HELB on the turbine deck showed that all the areas containing safe shutdown equipment remain mild (less than 120 °F). The temperatures and pressures for these area remained unchanged from the 1670 MWt to the rerate conditions, except for the 4 KV Load Center Division II area which increased in temperature by 0.1 °F to 103.1 °F.

The analysis of the feedwater break in the steam chase showed that the pressure peaks throughout the reactor building remain unchanged from the 1670 MWt to the 1880 MWt conditions. The majority of the temperature increases throughout the entire reactor building were less then 1 °F. The maximum increase in peak temperature was 3.2 °F to 171.6°F for the 935 ft. elevation of the southwest portion of the reactor building. The equipment located in this area is qualified to 197 °F based on the current bounding HELB analysis calcuiated with RELAP 4 / MOD 5. For the feedwater break in the steam chase at rerate conditions, all reactor building areas that contain equipment needed to mitigate the HELB are bounded by current the 1670 MWt qualification analyses calculated with RELAP 4 / MOD 5.

RWCU Line Break

The RWCU high energy line break analysis was revised to reflect new, more conservative assumptions. See NSP's submittal dated April 11, 1997, "Revision One to License Amendment Request Dated July 26, 1996 Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability." The revised RWCU HELB at 1670 MWt conditions is based on a mass flow rate of 719.3 lbm/sec, an enthalpy of 524.6 BTU/lbm, and a duration of 616.5 seconds.

A modification is planned to install automatic isolation logic for the RWCU System for the 1998 refueling outage. After the modification, the RWCU system will automatically isolate on high flow rate or high area temperature. The RWCU HELB at 1775 MWt conditions with automatic isolation was evaluated using a mass flow rate of 730.2 lbm/sec and an enthalpy of 523.7 BTU/lbm for a duration of 26.5 seconds.

The amount of energy postulated to be released from the RWCU HELB at 1670 MWt and the rerate RWCU HELB with automatic isolation is showed in Table 12-2 below.

Table 12-2

	RWCU HELB 1670 MWt	RWCU HELB AUTOMATIC ISOLATION 1775 MWt
Energy (BTUs)	232,633,060	10,133,752

The effects of the RWCU high energy line breaks at rerate conditions were evaluated and the resulting temperatures and pressures in most cases are substantially reduced. The temperatures and pressures for all current harsh areas were reduced, except for the 896' elevation, equipment and floor drain tank room, which increased in temperature by 0.1 °F to 140.7 °F. The 896' elevation equipment and floor drain tank room initial temperature prior to the HELB was 140.0 °F.

The environmental qualification profiles are under development and will be completed prior to implementing rerate. See NSP's response to question 16.

13. *Section 10.1.1 states that GOTHIC was benchmarked against previously approved high energy line break analysis outputs, and the comparison results were found to be in close agreement. Explain what is meant by "close agreement." Was the output of GOTHIC compared to the previously approved EDSFLOW output using the same input assumptions? Are the GOTHIC results more or less conservative than EDSFLOW? Provide the corresponding temperature, pressure, and humidity profiles that support the statement.*

<u>NSP Response</u>

Vectra calculation 091-19407-C-3 documents a verification comparison of outputs results between a GOTHIC Version 4.0 thermal/hydraulics computer model of the turbine building and the same model results obtained previously using RELAP 4 / MOD 5 computer code. This comparison used the same input assumptions and the same model inputs. The results demonstrate that the GOTHIC model results are comparable with minor differences in peak temperatures that are within 5%. EDSFLOW is ABB Impel's proprietary version of RELAP 4 / MOD 5 thermal-hydraulics computer code. A copy of this calculation is provided as Attachment 6. Calculation table 3-1 provides a comparison of the GOTHIC and RELAP 4 / MOD 5 results.

14. *Section 10.1.1.1 states that the mass and energy release for the Main Steam Line Break (MSLB) outside containment was calculated using the SAFER/GESTR-LOCA model. Is this the same model that was used during the initial licensing of Monticello for MSLBs? If not, explain the differences between the models and any changes in assumptions.*

NSP Response

The SAFER/GESTR-LOCA model is not the same model that was used in the original Monticello FSAR calculations for the Main Steam Line Break (MSLB) outside containment. A description of the SAFER/GESTR-LOCA model, code qualification, and documentation of NRC approval is provided in NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," October 1984.

The assumptions used in the original FSAR main steamline break analysis are described in Items A through L in Section 14.7.3.1.1 of the MNGP USAR (Items M through P apply to a later TRACG analysis for the MSLB). There are no significant differences in the system modeling assumptions such as reactor conditions, break modeling, isolation valve characteristics, between the original FSAR calculations and the SAFER/GESTR-LOCA analysis. The significant differences between the original FSAR calculations and the SAFER/GESTR-LOCA analysis are in the modeling of the water level rise in the vessel and the quality of the steam-water mixture discharged through the break.

In calculating the rate of water level rise, the original FSAR assumed that the steam bubbles formed during depressurization rise at an average velocity of about 2 feet per second relative to the liquid. The SAFER/GESTR-LOCA model uses either a bubble rise or a drift flux formulation, depending on void fraction and flow conditions, to determine the vapor flow rate leaving the liquid region and resulting level swell during the transient. This modeling difference affects the time that the two-phase mixture reaches the level of the main steam line inlets. In the original FSAR calculations, the two-phase mixture reached the level of the steam lines in about 2 seconds. The two-phase mixture reached the level of the steam lines in about 4 seconds in the SAFER/GESTR-LOCA analysis, resulting in less liquid mass release during the accident.

In the original FSAR calculations, a constant quality is assumed at the steam line inlets. This assumption neglects the actual increase in quality that occurs as the water level in the vessel decreases during the latter part of the accident. In the SAFER/GESTR-LOCA analysis, the quality of the steam-water mixture discharged through the break is determined by the quality of the two-phase mixture in the vessel and how much of the steam line inlet is covered by the mixture. The SAFER/GESTR-LOCA analysis accounts for the change in quality during the accident which results in less liquid mass release during the accident.

As a result of these two modeling differences, the mass release calculated in the SAFER/GESTR-LOCA analysis is significantly less than the mass release determined in the original FSAR calculations. It should be noted that both analyses neglected the flow resistance in the main steam lines and averaging manifold. As demonstrated by the TRACG MSLB analysis in Section 14.7.3 of the MNGP USAR, this assumption introduces a large conservatism into the mass and energy release calculation.

Independent of the model used in the analysis, the MSLB mass and energy releases for power rerate are bounded by the releases for the current power condition. The mass and energy releases for the steam line break are largely determined by the amount of liquid discharged through the break. Following the break, the vessel rapidly depressurizes because the steam generation from the decay power cannot make up the steam loss through the break. The rapid depressurization causes the water in the vessel to flash and swell up to the steamlines, resulting in a steam-water mixture flowing out the break. This mixture flow continues until the MSIVs close. The steamline break flow is determined by the reactor pressure and the steamline flow restrictor area; these analysis inputs remain unchanged for power rerate. Therefore, the flow through the break is not affected by power rerate.

The initial core power determines the amount of steam generation during this period, which in turn determines the depressurization rate and resulting level swell. A higher initial core power level results in a higher steam generation rate. The combination of the unchanged break flow and higher steam generation rate results in a lower vessel depressurization rate and delays the level swell. Because the MSIV closure time is not affected by power level, the delayed level swell results in less steam-water mixture being released out the break. Therefore, the mass and energy release for a steamline break are less limiting at higher power levels and the consequences of a MSLB at rerate power conditions are bounded by the analysis for current power. See Exhibit A to NSP's submittal dated April 11, 1997, "Revision One to License Amendment Request Dated July 26, 1996 Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability," for a description of the MSLBA.

15. *Section 10.1.1.6 states that the Moody Slip break flow model was used to calculate the mass flux for the steam jet air ejector steam line break. Is this the same model that was used during the initial licensing of Monticello for the steam jet air ejector steam line break? If not, explain the differences between the models and any changes in assumptions.*

NSP Response

The Moody Slip break flow model used for rerate was also used for the initial MNGP evaluation of the steam jet air ejector line break.

16. *What is the basis for stating that all equipment affected by the various line breaks discussed in Section 10.0, will remain qualified?*

NSP Response

An evaluation comparing power rerate environments to the existing equipment environmental qualification documentation was performed. The evaluation compared the peaks and profile outputs of GOTHIC models at 1670 MWt and at rerate conditions to the peaks and profile outputs from RELAP 4 / MOD 5 models available in the EQ Central File. The GOTHIC models used provided greater details and a more accurate representation of the plant than the RELAP models. The GOTHIC model includes additional volumes and areas. It also includes additional flow vent paths due to ventilation ducting and subdivides some rooms. In general, the RELAP model outputs bounded the GOTHIC model outputs. In the areas where the GOTHIC model outputs were bounding, qualification for affected equipment was evaluated by comparing the resulting peaks and profiles to component data in the EQ Central File and in the test reports. This evaluation confirmed that the current environmentally qualified equipment would remain qualified at power rerate conditions. The evaluation also determined that the scope of equipment in the EQ program does not change because operation at power rerate does not involve the creation of any new harsh environmental areas.

The service life and maintenance intervals for affected equipment within the EQ program will be adjusted, as necessary, to reflect power rerate operating conditions.

17. *What testing will be performed on the feedwater and condensate systems prior to the implementation of power rerate?*

NSP Response

Testing performed prior to implementation is based on the guidance provided in the "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32428P, February 1995. Some additional plant-specific testing will be performed based on recommendations from vendors and from the NSP engineering staff. The feedwater and condensate system testing includes the following.

Verification of acceptable span and calibration of the feedwater flow transmitters.

Verification of acceptable margin to reactor feedwater pump suction pressures and temperatures.

Verification that actual condensate and feedwater system pressures and flows coincide with predicted values used in the feedwater system runout analysis.

Verification of the ability to achieve >105% of the rated feedwater flow that corresponds to a reactor thermal power of 1775 MWt at the design turbine inlet pressure. This is the expected normal operating pressure for power rerate.

Verification that the maximum runout capability is less than the value assumed in the fuel thermal limits transient evaluation.

Verification of appropriate feedwater regulating valve setup consistent with linear regulation between flow and demand for feedwater.

Transient performance testing is performed to confirm appropriate control system response of the reactor level control system to a level/flow transient.

Following implementation of power rerate, testing will also be done to confirm that the level in the high pressure and high intermediate pressure feedwater heaters is adequate to prevent bypassing of steam into the subcooling zone of the feedwater heaters. A system walkdown will be performed to verify that there is no unexpected flow induced vibration or motion of condensate and feedwater lines including small bore branch connections. The condensate and feedwater pump motor currents also will be measured to verify that the load study assumptions are correct.

18. *For the slightly increased power rerate temperature and pressure conditions, do the non-metallic parts of non-electrical equipment/components (pumps, heat exchangers, etc.) continue to meet the following design and qualification requirements:*

a. Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs [Loss of cooling accidents].

b. Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety related functions.

c. Design control measures shall be established for verifying the adequacy of design.

d. Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

<u>NSP Response</u>

The design and qualification requirements above continue to be met at power rerate conditions. The MNGP Operational Quality Assurance Plan is provided in Appendix C of the USAR. This plan provides the basis for the site's Quality Assurance Program that is implemented by documented administrative controls. All structures, systems and components (SSC) that could be impacted by changes associated with the Power Rerate Program were identified. Engineering Evaluation Task Reports were prepared for all impacted equipment and topical areas. Each Engineering Evaluation Task Report identified how the structure, system or component was impacted and performed an evaluation to determine adequacy or the need for additional actions. Evaluations included consideration of service conditions such as changes in system pressures, temperatures, and flow rates. In most cases, the SSC was shown to be within its original design capabilities and no additional actions were needed.

If any SSC, including non-metallic parts of non-electrical components, were impacted in such a manner that its design needed to be changed to assure compatibility with normal or accident conditions, these changes were done under NSP's Quality Assurance Program. These types of changes were processed under the requirements of the modification process. The Quality Assurance Program as implemented by the modification process provides controls to insure that the design and qualification requirements listed in parts 18 a through d are met.

19. *Describe the Monticello power rerate process. Describe in detail the extent of the NSP's oversight and independent review of the contractor work and the QA [quality assurance] involvement in the power rerate process. Provide a list all engineering evaluation reports (i.e., transient analyses, pipe stress analysis, RPV [reactor pressure vessel] stress report, system evaluation reports, analysis basis documents, etc.) that were completed to support the conclusions documented In Exhibit E of the license amendment request. For each report indicate preparer (i.e., GE or NSP) and reviewer organization. Describe the process used to ensure accuracy of plant-specific input assumptions and acceptability of results.*

NSP Response

By letter dated Jan. 20, 1997, NSP provided some general information that described the rerate processes. This information continues to apply. These processes are intended to assure the validity of the rerate design basis inputs and to assure compliance with design basis requirements. Supplemental information on these processes is provided below.

The rerate activities, including analyses, testing, data, inputs, procedures, and training, were and will continue to be conducted under the provisions of the NSP and GE Nuclear Energy Quality Assurance Programs. Audits and surveillance activities were conducted to measure the effectiveness of these quality programs.

An audit of the power rerate project and the licensing submittal was performed by the NSP Quality Assurance group. The principal objective of this audit was to confirm that the material that supported the rerate license amendment request (then Rev. 0) was technically sound, comprehensive, and accurate. This audit included a review of selected topical reports and supporting information. It also included detailed examinations of selected design basis documents, USAR sections, engineering evaluations, calculations, and power rerate topical reports. The audit focused on those systems more directly affected by rerate such as primary containment, electrical power, feedwater and condensate, and power conversion systems. The findings from this audit were resolved and incorporated into Rev. 1 of the power rerate license amendment request as appropriate.

A separate independent review of power rerate was performed between January and April 1997. This independent review was performed by three outside contractors and one retired NSP employee with broad and extensive experience at MNGP. The principal objectives of this review were to: review technical information with the focus on equipment required for plant safety; perform a compliance review against NRC and GE guidance documents; and review the MNGP Technical Specifications against the rerate license amendment request (then Rev. 0). As part of the independent audit, the team visited the NSP Nuclear Analysis Department and the GE San Jose facility to review the work associated with the rerate project. The NSP visit focused on rerate transient analyses, and the GE visit focused on the containment, LOCA, and the radiological analysis. This review concluded that, in general, the request was technically correct and in compliance with the guidance documents. The open items generated from this review were resolved and incorporated into Rev. 1 of the power rerate license amendment request as appropriate.

The power rerate processes were deliberately configured to address design basis issues. The Monticello staff had heightened awareness of design basis issues as engineers and supervisors responsible for the power rerate task reports and the review of safety related systems and special design programs (Appendix R, ATWS, SBO, etc.) had recently participated in the development and maintenance of a comprehensive Monticello design basis reconstitution program. NSP was mindful of the design basis errors associated with the Maine Yankee and Brunswick uprates and took specific programmatic steps to provide barriers to prevent design basis errors during the development of power rerate at MNGP.

Each task report developed by NSP, including the associated design inputs, received the following minimum set of reviews. Many task reports received additional reviews by teams of subject matter experts.

System Engineer or Program Engineer
NSP Engineering Supervisor (or designee)
GE Cognizant Engineer
NSP Technical Review Team

Each task report developed by GE, including the associated design inputs, received the following set of reviews.

GE Cognizant Engineer
GE Supervisor
NSP Engineering Supervisor (or designee)
NSP Technical Review Team

The rerate license amendment request has received the following set of reviews.

NSP Subject Matter Experts (Engineering, Transient Analyses, Licensing etc.)
GE Subject Matter Experts
NSP Technical Review Team
MNGP Operations Committee
MNGP Safety Audit Committee
MNGP Quality Assurance Audit (Rev. 0)
Independent Audit Group Review (Rev. 0)

The majority of the power rerate analyses were performed by senior NSP engineers and supervisors with extensive experience at MNGP. The GE scope was limited to those analyses that GE has historically performed for NSP such as accident modeling. NSP conducts its own transient analyses. According to the Power Rerate Program Plan, NSP and GE were required to mutually agree on the significant system and analysis parameters used for GE rerate engineering analyses. GE did not have the authority to unilaterally alter, neglect, or judge the adequacy of the power rerate analyses inputs or results. It was NSP's ultimate responsibility to certify that rerate analyses and results were in compliance with the applicable licensing criteria and with the MNGP design bases.

All task reports were developed by NSP or GE. Some power rerate evaluations were contracted to consultants who routinely perform similar work for NSP at Monticello. Various power rerate structural calculations were performed by the Minneapolis office of Duke Engineering Services (then Vectra). The Vectra engineers at this office conducted the original Mark I program analyses for MNGP. This work was accomplished under the Vectra QA Plan. Tenera, NSP's principal IPE and IPEEE consultant, was contracted to perform work in the PRA area.

The power rerate program included a review of the power rerate impact on the USAR, staff safety evaluation reports, and staff generic communications. As the current licensing basis evolves, an ongoing analysis of power rerate impact is continuing until the power rerate program is approved. The scope and extent of the licensing evaluations are described in Section 11 of NSP's submittal dated December 4, 1997.

Please see Attachment 8 for a description of the power rerate work scope. Some numbers have been superseded because the associated tasks and sub-tasks were incorporated into other reports.

Attachment 5

Rerate Questions from Conference Calls Between Staff and NSP
on March 10, and March 17, 1998

1. *Does the increased power level of the reactor affect the EOC transient (accident) analysis due to the increased plutonium buildup and the corresponding decrease in delayed neutron fractions. How was this considered in the transient analysis Does the increased plutonium affect the EOC or the next BOC.*

NSP Response

The transient analyses performed by NSP for power rerate incorporates the changes in plutonium inventory and reactor kinetics data (i.e. neutron delay fraction) as a consequence of an increase in power. This incorporation is inherent in the methodology in that lattice physics cross sections and kinetics data are input into the 3-D simulator code. The 3-D simulator code output is used to construct reactivity coefficients and kinetics data for the transient analysis. The transient analyses performed at the end of hot full power and at the beginning of cycle include the effects of the plutonium by virtue of the output link to the lattice physics data.

The only design basis accident that may be affected by a change in reactivity coefficients and kinetics data is the Control Rod Drop Accident (CRDA). The CRDA is analyzed generically for Banked Position Withdrawal plants (including MNGP). The plant-specific control rod withdrawal sequences used throughout the cycle are analyzed using the 3-D simulator code to confirm that the sequences are within the generic CRDA requirements for maximum rod worth. As described above, the 3-D simulator code incorporates the changes in plutonium and reactor kinetics data due to power rerate. Therefore, the effects of the plutonium buildup are considered by virtue of the use of the 3-D simulator to confirm the acceptability of the plant-specific control rod sequences.

2. *In question 10 of the March 6, 1998 submittal, it appears that the maximum calculated stresses are very close to the allowable stress limit. The stress ratio for the main steam line was 97% and 99% for the torus attached piping. Provide a comparison of maximum stress and fatigue usage factors for the existing design basis analysis and for the power rerate condition. Also provide the code and code edition and the location at which the maximum stresses occur.*

NSP Response

No fatigue usage evaluation is performed for the main steam lines. The governing code for the main steam lines is ANSI B31.1 which does not require a fatigue evaluation to be performed. Fatigue usage for the torus attached piping was performed generically as part of the original Mark I Torus Attached Piping Project. The fatigue usage for all torus attached piping was shown to be less than 0.5. No new fatigue usage was calculated for rerate since rerate has no significant impact on the current fatigue usage. See NSP's response to Question 22 of NSP's submittal dated September 5, 1997.

At present power levels, the maximum stress ratio for main steam for the "DW + P + TSVC" load case is 88% (15926 psi/18000 psi), and the location is near the connection of the main steam line to the SRV piping. At present power levels, the maximum stress ratio for torus attached piping for the "TH" load case is 88% (19800 psi / 22500 psi), and the location is at the torus penetration.

The design code for the main steam lines is ANSI B31.1, 1977 edition with addenda up to and including Winter 1978. The code for torus attached piping is ASME Code Section III, 1977 edition with addenda up to Winter 1978.

3. *Please provide an evaluation of the MOVs contained in systems such as Reactor Core Isolation Cooling (RCIC), High Pressure Core Injection (HPCI), and Core Spray for the changes in reactor operating pressure.*

NSP Response

Changes in operating pressures affects the amount of thrust/torque that is required to reposition a valve. As part of Monticello's MOV evaluation, these thrust/torque requirement changes are compared to the actuators thrust/torque capabilities and actuator/valve weak link allowables. An evaluation of Monticello MOVs with design basis functions at rerate conditions, which includes operating pressure changes, was performed which compared the MOVs thrust/torque requirements against their capabilities/allowables and confirmed that these MOVs are capable of performing their required functions at rerate conditions.