UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT DOCKET NO. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE DPR-22

REVISION 1 TO LICENSE AMENDMENT REQUEST DATED JULY 26, 1996
SUPPORTING THE MONTICELLO NUCLEAR GENERATING PLANT
POWER RERATE PROGRAM

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Monticello Operating License as shown on the attachments labeled Exhibits A, B and C. Exhibit A describes the proposed changes, describes the reasons for the changes, and contains a Safety Evaluation, a Determination of Significant Hazards Consideration and a summary of the MNGP Power Rerate Environmental Evaluation. Exhibit B contains current Technical Specification pages marked up with the proposed changes. Exhibit C provides the affected MNGP Technical Specification pages with the proposed changes incorporated. Supporting Exhibits D through K, Exhibit F coversheet only, are attached.

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 4th day of December 1997 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 and that it is not interposed for delay.

Samuel I. Shirey

Samuel I. Shirey
Notary Public - Minnesota
Sherburne County
My Commission Expires January 31, 2000

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

General Electric Company

AFFIDAVIT

I, **George B. Stramback,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") 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 is contained in the GE proprietary report NEDC-32546P, *Power Rerate Safety Analysis Report for Monticello Nuclear Generating Plant,* Revision 1, Class III (GE Proprietary Information), dated December 1997. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE 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 10 CFR 9.17(a)(4), 2.790(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), and 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 General Electric's competitors without license from General Electric 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, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

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

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and 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. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains detailed results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power rerate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are <u>not</u> so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive

physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power rerate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 3rd day of December 1997.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 3rd day of December 1997.

Anna Hanlin

Notary Public, State of California

ANNA HANLIN
COMM. #1030164
NOTARY PUBLIC-CALIFORNIA
SAN FRANCISCO COUNTY
My Comm. Expires June 19, 1998
PC01

PROPRIETARY INFORMATION NOTICE

This document contains proprietary information of the General Electric Company (GE) and is furnished to Northern States Power Company (NSP) in confidence solely for the purposes stated in the transmittal letter. No other use, direct or indirect, of the document or the information it contains is authorized. NSP shall not publish or otherwise disclose it or the information to others without written consent of GE, and shall return the document at the request of GE.

IMPORTANT INFORMATION REGARDING

CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the Power Rerate contract between Northern States Power Company (NSP) and GE, as identified in Purchase Order Number PH0303SQ, dated December 27, 1994, as amended to the date of transmittal of this document, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than NSP, or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, GE makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

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COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 3rd day of December 1997.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 3rd day of December 1997.

Anna Hanlin

Notary Public, State of California

ANNA HANLIN
COMM. #1030164
NOTARY PUBLIC-CALIFORNIA
SAN FRANCISCO COUNTY
My Comm. Expires June 19, 1998
PC01

EXHIBIT A

REVISION 1 TO LICENSE AMENDMENT REQUEST DATED JULY 26, 1996

EVALUATION OF PROPOSED CHANGES TO FACILITY OPERATING LICENSE DPR-22
AND THE TECHNICAL SPECIFICATIONS FOR OPERATING LICENSE DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

MONTICELLO, MINNESOTA

LICENSE NO. DPR-22

DOCKET NO. 50-263

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EXHIBIT A

Monticello Nuclear Generating Plant

Revision 1 to License Amendment Request Dated July 26, 1996

Evaluation of Proposed Changes to Facility Operating License DPR-22
and the Technical Specifications for Operating License DPR-22

I. Reason for Proposed Changes

Northern States Power (NSP), licensee under Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP) requests that the license be amended as proposed herein to reflect the MNGP Power Rerate program to be implemented at MNGP. Specifically, this amendment request proposes to increase the maximum reactor core power level by 6.3%, to 1775 megawatts thermal (MWt) from the current limit of 1670 MWt. This request includes the associated changes to the plant Technical Specifications based on the MNGP Power Rerate evaluations.

Continuing improvements and sophistication in the analytical techniques (i.e., computer codes and data) based on several decades of Boiling Water Reactor (BWR) safety technology, plant performance feedback, and improved fuel and core design, have resulted in a significant increase in the difference between calculated safety analysis results and the current licensing limits which establish margins of safety. This available analysis margin, combined with the excess capability of the as-designed equipment, systems and components, provide the potential for an increase of approximately 6.3% in the full power rating of the plant without the need to perform major Nuclear Steam Supply System (NSSS) or Balance-of-Plant (BOP) hardware modifications. The full power level can be increased safely, and the installed systems and equipment are capable of performing required functions at rerate conditions.

The strategy for achieving higher power is to expand the power/flow map by increasing reactor power within existing rod and core flow control lines. However, there will not be an increase in the core flow at 100% of rerate power over the current licensed value of core flow at 100% of rated power. See new commitment regarding additional analysis for operation above currently licensed rated core flow values as represented in Figure 5 of the Core Operating Limits Report in Exhibit H of this LAR revision.

Increased reactor thermal power results in an increase in reactor steam flow with a corresponding higher turbine inlet steam flow. Coordination of MNGP Power Rerate with the recently completed modifications to the high pressure steam turbine provides for a redesigned turbine steam flow path to achieve increased steam flows and ultimately a higher operating electrical power. No changes are required to the rated core flow, rated reactor pressure, or turbine throttle pressure to implement MNGP Power Rerate.

This License Amendment Request provides a discussion and description of the proposed changes, a safety assessment of the proposed changes, information supporting a finding of No Significant Hazards Consideration in accordance with the criteria of 10CFR50.92, and an environmental evaluation demonstrating no significant effect on the human environment and exclusion from environmental review in accordance with 10CFR51.22(c)(9).

Pursuant to 10CFR50.90, NSP hereby proposes the following changes.

II. Description of the Proposed Changes

The proposed Operating License and Appendix A changes associated with the planned implementation of MNGP Power Rerate are as follows. Changes to the current Operating License and Technical Specifications are indicated by bold italic text.

A. Changes to the Facility Operating License and Technical Specification Changes to Reflect the Proposed MNGP Rerate Power Level.

1. Operating License DPR-22, Docket No. 50-263, page 3, paragraph C.1, Maximum Power Level.

   a) The Facility Operating License for the Monticello Nuclear Generating Plant states:

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1670 megawatts (thermal).

   b) The Facility Operating License for the Monticello Nuclear Generating Plant is proposed to be changed to state:

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ***1775*** megawatts (thermal).

2. Technical Specifications, Section 1.0, DEFINITIONS, paragraph 1.R, page 4.

   a) The definition for Rated Neutron Flux provided in Section 1.0, page 4 of the Technical Specifications states:

      R. Rated flux is the neutron flux that corresponds to a steady-state power level of 1670 thermal megawatts.

   b) The definition is proposed to be changed to state:

      R. Rated flux is the neutron flux that corresponds to a steady-state power level of ***1775*** thermal megawatts.

3. Technical Specifications, Section 1.0, DEFINITIONS, paragraph 1.S, page 4.

   a) The definition for Rated Thermal Power provided in Section 1.0, page 4 of the Technical Specifications states:

      S. Rated thermal power means a steady-state power level of 1670 thermal megawatts.

   b) The definition is proposed to be changed to state:

      S. Rated thermal power means a steady-state power level of ***1775*** thermal megawatts.

4. Technical Specifications, Section 2.3, Bases, pages 14 and 15.

   a) Revise the bases discussion to reflect that the abnormal operational transients have been analyzed to thermal powers of 1775 MWt and to reflect the proposed change to the licensed power level of 1775 MWt on page 14 of the bases for section 2.3.

   b) Revise paragraph 'A' on page 15 of the bases for section 2.3 to reflect the proposed change to the licensed power level of 1775 MWt.

B. APRM Neutron Flux Scram, APRM Rod Block Monitor and Single Loop Operation

1. Technical Specification, Section 2.0, SAFETY LIMITS, subsection 2.3, FUEL CLADDING INTEGRITY , Specifications 2.3.A.1.a, and 2.3.A.1.b, page 6.

   a) The specifications states:

   The Limiting safety system settings shall be as specified below:

   2.3.A Neutron Flux Scram

   1. APRM - The APRM flux scram trip setting shall be:

   a. For two recirculation loop operation (TLO):

   $S \leq 0.66W + 70\%$ *where*

   S = Setting in percent of rated thermal power, rated power being 1670 MWt.
   W = Percent of the drive flow required to produce a rated core flow of 57.6 x 10^6 *lb/hr.*

b. For single recirculation loop operation (SLO):

$S \leq 0.58(W - 5.4) + 62\%$

b) The specifications are proposed to be changed to state:

The Limiting safety system settings shall be as specified below:

2.3.A Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:

   a. For two recirculation loop operation (TLO):

      $S \leq 0.66W +$ ***65.6***% where

      S = Setting in percent of rated thermal power, rated power being ***1775*** MWt.
      W = Percent of ***recirculation*** drive flow required to produce a core flow of 57.6×10^6 lb/hr.

   b. For single recirculation loop operation (SLO):

      $S \leq$ ***0.66***(W - 5.4) + ***65.6%***

2. Technical Specification, Section 3.2, PROTECTION INSTRUMENTATION, TABLE 3.2.3, "Instrumentation That Initiates Rod Block," Function item 3, page 56.

   a) The specification provides limiting conditions for operation for instrumentation that provides control rod block actuation per MNGP Technical Specification 3.2.C.2.b, and specifies that the Average Power Range Monitor (APRM) provides input to the Rod Block Monitor (RBM) to inhibit rod movement at the following trip settings:

      a. Upscale

      (1) TLO, Flow Biased $\leq 0.66W + 58\%$

      (2) SLO, Flow Biased $\leq 0.58(W - 5.4) + 50\%$

b) The specification is proposed to be changed to such that TABLE 3.2.3, Function item 3, specifies the following APRM Upscale trip settings:

a. Upscale

(1) TLO, Flow Biased $\leq$ 0.66W + ***53.6%***

(2) SLO, Flow Biased $\leq$ ***0.66***(W - 5.4) + ***53.6%***

3. Technical Specification Section 2.3, Bases, Page 16. Technical Specification Section 3.5/4.5, Bases, Paragraph F, Recirculation System, Page 114.

a) The discussion in the Technical Specifications bases concerning special operating features found on pages 16 and 114 is to be revised to provide reference to the evaluation performed for MNGP Power Rerate. This evaluation demonstrated that these features remain as acceptable operating modes without an adverse effect on plant safety. The MNGP Power Rerate evaluations have provided a revised APRM flow-biased scram and rod block equation for single loop operation consistent with the Maximum Extended Load Line Limit Analysis (MELLLA) which assures adequate core protection for the postulated transient events and accidents. The bases for Sections 3.5/4.5 are also being updated to include the ARTS methodology which has been previously approved for MNGP.

C. Safety Relief Valves

1. Technical Specification Section 2.2, Bases, page 23; Section 2.4 Bases, page 24; Section 3.6 and 4.6, Bases, paragraph E, page 150.

a) On page 23 of the bases for section 2.2, the bases discussion is revised to clarify the design basis pressurization event as it applies to SRV capacity and to clarify the reactor pressure safety limit.

b) On page 24 of the bases for section 2.4, paragraph 2, the bases discussion concerning safety/relief valves is revised to reflect the proposed change in the licensed thermal power limit to 1775 MWt. The bases discussion is revised to reflect that the abnormal operational transients for reactor pressure protection have been analyzed assuming five of the eight safety relief valves (SRVs) are operable and that they open at 3% over their setpoint to reflect analysis performed for MNGP Power Rerate. The discussion of safety/relief valve setpoint compliance with the ASME Boiler and Pressure Vessel Code is revised to reflect the Code requirement that at least one safety relief valve is set to open at a pressure no greater than design pressure. The bases are revised to reflect that the design function of the HPCI and RCIC systems have been conservatively evaluated at the upper limit of the safety/relief valve setpoint and have been demonstrated to be acceptable. The bases is also being revised to clarify that the maximum operating pressure of the HPCI and RCIC systems is limited by the safety grade single-failure proof Low-Low Set SRV system. See NSP's response to Question 23 of the staff's Rerate RAI dated September 5, 1997.

On page 24 of the bases for section 2.4, paragraph 3, the bases discussion is revised concerning safety/relief valve setpoint deviation to reflect that the ANSI/ASME Code OM-1-1981 specifies an as found acceptance criteria of 3% above the valves set pressure, which is further supported by analysis of overpressure events performed for MNGP Power Rerate. Thus the as-found Safety/relief valve setpoints can be as much as 22.3 psi above the 1120 psig (1109 plus 1% tolerance) as-left setpoint.

c) On page 150 of the bases for section 3.6 and 4.6, revise the bases discussion to reflect that the applicable code (ANSI/ASME OM-1-1981) specifies an as-found tolerance for safety/relief valve setpoints of 3%, and that the MNGP Power Rerate analyses assumed a valve setpoint 3% high (1142.3 psig) with no adverse effect on overpressure protection. In addition, the bases is revised to reflect License Amendment 92, which reduced the test frequency from seven of the valves being tested every refueling outage to 20% of the valves being tested within any 24 months in accordance with ANSI/ASME OM-1-1981.

D. Condenser Low Vacuum Scram

1. Technical Specification Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS, Trip Function item 9, page 28.

   a) The specifications states that the Turbine Condenser Low Vacuum Limiting Trip Settings is $\geq$ 23 inches of mercury (23 in. Hg).

   b) The specification is proposed to be changed to state that the Turbine Condenser Low Vacuum Limiting Trip Setting is $\geq$ ***22*** inches of mercury (22 in. Hg).

2. Technical Specification Section 3.1, Bases, page 37.

   a) The bases for the Condenser Low Vacuum Scram, is proposed to be revised to reflect that scram function occurs at greater than or equal to 22 inches of mercury (22" Hg) vacuum based on MNGP Power Rerate evaluations.

E. Turbine Control Valve Fast Closure Scram and Turbine Stop Valve Scram

1. Technical Specification, Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS, Required Conditions when minimum conditions for operation are not satisfied, item D; page 30.

   a) Item D applies to TABLE 3.1.1, item 11, Turbine Control Valve Fast Closure; and TABLE 3.1.1, item 12, Turbine Stop Valve Closure; found on page 29 of the MNGP Technical Specifications. The specification establishes required plant conditions to be established when the operability requirements of TABLE 3.1.1 are not satisfied and the limiting conditions for operation of specification 3.1.B.1 and 3.1.B.2 can not be satisfied and states:

      D. Reactor power less than 45% (751.5 MWt).

   b) The limiting condition for plant operation is to be revised based upon MNGP Power Rerate analysis. The specification is proposed to be changed to state:

      D. Reactor power less than 45% (***798.75*** MWt).

2. Technical Specification, Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS," Allowable Bypass Conditions, item d; page 30.

   a) Item 'd' applies to TABLE 3.1.1, item 11, Turbine Control Valve Fast Closure; and TABLE 3.1.1, item 12, Turbine Stop Valve Closure; found on page 29 of the MNGP Technical Specifications. The specification establishes allowable plant conditions for which the trip functions may be bypassed as an amplifying note to the column with heading "Modes in which function must be Operable or Operating" in TABLE 3.1.1, and states:

      d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is ≤ 45% (751.5 MWt).

   b) The allowable bypass condition is to be revise based upon MNGP Power Rerate analysis. The specification is proposed to be changed to state:

      d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is < 45% (***798.75*** MWt).

3. Technical Specification Section 2.3, Bases, page 19. Technical Specification Section 3.1, Bases, page 38.

a) Revise bases for section 2.3, paragraph 'E' and 'F', on page 19; and bases section 3.1, paragraph 5, on page 38 concerning the discussion of plant conditions for which the turbine control valve fast closure scram and turbine stop valve scram may be bypassed to reflect the plant conditions for 1775 MWt

b) The bases are modified to reflect that the bypass setpoint is based upon the turbine first stage pressure indicative of 30% thermal power which provides margin to conservatively account for the potential for 14% of thermal power delivered to the main condenser via the main steam bypass valves consistent with the analytical limit established in the transient analysis.

F. Shutdown Cooling Isolation

1. Technical Specification Section 3.2, <u>PROTECTIVE INSTRUMENTATION</u>, TABLE 3.2.1, "Instrumentation That Initiates Containment Isolation Functions," Function item 6, page 50.

   a) The specifications provides limiting conditions for operation for instrumentation that initiates primary containment isolation per MNGP Technical Specification 3.2.A, and specifies that the Residual Heat Removal (RHR) shutdown cooling supply isolation valves will receive an isolation signal based on reactor pressure with a trip setting of ≤ (less than or equal to) 75 psig at the pump suction.

   b) The specification is proposed to be changed such that TABLE 3.2.1, Function item 6, specifies a trip setting of ≤ (less than or equal to) 75 psig at ***the reactor steam dome***.

G. Low Pressure Core Cooling Pump Pressure/Automatic Depressurization Interlock

1. Technical Specification Section 3.2, <u>PROTECTION INSTRUMENTATION</u>, TABLE 3.2.2, "Instrumentation That Initiates Emergency Core Cooling Functions," Function item C.3, page 53.

   a) The specification provides limiting conditions for operation for instrumentation that initiates the Automatic Depressurization System (ADS) portion emergency core cooling systems per MNGP Technical Specification 3.2.A. Item C.3 specifies a limiting trip setting of less than or equal to 100 psig for the low pressure core cooling pumps discharge pressure permissive for ADS system actuation.

   b) The specification is proposed to be changed such that TABLE 3.2.2, Function item C.3, specifies a trip setting of ≥ (greater than or equal to) ***60 psig and*** ≤ (less than or equal to) ***150*** psig.

2. Technical Specification Section 3.2, Bases, page 65.

   a) Add information to clarify the bases for the Automatic Depressurization Low Pressure Core Cooling pumps discharge pressure interlock trip setting. The trip setting for the low pressure ECCS pump permissive for ADS is such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid a discharge pressure permissive when the pumps are not running.

H. Containment Cooling and Containment Spray

1. Technical Specification Section 3.5, CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS, Specification 3.5.C, Containment Spray/Cooling System, page 104.

   a) The specification states:

      C. Containment Spray/Cooling System

      1. Except as specified in 3.5.C.2, 3 and 4 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212° F. A containment/spray cooling subsystem consists of the following equipment powered from one division:

      2 RHR Service Water Pumps
      1 Heat Exchanger
      *2 RHR Pumps**
      Valves and piping necessary for:
      Torus Cooling
      Drywell Spray

      2. One RHR Service Water Pump may be inoperable for 30 days.

      3. One RHR Service Water Pump in each subsystem may be inoperable for 7 days.

      4. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.

      5. If the requirements of 3.5.C.1, 2, 3 and 4 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

      ** For allowed out of service times for the RHR pumps see Section 3.5.A.*

b) The specification is proposed to be changed to state:

C. Containment Spray/Cooling System

1. Except as specified in 3.5.C.2 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. A containment/spray cooling subsystem consists of the following equipment powered from one division:

   1 RHR Service Water Pump
   1 RHR Heat Exchanger
   1 RHR Pump*
   Valves and piping necessary for:
      Torus Cooling
      Drywell Spray

2. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.

3. If the requirements of 3.5.C.1 or 2 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

* For allowed out of service times for the RHR pumps see Section 3.5.A.

I. Editorial Changes

1. Technical Specifications, Table of Contents, Page ii.

   a) The subject headings for subsections A, B and C of Section 3.4 and 4.4, Standby Liquid Control System, are to be revised to reflect the headings in the body of the specifications. Heading A is to be changed from "Normal Operation" to "System Operation." Heading B is to be changed from "Operation with Inoperable Components" to " Boron Solution Requirements." Heading C, which states "Volume Concentration Requirements" is to changed such that the phrase "Volume Concentration Requirements" is deleted as subsection C has no heading in the body of the specifications.

   b) The subject heading for subsection A of Section 3.5 and 4.5, Core and Containment/Spray Cooling Systems is to be revised to reflect the heading in the body of the specifications. Heading A is to be changed from "ECCS" to "ECCS Systems."

c) Page listings for various subsections are to be revised to reflect actual page numbering in the body of the specifications. The page numbers listed for 3.5/4.5 Bases is to be changed from 109 to 110; 3.6 and 4.6 Bases is to be changed from 144 to 145; subsection F, Deleted, of section 3.6 and 4.6, Primary System Boundary, is to be added as page 128; and subsection E, Combustible Gas Control System, of section 3.7 and 4.7, Containment Systems, is to be changed from 171a to 172.

2. Technical Specifications, Table of Contents, Page iii.

   a) Page listings for various subsections are to be revised to reflect actual page numbering in the body of the specifications. The page numbers listed for subsection 4, Station Battery System, of Section 3.9 and 4.9, Auxiliary Electrical Systems, is to be changed from 202 to 203 and 4.11, Bases, is to be changed from 217 to 218.

   b) Subsection 5, 24V Battery Systems, is added under Section 3.9 and 4.9, Auxiliary Electrical Systems, to reflect the body of the specifications.

3. Technical Specification Section 3.5/4.5, Section A, ECCS System, Bases, page 112.

   a) Revise the last sentence of the Section A, ECCS Systems, bases discussion found at the top of page 112 concerning the allowed out-of-service time for the selected safety/relief valves which form part of the Automatic Depressurization System (ADS). The bases currently states an incorrect allowed out-of-service time of seven (7) days. The bases is proposed to be revised to be consistent with MNGP specification 3.5.A.3.h which establishes an allowed out-of-service time of fourteen (14) days.

4. Technical Specification Section 3.5/4.5, Section B, RHR Intertie Line, Bases, page 112.

   a) The basis discussion for the RHR intertie line is revised to clarify the use and purpose of the intertie line.

5. Section 3.1 Bases, page 39.

   a) The bases were revised to include a statement on the application of GE setpoint methodology. See NSP's response to Question 36 contained in NSP's rerate RAI response letter dated September 5, 1997.

6. Section 3.2 Bases page 68.

   a) Delete first two sentences in the last paragraph.

J. Changes to Reactor Water Level Instrument Setpoints.

1. Technical Specifications, Table 3.1.1 Reactor Protection System (Scram) Instrument Requirements, page 28

    a) Item 7. Limiting Trip Setting. Delete reference to note 6. Place "annulus" in parentheses after 7 in.

    b) Page 30. Delete note 6.

2. Technical Specifications, Table 3.2.1 Instrumentation That Initiates Primary Containment Isolation Functions

    a) Page 49. Item 2. Trip Setting. Change to ≥ 7" (annulus).

    b) Page 50. Item 3. Trip Setting. Change to ≥ 7" (annulus).

3. Technical Specifications, Bases Section 3.2, Page 64.

    a) Change paragraph 4 to include discussion of level change due to higher pressure drop across the dryer/separator at rerate conditions. Delete reference to 10'6" above the top of active fuel.

III. Safety Assessment of the Proposed Change

A. Changes to the Facility Operating License and Technical Specifications to Reflect the Proposed MNGP Rerate Power Level

Operating License DPR-22, Docket No. 50-263, page 3, paragraph C.1.

Technical Specification, Section 1.0, DEFINITIONS, paragraph 1.R, page 4.

Technical Specification, Section 1.0, DEFINITIONS, paragraph 1.S, page 4.

Technical Specification, Section 2.3, Bases, pages 14 and 15.

Changes are proposed to increase the maximum reactor power level to 1775 MWt.

The safety analysis prepared by GE and NSP to support this amendment request and the implementation of the MNGP Power Rerate program is provided in Exhibit E. The evaluation demonstrates that MNGP can operate safely with the proposed increase of the maximum reactor power to 1775 MWt, with an increase of approximately 7% in main turbine inlet steam flow and the required increases of the flow, temperature, pressure, and capacity in supporting systems and components. Exhibit E provides a summary of the detailed evaluations performed by GE and NSP in support of the MNGP Power Rerate program. These detailed evaluations were performed, and the results are presented, in accordance with the guidelines in GE Topical Report NEDC-32424P, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1995. By letter to the General Electric Company (GE) dated February 8, 1996, this topical report was accepted by the NRC staff. Resolution of generic issues associated with the power rerate are addressed in GE Topical Report, NEDC-32523P "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" March 1996 and NEDC-32523P, Supplement 1, June 1996. These topical reports are currently under review by the NRC staff.

MNGP was originally licensed at 1670 MWt. The accident evaluations, as well as a majority of the plant specific evaluations performed in support of the MNGP Power Rerate have been performed assuming a reactor power of 1880 MWt. This power level represents a bounding analytical limit which is approximately 112.6% of the existing licensed limit of 1670 MWt, and also approximately 6% above the thermal power level of 1775 MWt proposed by MNGP Power Rerate. The safety analysis of design basis accidents are based on a power level 102% of 1880 MWt, unless the 2% power factor is already accounted for in the analysis methods. For analyses performed for a thermal power of 1880 MWt, the analyses demonstrated operating margin to criteria establishing margins of safety, thus additional operating margin is demonstrated and assured for the proposed power rerate to 1775 MWt.

The evaluation of transient events, as well as the evaluation of plant instrumentation setpoints, has been performed assuming a plant steady state power level of 1775 MWt. This approach was taken to provide an evaluation of cycle specific limits and plant operational setpoints consistent with the plant conditions proposed by MNGP Power Rerate. The evaluation of operational transients are based on a power level of 102% of 1775 MWt as required by the NRC approved analysis methods specified in the MNGP Technical Specifications. The evaluation demonstrated operating margin to criteria establishing margins of safety for the proposed power rerate to 1775 MWt.

The MNGP Power Rerate evaluation basis assures that the power dependent margin prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the code design rules will be maintained, as will other margin-assuring acceptance criteria used to determine the acceptability of the plant. Exhibit E, NEDC-32546P, provides detailed information on the evaluations performed, the specific steady state power level assumed in the evaluations, and the results of the MNGP Power Rerate evaluations. Exhibit G, NEDC-32514P, reports the results of the MNGP specific analysis performed by GE using the SAFER/GESTR-LOCA methodology which demonstrates conformance with the Emergency Core Cooling System (ECCS) acceptance criteria of 10CFR50.46.

The MNGP Power Rerate evaluations have been performed taking into account the implementation of the following previously-approved analyses.

Maximum Extended Load Line Limit Analysis (MELLLA) and Increased Core Flow (ICF), implemented by License Amendment 84, issued January 27, 1993,

Average Power Range Monitor-Rod Block Monitor Technical Specifications (ARTS) Technical Specification improvement program, implemented by License Amendment 29, issued November 16, 1984, and

Single Loop Operation, implemented by License Amendment 47, issued October 22, 1986.

Increasing the licensed maximum thermal power level of MNGP to 1775 MWt can be accomplished safely. This safety evaluation summarizes the information provided in Exhibit E, NEDC-32546P (i.e., the safety significant plant reactions to events analyzed for licensing the plant and potential effects on various margins of safety).

1. Fuel Thermal Limits

No change is required in the basic fuel design to achieve the rerate power levels or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for the power rerate. The current fuel operating limits, such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR), will not change due to the rerate power level. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria, to establish cycle specific operating limits, and to determine compliance with the cycle specific Safety Limit Minimum Critical Power Ratio. Analyses for each fuel reload will continue to demonstrate satisfaction of the criteria accepted by the NRC as specified in the MNGP Technical Specifications.

2. **Makeup Water Sources**

The Boiling Water Reactor design concept includes a variety of methods to pump water into the reactor vessel to deal with all types of events. The safety-related cooling water sources alone will maintain core integrity by providing adequate water for core cooling. In addition, there are non-safety related sources of cooling water. These safety and non-safety related sources are diverse in that they provide high and low pressure, high and low volume, means of delivering water to the vessel. These means include the feedwater/condensate pumps, the service water system, the Residual Heat Removal Service Water (RHRSW) system, the Low Pressure Coolant Injection system (LPCI) system, the Core Spray (CS) system, the High Pressure Coolant Injection (HPCI) system, the Reactor Core Isolation Cooling (RCIC) system, the Standby Liquid Control (SLC) pumps, the Control Rod Drive (CRD) pumps, and the Fire Protection pumps. Many of these diverse water supplies are redundant in equipment and also redundant in systems (e.g., there are several LPCI/CS pumps and complete redundant piping systems).

MNGP Power Rerate does not result in an increase or decrease in the available water sources or their capability to provide cooling water, nor does it change the selection of those assumed to function in the safety analyses. NRC-approved methods were used for analyzing the performance of the Emergency Core Cooling Systems (ECCS) during loss-of-coolant-accidents.

MNGP Power Rerate results in an increase in decay heat proportional to the core power increase. A design requirement for the Residual Heat Removal system is that the system be capable of cooling the reactor water to 125°F in 24 hours with allowance for using both Residual Heat Removal heat exchangers. For the worst case conditions, the time to reach 125°F is increased. However, the existing cooling capacity remains adequate to satisfy this design requirement for a bounding reactor power of 1880 MWt. MNGP has committed to provide an alternate shutdown cooling path consistent with the guidance of Draft Regulatory Guide 1.139. An alternate shutdown cooling analysis was performed for a bounding reactor power of 1880 MWt. The results of this analysis show that the power rerate has no significant effect on the alternate shutdown cooling analysis. The licensing basis is specified in Updated Safety Analysis Report (USAR) Sections 6.2.3.3.4 and 10.2.4.

3. Design Basis Accidents

Design basis accidents are very low probability postulated events whose characteristics and consequences are used in the design of the plant so that the plant systems can be designed to mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of postulated pipe break sizes in the largest recirculation, steam, and feedwater lines and a postulated break in one of the ECCS lines. This break range bounds the full spectrum of large and small, high and low, energy line breaks. The success of the plant systems in dealing with the range of postulated pipe breaks up to the bounding postulated Loss of Coolant Accident (LOCA), while accommodating a single active equipment failure in addition to the postulated LOCA, has been assessed and demonstrated. This assessment included the following.

Challenges to Fuel or ECCS Performance Analyses in accordance with the rules and criteria of 10CFR50.46 and Appendix K wherein the predominate figure of merit is the fuel Peak Clad Temperature (PCT).

Challenges to the Containment wherein the primary figures of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression pool temperature for long term cooling.

Design basis accident radiological consequences calculated and compared to the criteria of 10CFR100.

a) Design Basis Accident Challenges to Fuel

The Emergency Core Cooling System (ECCS) is described in Section 6.2 of the plant Updated Safety Analysis Report. The MNGP Power Rerate ECCS performance evaluation was conducted by application of the 10CFR50, Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. As mentioned above, a complete spectrum of pipe breaks is investigated. As shown in Table 4-2 of Exhibit E, the safety margins established by 10CFR50.46 are maintained for a bounding reactor power increase to 1880 MWt. Therefore, the ECCS safety margins are not affected by MNGP Power Rerate. In addition, the ECCS performance evaluation showed that no change in the Maximum Average Planer Heat Generation Rate (MAPLHGR) or Linear Heat Generation Rate (LHGR) limits are required for the power rerate. Exhibit G, reports the results of the MNGP specific analysis performed by GE which demonstrates conformance with the Emergency Core Cooling System (ECCS) acceptance criteria of 10CFR50.46.

b) Design Basis Accident Challenges to the Containment

Table 4-1 of Exhibit E provides the results of analyses of the plant containment response to the most severe LOCA for a bounding reactor power of 1880 MWt. The effect of the power rerate on the short term containment response (peak pressure) as well as the long term containment response for containment temperature confirms the adequacy of the plant primary containment for a bounding reactor power of 1880 MWt. An analytical power level of 1880 MWt bounds the decay heat associated with the 1775 MWt power level with a one sided confidence interval of 95%. See NSP's response to Question 51 of the staff's rerate RAI dated September 5, 1997.

Short-term containment response analyses were performed for the limiting design basis LOCA consisting of a double-ended guillotine break of a recirculation suction line to demonstrate that operation at a bounding reactor power of 1880 MWt will not result in exceeding the containment design limits. This limiting design basis LOCA event results in the highest short-term containment pressures and dynamic loads. The analysis determined that for a bounding reactor power of 1880 MWt, the maximum drywell pressure value is bounded by the current USAR analysis value and by the containment design pressure. The power rerate to 1775 MWt has no adverse effect on the containment structural design pressure.

The long-term bulk suppression pool temperature response was evaluated for the limiting design basis LOCA. The peak suppression pool temperature increases 10°F to a value of approximately 194°F for the design basis LOCA. Thus peak suppression pool temperature remains within the 281°F suppression chamber design temperature and the torus attached piping analysis temperature of 195°F. Analysis confirmed that ECCS pump NPSH is not adversely affected with this temperature response. The power rerate has no adverse effect on the long term suppression pool temperature response to the design basis LOCA. Local suppression pool temperature limits for SRV discharge were analyzed . The analysis showed that the local pool temperature limit is not exceeded for bounding reactor power increase to 1880 MWt.

The drywell temperature response was analyzed for a series of small and intermediate sized steam line breaks. Steam line breaks impose high drywell gas temperatures for relatively long time periods. The peak drywell gas temperature for the steamline break was 331°F for the bounding 1880 MWt power conditions. All drywell equipment required to be operable in accordance with 10CFR50.49 has been qualified to at least 335°F, thus the power rerate has no adverse effect on drywell equipment environmental qualification. Because the peak drywell gas temperature exceeded the drywell shell design temperature of 281°F, additional analyses were performed to determine the drywell shell temperature. The results of these analyses showed that the peak drywell shell temperature for the bounding 1880 MWt power conditions was 273°F. This temperature is within the design temperature of 281°F and factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt. Analysis of the drywell gas temperature response for a bounding reactor power of 1880 MWt has confirmed no adverse effect on the containment structure, thus MNGP Power Rerate is acceptable

For a bounding reactor power of 1880 MWt, the effect of a reactor power increase on the conditions which affect the containment dynamic loads have been determined and judged satisfactory. Where plant conditions for the bounding reactor power of 1880 MWt are within the range of conditions used to define the original dynamic loads, current safety criteria are met and no further structural analysis was performed. The containment dynamic loads were found to be acceptable and bounding for the power rerate to 1775 MWt.

c) Design Basis Accident Radiological Consequences:

The USAR provides the radiological consequences for each of the design basis accidents. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. For power rerate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor which will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of the Control Rod Drop Accident, Loss of Coolant Accident, the Main Steam Line Break Accident and the Refueling Accident were evaluated for initial licensing of MNGP using the General Electric analysis method as described in APED 5756, "Analytical Methods for Evaluating the Radiological Aspects of the GE BWR," March 1969. This evaluation demonstrated that the radiological consequences of the design basis accidents are well within the criteria of 10CFR100. The results of this analysis are provided in Sections 14.7.1.6, 14.7.2.4, 14.7.3.3, and 14.7.6.4.2 of the USAR for the Control Rod Drop Accident, Loss of Coolant Accident, the Main Steam Line Break Accident and the Refueling Accident, respectively.

In addition to the design basis radiological analysis that was performed for initial licensing, an evaluation was performed to establish a conservative comparison between the design basis radiological analysis and the analysis inputs suggested in AEC Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power And Test Reactor Sites." Section 14.7.7 of the USAR describes the comparative evaluation that was performed. This comparative evaluation established dose multipliers which, if multiplied by the appropriate doses presented in Section 14.7.1.6, 14.7.2.4, 14.7.3.3, and 14.7.6.4.2 of the USAR, yields doses equivalent to those obtained by an analysis using the TID

14844 inputs provided in USAR Section 14.7. This dose sensitivity analysis of the design basis evaluation concluded that the radiological consequences of the design basis accidents are well within the criteria of 10CFR100 when the differing analysis inputs contained in TID 14844 were accounted for.

The Atomic Energy Commission (AEC), predecessor to the NRC, performed an independent evaluation of the radiological consequences of the MNGP design basis accidents. The AEC summarized the results of this independent evaluation in the Safety Evaluation issued March 18, 1970 in support of issuance of the MNGP provisional operating license. The AEC concluded that the radiological doses that could result from any design basis accidents are well within the guideline values given in 10CFR100. The AEC evaluation results are summarized in the following table.

Calculated Potential Offsite Doses (Rem)
From Design Basis Accidents

	Two-Hour Dose At Site Boundary (0.3 Mile)		Course of Accident Dose At Low Population Zone (1 Mile)	
Accident	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of Coolant	82	3	150	3
Refueling	13	0.4	5	0.2
Control Rod Drop	5	0.1	11	0.3
Steamline Break	30	Negligible	10	Negligible

For MNGP Power Rerate, the radiological consequences of the limiting design basis accidents were re-evaluated. This evaluation was performed using inputs and evaluation techniques consistent with the current regulatory guidance, the current GE analysis methods, and relevant portions of the plant design basis. The inputs used in the MNGP Power Rerate evaluation provide a conservative assessment of the potential radiological consequences. The inputs and evaluation methods used for MNGP Power Rerate differ from those used in the current licensing basis evaluation presented in the USAR and the AEC safety evaluation, as the MNGP Power Rerate evaluations use the more contemporary NRC staff approved methods. However, the conclusions of these evaluations are consistent with the original licensing basis evaluations. The radiological consequences of the limiting design basis accidents remain well within 10CFR100 guidelines for a bounding reactor power of 1880 MWt. In addition, the radiological consequences of the limiting design basis accidents remain within those established in the AEC safety evaluation, which established the basis for initial licensing of MNGP. The results from the MNGP Power Rerate evaluation of the potential radiological consequences of the limiting design basis accidents are summarized in Section 9.2 of Exhibit E.

The Main Steam Line Break Accident (MSLBA) outside containment was not specifically analyzed for the power rerate because the releases are bounded by those for the hot standby condition. The radiological release for the steam line break is 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 initial reactor pressure and the steamline flow restrictor area, both of which are unchanged for the power rerate. Therefore, the flow through the break is not affected by the power rerate. The initial core power determines the amount of steam generation, 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 constant, the delayed level swell results in less steam-water mixture being released out the break. Therefore, the limiting radiological consequences of a steamline break are not affected by the power rerate.

The radiological analyses reflect an improvement for the control room emergency filtration system filter efficiency, a conservative reduction for the standby gas treatment system filter efficiency and reduced control room ventilation bypass leakage. In addition, the transport of non-organic iodine was modeled with methods which differ from the analysis reflected in the USAR. Non-organic iodine transported through the steam lines and condenser is subject to plate-out and re-suspension inside the pipes and the condenser. The BWROG methodology for evaluating MSIV leakages and condenser releases was used in the radiological release analyses. A modification to the control room emergency filtration system is to be performed prior to implementation of the power rerate to establish the improved control room ventilation bypass leakage. By letter dated July 26, 1996, with subject, Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability, proposed changes to the Technical Specifications were submitted to establish revised requirements consistent with improved control room emergency filtration system filter efficiencies as well as a revised analysis for the main steam line break at a hot standby condition.

The MNGP Power Rerate accident dose analyses, using inputs consistent with the current guidance and methods, were performed for the current rated power level (1670 MWt) and a bounding reactor power of 1880 MWt as described in Section 9.2 of Exhibit E. Thus a representative comparison between the design basis accident radiological consequences prior to implementation of the power rerate and subsequent to the power rerate is provided. A comparison of the 1670 MWt and

1880 MWt evaluation results demonstrates that the radiological consequences increase approximately in proportion to the increase in power. This evaluation demonstrated acceptable results for a bounding power increase to 1880 MWt, thus the radiological consequences of design basis accidents are well within the guidelines of 10CFR100 for the proposed power rerate to 1775 MWt.

It is therefore concluded that the radiological consequences of an accident subsequent to implementation of MNGP Power Rerate are slightly increased approximately proportional to the increase in power and that these consequences remain well below regulatory guidelines. The evaluation for MNGP Power Rerate was performed at 102% of 1880 MWt where applicable. The results remain below the 10CFR100 guideline values as well as the licensing basis established in the March 18, 1970 AEC safety evaluation. Therefore, the postulated radiological consequences are clearly within the regulatory guidelines and all radiological safety margins are maintained for the power rerate to 1775 MWt.

4. Transient Analyses

The effects of plant transients were evaluated at the rerate power level of 1775 MWt. The transient events were evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The SLMCPR is determined using NRC-approved methods. The Power Rerate transient analyses were performed using the approved methodology specified in the plant Technical Specifications. The limiting transient events are slightly more severe when initiated from the rerate power level. The power rerate transient evaluation results show a slightly more limiting event initial CPR (≤0.02) than that initiated from the present rated power level for the near limiting transients. However, for the most limiting transient, the evaluation of a representative core showed that no change is required to the Operating Limit MCPR for the power rerate and that the integrity of the SLMCPR is maintained. Table 9-3 of Exhibit E provides the evaluation results for the limiting and near limiting transients. The margin of safety established by the SLMCPR is not affected by the power rerate to 1775 MWt. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The fuel thermal-mechanical limits at the power rerate conditions are within the specific design criteria for the GE fuels currently loaded in the MNGP core. Also, the power-dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) methods developed as part of the core performance improvement program remain applicable to rerate conditions. The transient event evaluation confirmed that MNGP Power Rerate has no significant effect on the power-dependent and flow-dependent MCPR and MAPLHGR limits.

5. Evaluation Conservatism

Design Basis Accidents and transients are postulated at the power rerate conditions to evaluate challenges to the fuel, containment, and offsite dose limits. These challenges have been evaluated in accordance with conservative regulatory procedures such that the evaluation results are more severe than the expected effects from the postulated accident. Rerate analyses use fuel designed to current NRC-approved criteria. The reactor is operated within limits established using NRC-approved methodologies to produce more heat and thus increased steam flow to the turbine. Accepted design criteria are used to assure equipment mechanical performance at rerate conditions. The potential for transient events is not increased due to the power rerate. Operating margin to instrumentation setpoints have been evaluated using approved methodology. Appropriate changes are to be implemented prior to MNGP Power Rerate implementation to maintain operating margin when evaluated as necessary. The offsite dose evaluation performed using the current guidance and methods provides a more severe design basis accident radiological consequences scenario than the effects of the postulated Loss of Coolant Accident which produces the greatest challenge to the fuel and/or containment. That is, the design basis accident which produces the highest fuel Peak Clad Temperature (PCT) and/or containment pressure does not damage large amounts of fuel and, thus, the source terms and doses are much smaller than those postulated in the evaluation of radiological consequences.

A comprehensive assessment of the effect of MNGP Power Rerate on plant risk has been obtained by performance of a sensitivity study on the plant probabilistic risk assessment (PRA). This included the effect of the power rerate on severe accidents and other external events. The analysis concluded that no new vulnerabilities to severe accidents were created due to power rerate and that only an insignificant increase in core damage frequency occurs due to changes in event timing, with a resulting minor effect on time available for human actions following some events. Thus, the analysis provided confirmation of no significant increase in the probability of previously evaluated accidents.

6. Non-LOCA Radiological Release Accidents

All of the other radiological releases discussed in the USAR are either unchanged because they are not power-dependent, or increase approximately in linear proportion to the amount of the rerate. The dose consequences for all of the non-LOCA radiological release accident events are bounded by the "Design Basis Radiological Consequences" events discussed above.

7. Equipment Qualification

Safety related electrical equipment within the scope of 10CFR50.49 was evaluated to assure qualification for the normal and accident conditions associated with a power rerate to 1775 MWt. Applicable conservatisms were applied to the environmental parameters as required.

The current normal conditions for temperature, pressure, and humidity are unchanged for the power rerate conditions. Some HELB profiles were found to increase an insignificant amount. The radiation levels under normal and accident conditions were conservatively evaluated to increase approximately 6.3% with a power rerate to 1775 MWt. The environmental qualification for equipment with the scope of 10CFR50.49 was found acceptable for the power rerate to 1775 MWt. For a limited scope of environmentally qualified equipment, maintenance interval changes are to be implemented to address the power rerate effects on service life.

Plant equipment and instrumentation has been evaluated against the criteria appropriate for rerate. Due to the very limited number of systems with only minor increases in system temperatures, pressures, or power requirements with MNGP Power Rerate, it was found that equipment qualification was satisfactory for the power rerate conditions. See NSP's response Questions 1 though 5 to the Staff's rerate RAI dated September 5, 1997 for additional information.

8. Balance-of-Plant (BOP)

BOP systems/equipment used to perform safety-related and normal operation functions have been reviewed for rerate in a manner comparable to that for safety-related NSSS systems/equipment. This includes, but is not necessarily limited to, all or portions of the Main Steam, Feedwater, Turbine, Condenser, Condensate, Service Water, Emergency Service Water, Emergency Diesel Generator, BOP Piping, and Support Systems. Significant groups/types of BOP equipment/systems are justified for rerate by generic evaluations. Plant-unique evaluations justify the power rerate operation for BOP systems/equipment that are not generically justified.

9. Auxiliary Power System

The MNGP auxiliary electrical system is discussed in Sections 8.2, 8.3 and 8.4 of the USAR. The MNGP auxiliary power system can be supplied by any of the three separate offsite power transformers (2R, 1R and 1AR) or directly from the Emergency Diesel Generators to supply equipment required for safe plant shutdown, to maintain a safe shutdown condition or operate required safeguards equipment following an accident. The 1AR transformer or the Emergency Diesel Generators are redundant AC sources for safety related plant loads and do not normally supply power generation loads. The current plant Technical Specifications and bases concerning offsite electrical power sources were established in the plant Technical Specifications via Amendment 51 to the Facility Operating License, issued October 16, 1987.

The power rerate does not result in any increased loading of the electrical buses supplying the engineered safeguards equipment. At the power rerate conditions, the condensate and feedwater pump motor loading on the 1R or 2R transformers will increase. A modification to the condensate pumps not related to power rerate will increase pump efficiency. This will reduce horsepower requirements and is expected to result in no significant increase in condensate pump motor loads at rerate conditions. An evaluation of the auxiliary power system for the power rerate conditions confirmed that the system has sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the required engineered safeguards equipment following postulated accidents. Because of load shedding, rerate does not affect the loading or operation of the 1AR transformer. The 2R transformer has a very large load margin and is more than adequate to supply power rerate loads. A detailed analysis of the effect of power rerate on the 1R transformer is provided in Exhibit I.

10. Instrumentation

The control and instrumentation signal ranges and analytical limits for setpoints were evaluated to establish the effects of the changes in various process parameters such as power, system pressure, neutron flux, and feedwater flow. As required, analyses were performed to determine the need for setpoint changes for various functions (e.g., APRM neutron flux scram setpoints). In general, setpoints are to be changed only to maintain adequate difference between plant operating parameters and trip setpoints, while ensuring satisfactory safety performance is demonstrated. The revised setpoints have been established using the GE setpoint methodology provided in NEDC-31336, "General Electric Instrument Setpoint Methodology," as guidance.

11. Licensing Evaluation

The applicable plant licensing commitments, Bulletins, Circulars, Notices, etc. were evaluated for the effects of the power rerate. No adverse effect on MNGP licensing commitments due to MNGP Power Rerate were identified. Other special events and features such as Environmental Qualification (EQ) program, Station Blackout, Fire Protection, and Motor Operated Valves (MOVs) have been evaluated to assure safe plant operation for rerate conditions.

By letter dated September 5, 1997, NSP stated that it would respond to part 2 of Question 23 (MOVs) of the staff's rerate RAI at a later date. An evaluation of the Generic Letter 89-10 MOV requirements was conducted. This evaluation included ambient and system operating conditions associated with power rerate. The torque switch for one motor operator must be adjusted, see Exhibit D for a description of this task. The MOV evaluation demonstrated that all safety-related MOVs will be capable of performing their intended functions at rerate conditions.

12. Conclusion

The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

B. APRM Neutron Flux Scram, APRM Rod Block Monitor and Single Loop Operation

Technical Specification Section 2.0, SAFETY LIMITS, subsection 2.3, FUEL CLADDING INTEGRITY , Specifications 2.3.A.1.a, and 2.3.A.1.b, page 6.

Technical Specification Section 3.2, PROTECTION INSTRUMENTATION, TABLE 3.2.3, "Instrumentation That Initiates Rod Block," Function item 3, page 56.

Technical Specification Section 2.3, Bases, Page 16. Technical Specification Section 3.5/4.5, Bases, Paragraph F, Recirculation System, page 114.

Changes are proposed to the plant Technical Specifications and Bases to modify the limiting conditions for operation for the APRM flow biased neutron flux scram and APRM flow biased rod block functions for two loop and single loop operation.

The current Technical Specification limiting trip settings for single loop operation reflect those for a facility which has not implemented Maximum Extended Load Line Limit (MELLL). While these trip settings are conservative, single loop operations requires gain adjustments to the APRMs which places a burden on plant staff and complicates the transition to single loop operation when such operation is required. In conjunction with evaluations performed to support implementation of MNGP Power Rerate, evaluations have been performed which demonstrates that the single loop APRM flow biased neutron flux scram and rod block trip settings can be revised to be consistent with MELLL power operations and eliminate the APRM gain adjustment operation.

Technical Specifications were established to allow operation in a single loop condition by Amendment 47 to the MNGP Technical Specifications, issued on October 22, 1986. As reflected in the staff Safety Evaluation Report supporting Amendment 47, the APRM flow biased neutron flux scram and rod block trip settings were decreased to account for back flow through the inactive jet pumps by application of a reverse flow correction for single loop operation. This correction factor was established consistent with previously-approved single loop operation Technical Specification changes. Thus for single loop operation the following APRM flow biased trip settings were established.

APRM Neutron Flux Scram: $S \leq 0.58\ (W\text{-}dw) + 62\%$
APRM Rod Block: $S \leq 0.58\ (W\text{-}dw) + 50\%$

Where, S = Setting in percent of rated thermal power,
W = Recirculation drive flow in percent, and

dw = 0 for two loop operation and 5.4 for single loop operation.

Technical Specifications were established to allow operation with an expanded operating domain resulting from the Maximum Extended Load Line Limit Analysis (MELLLA) and Increased Core Flow (ICF) analysis by Amendment 84 to the MNGP Facility Operating License, issued January 27, 1993. As requested in NSP's amendment request and issued in Amendment 84, new two loop APRM flow biased neutron flux scram trip settings and rod block trip settings were established consistent with the MELLL and ICF analyses, the single loop trip settings were not changed at that time. The single loop trip settings remained conservative. Thus the following APRM flow biased trip settings were established by Amendment 84.

Two Loop APRM Neutron Flux Scram:	$S \leq 0.66\ W + 70\%$
Two Loop APRM Rod Block :	$S \leq 0.66\ W + 56\%$
Single Loop APRM Neutron Flux Scram:	$S \leq 0.58\ (W-5.4) + 62\%$
Single Loop APRM Rod Block :	$S \leq 0.58\ (W-5.4) + 50\%$

Where, S = Setting in percent of rated thermal power, and
W = Recirculation drive flow in percent.

For MNGP Power Rerate, the two loop and single loop operation APRM flow biased neutron flux scram and rod block limiting trip settings were reanalyzed. The MELLLA trip setpoints determined for two-loop operation were confirmed to be acceptable for single loop operation with a correction applied to account for the effective drive flow applied when operating in single loop.

The APRMs will be re-calibrated to reflect 1775 MWt as the 100% core thermal power due to MNGP Power Rerate. The MELLL region remains constant in terms of absolute core thermal power. Thus in terms of percent of thermal power, with the 1775 MWt rerate power equal to 100%, the MELLL region boundary points change to a lower percentage of 100% power with the same flow. Therefore, the sloped portions of the APRM flow biased rod block line and scram lines will shift downward and the setpoints will be changed accordingly as proposed in the requested changes for MNGP Technical Specification 2.3.A. These changes will maintain the same margin from the upper limit of the MELLL region of the power to flow map to assure the change is conservative (refer to Figure 2-1 and Section 5.1.2 of Exhibit E).

Transient events have been evaluated for MNGP Power Rerate. The limiting events for each limiting transient category were evaluated to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. No changes to the basic characteristics of any of the limiting events are caused by the power rerate as shown in Section 9.1 of Exhibit E. For pressurization, flow decrease, and cold water increase transients, the consequences of the transient are primarily dependent on power level. The maximum power output for operation of MNGP with one recirculation loop will remain bounded by the maximum power output for two loop operation, thus transient results for full power remain more limiting for both the thermal and overpressure consequences of single loop operation. The one pump seizure transient from a single loop condition was re-evaluated for the power rerate. This event is treated in the USAR as an accident, but was analyzed as a transient to conservatively evaluate the MCPR. The analysis concluded that the MCPR for the pump seizure event from a single loop condition remains greater than the fuel integrity safety limit; therefore, no fuel failures were postulated to occur as a result of the analyzed event. The transient evaluations credit the APRM high neutron flux scram for the mitigation of transient event consequences. The APRM flow biased neutron flux scram and the APRM flow biased rod block provide a redundant anticipatory protection feature for the protection demonstrated analytically for the APRM high neutron flux scram. Thus the proposed change to the Technical Specifications provides a conservative redundant scram feature.

NSP has selected the BWROG long-term stability solution Option 1-D for MNGP. This solution consists of an administratively controlled exclusion region and an analytical demonstration that, if an oscillation were to occur, (1) only core-wide mode oscillations would be expected (due to the small core/tight inlet orifice design) and (2) the flow biased APRM neutron flux trip would provide protection. The Option 1-D exclusion boundary is fuel cycle dependent and represents a line of constant stability margin. The boundary is core power and flow dependent and is computed using the approved licensing procedure. MNGP Power Rerate has been factored into the Option 1-D analysis so that the resulting exclusion region is representative of the power rerate conditions.

An evaluation was performed to determine the effect of the power rerate on core stability issues. The APRM flux trip instability protection feature of Option 1-D provides for a conservative calculation of the MCPR for an anticipated stability-related oscillation. Option 1-D relies on the APRM flow biased neutron flux scram for detection and suppression of thermal hydraulic instabilities as an added level of protection and conservatism to the cycle specific defined operating exclusion region. The nominal APRM trip setpoint is input as a percentage of rated power in the statistical calculation of the hot bundle oscillation magnitude in the evaluation of the MCPR for the analyzed bounding transient. For the worst case conditions with the transient initiated from natural circulation, detection and suppression of the stability related oscillation was credited based on the two loop APRM flow biased neutron flux scram settings. For this analysis the licensing criteria was satisfied in that the MCPR safety limit was not violated using inputs consistent with the proposed Technical Specification changes. In addition, this analysis provided a worst case bounding demonstration of acceptable CPR performance with respect to the single loop APRM flow biased neutron flux scram function. Adequate detection and suppression of potential thermal-hydraulic instabilities is provided by the changes proposed to the APRM flow biased neutron flux scram settings. To ensure that MNGP will maintain the same level of protection against the occurrence of a thermal-hydraulic instability, the instability exclusion region boundaries are maintained constant with respect to actual power level (MWt) by adjustment of the percent power on the revised power flow map for MNGP Power Rerate.

The proposed changes to the APRM flow biased neutron flux scram and rod block settings provide conservative protective settings to ensure that the fuel integrity safety limits and Reactor Primary Coolant Boundary pressure limits are not exceeded. The two loop settings have been conservatively established to maintain a level of protection under rerate power conditions consistent with the current Technical Specifications. The single loop settings have been conservatively established to be consistent with the two loop settings while ensuring the appropriate corrections are applied to account for single loop operation. Single loop operation has been demonstrated to be acceptable with the proposed changes. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

C. Safety Relief Valves

Technical Specification Section 2.2, Bases, page 23, Section 2.4 Bases, page 24, Section 3.6 and 4.6, Bases, paragraph E, page 150.

Bases revisions are submitted to reflect the analyzed actuation settings and complement of Safety Relief Valves. The "as-found" set pressure testing criterion as reflected in the Bases is to be changed from 2% to 3% consistent with the ANSI/ASME OM-1-1981 Code. The bases are to be revised to reflect that the MNGP Power Rerate evaluations assumed five of eight Safety Relief Valves to be operable for design basis accidents and transients. In addition, the bases is revised to reflect License Amendment 92, which reduced the test frequency from seven of the valves being tested every refueling outage to 20% of the valves being tested within any 24 months in accordance with ANSI/ASME OM-1-1981.

The Reactor Pressure Relief System consists of eight pilot operated safety/relief valves (SRVs) each equipped with a remote air actuator. The SRVs are located in the drywell and are mounted on the main steam lines (two per line) between the reactor vessel and the first main steam isolation valve. The discharge of each valve is piped to the suppression pool. The SRVs are designed to be self-actuating on overpressure or to be remotely operated with an air actuator. Solenoid valves are installed in the pneumatic supplies to the SRV actuators and are controlled manually from the main control room or automatically by the Automatic Depressurization System (ADS) or Low-Low Set System logic. Section 4.4 of the USAR provides additional information on design and function of the Reactor Pressure Relief System.

As discussed in the bases for the MNGP Technical Specifications and in USAR Section 14.5.1 and Section 14A, subsection 12.0, the SRVs assure that the reactor coolant system pressure safety limit is never reached during design basis accident and transient conditions. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the SRVs must be set to open at a pressure no higher than 105 percent of design pressure, with at least one SRV set to open at a pressure not greater than the design pressure, and the SRVs must limit the reactor pressure to no more than

110 percent of design pressure. The SRVs are sized according to the Code for a condition of Main Steam Isolation Valve (MSIV) closure while operating at 100% of rated thermal power, followed by a scram from indirect (high flux) means. The nominal setpoint for the safety function of the eight SRVs is 1109 psig with a one percent (1%) tolerance. With the SRVs set as specified, the maximum vessel pressure remains below the 1375 psig ASME Code limit.

The SRVs are tested in accordance with ANSI/ASME OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." The Code specifies an acceptance criterion of 3% of the stamped set pressure and specifies further conditions to be satisfied if the criterion is not met. The Code has been reviewed and approved by the ASME Code Committee for industry-wide use and has been incorporated by reference through the 1986 Edition of Section XI of the ASME Code in 10CFR50.55a, May 5, 1988 (Federal Register, Volume 53, page 16051).

The limiting design basis pressurization transient was evaluated for the 1775 MWt power rerate conditions using analysis inputs consistent with the those presented in the USAR. The evaluation assumed that only five of the eight safety relief valves are operable with valve actuation at 3% above the pressure setpoint of 1109 psig. At 1775 MWt conditions, a slightly higher Reactor Pressure Vessel peak pressure results due to the worst case over pressure transient (1287 psig), but the peak pressure remains below the 1375 psig ASME Code limit. Therefore, there is no decrease in the safety margin established by the ASME Code. The results of the rerate over pressure protection analysis are given in Figure 3-1 of Exhibit E. The proposed change to the Technical Specifications bases concerning the number of safety relief valves assumed available and the valve setpoint tolerance is consistent with the current USAR evaluation of the design basis MSIV closure event and is further supported and demonstrated to be acceptable based on the MNGP Power Rerate evaluations.

The High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system performance has been evaluated to confirm the acceptability of the safety relief valve "as-found" setpoint tolerance change. The conservative evaluation demonstrated that the potentially higher allowed system pressure has no adverse effect on HPCI system or RCIC system performance. The maximum operating pressure for the HPCI and RCIC systems is limited by the safety related single-failure proof SRV Low-Low Set system. See NSP's response to Question 23 of the staff's rerate RAI dated September 5, 1997. The change in the SRV as-found tolerance will have no effect on the ADS or Low-Low Set System functions or operation of the SRVs. The change has no effect on the low pressure portion of the ECCS or the Standby Liquid Control (SBLC) system. The MNGP Power Rerate analyses has demonstrated the acceptability of the proposed changes. Therefore, the change will have no effect on any design basis accident or transient analysis. The health and safety of the public will not be affected by this change. The changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

D. Condenser Low Vacuum Scram

Technical Specification Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS, Trip Function item 9, page 28.

Technical Specification Section 3.1, Bases, page 37.

Changes are proposed to the plant Technical Specifications and Bases to modify the trip setting for the condenser low vacuum scram function.

The condenser low vacuum scram function is an anticipatory scram and is not considered in the plant safety analysis. The function of this scram is to lessen the severity of a loss of condenser vacuum transient and to protect the condenser from an overpressure event. The bases discussion for Section 3.1 states that the condenser low vacuum scram function is intended as a back-up to the turbine stop valve closure scram. For a decreasing condenser vacuum transient, the plant instrumentation provides for Turbine Stop Valve (TSV) closure at 20" Hg, resulting in an associated turbine trip and a reactor scram signal from TSV position, with main steam bypass valve closure at 7" Hg. Thus, a complete loss of vacuum in the worst case is a turbine trip without bypass. The turbine trip without bypass transient is described further in section 14.4.5 of the USAR. The condenser low vacuum scram minimizes the resulting pressure transient and neutron flux transient associated with the turbine trip due to decreasing vacuum. The severity of the transient and the potential for a Safety Relief Valve lift resulting from the transient is thus minimized.

With MNGP Power Rerate the turbine trip without bypass transient remains bounding for a loss of condenser vacuum transient. The MNGP Power Rerate transient analysis has confirmed that fuel thermal-hydraulic limits and reactor coolant pressure boundary limits are not exceeded for the bounding turbine trip without bypass transient. The safety analysis credits the TSV closure scram for mitigating the consequences of the turbine trip without bypass transient.

The condenser vacuum is expected to decrease slightly during normal operation as a result of the power rerate. In order to maintain adequate operating margin, the Limiting Trip Setting for the condenser low vacuum scram setpoint in the plant Technical Specifications is proposed to be changed. The present Technical Specification value of $\geq$ 23 inches mercury is proposed to be changed to $\geq$ 22 inches mercury. Since the proposed change maintains the setpoint greater than the TSV closure at 20" Hg, the anticipatory function of the condenser low vacuum scram is maintained for the turbine trip without bypass transient. This revised setpoint has been established using the GE setpoint methodology as provided in NEDC-31336, "General Electric Instrument Setpoint Methodology," as guidance. As discussed above, the condenser low vacuum scram is not credited in the safety analyses. The proposed change in the setpoint will maintain the anticipatory scram function. Thus the proposed change will have no effect on safety and the current bases for the setpoint is maintained, while maintaining operating margin. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or a significant reduction in the margin of safety.

E. Turbine Control Valve Fast Closure Scram and Turbine Stop Valve Scram

Technical Specification Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS, Required Conditions when minimum conditions for operation are not satisfied, item D, page 30.

Technical Specification Section 3.1, REACTOR PROTECTION SYSTEM, TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS," Allowable Bypass Conditions, item d, page 30.

Technical Specification Section 2.3, Bases, page 19. MNGP Technical Specifications, Section 3.1, Bases, page 38.

Changes are proposed to the plant Technical Specifications to modify the limiting conditions for operation for the Turbine Control Valve (TCV) fast closure and Turbine Stop Valve (TSV) closure scram bypass function. Changes are submitted for the bases discussion of the scram bypass function to provide clarification and reflect the results of the power rerate evaluation.

A main generator load rejection or a turbine trip initiates a Turbine Control Valve (TCV) fast closure or a Turbine Stop Valve (TSV) closure to prevent damage to the turbine. To mitigate the ensuing reactor pressurization transient, an immediate reactor scram is provided based on indication of TCV fast closure or TSV closure. At low thermal power levels, the margins to fuel thermal-hydraulic limits and to the Reactor Primary Coolant Boundary pressure limits are large and the immediate scram is not necessary. Therefore, an automatic, low power bypass of these scrams is provided. This bypass is controlled by the turbine first stage pressure and bypasses the TCV fast closure and TSV closure scram when the turbine first stage pressure is below the bypass setpoint.

The function of this signal is to activate the turbine valve closure scram signals above a predetermined reactor thermal power level. The analytical basis for activation of the valve closure scrams is greater than 45% of thermal power. This analytical basis has been maintained for MNGP Power Rerate. The bases for this specification are to be clarified to properly reflect that the valve scrams are enabled at a turbine 1st stage pressure less than a pressure indicative of a thermal power of 30%. Ensuring the valve scrams are enabled prior to 30% thermal power ensures the analytical limit of 45% of rated core thermal power is satisfied with consideration for the capability of the main steam bypass valves to pass approximately 14% of rerate thermal power. This clarification is needed irrespective of power rerate. The present setting is correctly set according to the turbine 1st stage pressure equivalent of thermal power.

The scram bypass point , P_{BYPASS}, is an analytical limit in the transient analysis at MNGP. In the evaluation of the ARTS program, it was assumed that P_{BYPASS} was set at 45% thermal power in the derivation of the power dependent Critical Power Ratio (CPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits. See Part 16.1, Section 14A of the USAR. The transient events for MNGP Power Rerate were evaluated assuming an analytical limit (P_{BYPASS}) for bypass of this trip function equivalent to 45% of 1670 MWt. This limit is conservative compared to an analytical limit of 45% of 1775 MWt because of the associated increase in the power dependent MCPR limit. Future calculations done to support each core reload will use 1775 MWt as the analytical assumption in the transient analyses. See commitment 5 in Exhibit H.

In this part of the power/flow map, substantial margin exists to the MCPR limit. Based on the results of the transient analyses, it is reasonable to conclude that a small relative increase in the analytical limit at rerate conditions will not have a significant effect on the analysis results. NSP will confirm that the analysis results will remain acceptable with no challenge to fuel thermal-hydraulic limits or to the Reactor Primary Coolant Boundary pressure limits, thus the proposed changes to the plant Technical Specifications and Bases are acceptable as demonstrated by the MNGP Power Rerate safety analyses. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

The commitment to confirm that the 45% of 1775 MWt (798.75 MWt) turbine trip bypass setpoint does not significantly affect the conclusions above is contained in Exhibit H.

F. Shutdown Cooling Isolation

Technical Specification Section 3.2, <u>PROTECTIVE INSTRUMENTATION</u>, TABLE 3.2.1. "Instrumentation That Initiates Containment Isolation Functions," Function item 6, page 50.

Changes are proposed to the plant Technical Specifications to modify the reference point for the shutdown cooling supply isolation trip setting.

The shutdown cooling supply isolation trip setting is an interlock provided to prevent over pressurizing the shutdown cooling system by inadvertent operation of the valves in either the suction line from the recirculation loop or the reactor vessel head spray line. Additional information concerning this function is provided in section 10.2.4.3 of the USAR. This interlock is currently based on the system pressure at the suction to the Residual Heat Removal (RHR) pumps. The specification is proposed to be changed to indicate that the interlock is based on the reactor vessel steam dome pressure.

The interlock is provided for equipment protection to prevent an intersystem Loss of Coolant Accident. In addition, the analytical limit for the shutdown cooling supply isolation trip setting is an input to licensing basis analysis for confirming the capability to place the plant in a cold shutdown condition. The interlock controls the pressure at which the residual heat removal (RHR) system may be placed in service.

In order to achieve a nominal setpoint of 75 psig at the reactor steam dome for the MNGP Power Rerate analyses, an analytical limit of 88 psig was used for the shutdown cooling supply isolation trip setting for these licensing basis analysis. The relief valve on the RHR pump suction piping for protection of the pump and piping has a setpoint of 150 psig, which is lower than the maximum allowable pressure for the shutdown cooling supply isolation piping. For setpoint calculations, a reactor dome pressure of 88 psig was used as an analytical limit for the reactor high pressure shutdown cooling signal. As a result of the increased pressure interlock setpoint, the RHR pump suction pressure will increase to 127 psig (88 psig + 39 psig static head).

Piping analysis was performed to demonstrate that the change in the shutdown cooling interlock has no adverse effect on the pressure, thermal, and seismic stress acceptance criteria as specified in USAR Chapter 12 for the RHR and Residual Heat Removal Service Water (RHRSW) systems. This analysis identified the need for a modification to the RHR Heat exchanger supports. See Exhibit H for NSP's commitment to implement this modification.

There is sufficient margin in the proposed shutdown cooling isolation trip setting to the RHR pump suction relief valve set pressure, thus the change is acceptable and has no adverse effect on the health and safety of the public. In addition, piping analysis and, as appropriate, modifications will ensure that the change has no adverse effect on pressure, thermal and seismic stress acceptance criteria of USAR Chapter 12. The proposed change does not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or result in a significant reduction in the margin of safety.

G. Low Pressure Core Cooling Pump Pressure/Automatic Depressurization Interlock

Technical Specification Section 3.2, PROTECTIVE INSTRUMENTATION, TABLE 3.2.2, "Instrumentation That Initiates Emergency Core Cooling Functions," Function item C.3, page 53.

Technical Specification Section 3.2, Bases, page 65.

Changes are proposed to the plant Technical Specifications to modify the limiting conditions for operation for the Automatic Depressurization System (ADS) permissive provided by the low pressure core cooling pumps discharge pressure. Changes are submitted for the Technical Specification Bases to provide bases information for the ADS low pressure core cooling pump discharge pressure interlock trip setting.

The ADS provides redundancy to the High Pressure Coolant Injection (HPCI) system in the event that the HPCI system does not provide adequate core cooling during a small break loss of coolant event. The ADS is provided to depressurize the reactor, thus ensuring that an adequate source of coolant will be available via the Core Spray system or the Low Pressure Coolant Injection (LPCI) system (the low pressure portion of the Emergency Core Cooling System).

The ADS accomplishes reactor vessel depressurization by blowdown through automatic opening of the ADS related safety/relief valves which vent steam to the suppression pool. For small breaks, the vessel is depressurized in sufficient time to allow either the Core Spray system or the LPCI system to provide adequate core cooling to prevent any fuel clad melting. For large breaks, the vessel depressurizes through the break without assistance. Depressurization of the reactor vessel may be accomplished manually by the reactor operator or without operator action by the ADS system. The ADS initiation logic consists of a delay timer which is actuated on a Reactor Vessel low-low water level signal provided there is confirmation that at least one low pressure Emergency Core Cooling System (ECCS) pump is operating and ready for injection as indicated by the pump discharge pressure. Additional information concerning the ADS is provided in sections 6.2.5 and 14.7.2.3.1.5 of the USAR.

The low pressure ECCS pump permissive trip setting for actuation of the ADS is proposed to be revised. The actual trip setting for this permissive is not assumed in any transient or accident analysis. The ECCS LOCA analysis does assume that at least one low pressure ECCS pump is available and thus the ADS initiation logic is satisfied. The ECCS pump discharge pressure permissive setpoint for the ADS is not affected by the power rerate operating conditions; however, the power rerate analyses has identified that the current permissive trip setting can be improved. The setpoint for the low pressure ECCS pump permissive for ADS should be set at a value such that it is less than the pump discharge pressure when the pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not running. The current Technical Specification limit does not establish a lower limit on the trip setting to preclude a potentially false permissive signal. MNGP proposes to change the specification to provide a lower limit on the permissive signal. MNGP has evaluated the permissive setpoint using the GE setpoint methodology specified in NEDC-31336, "General Electric Instrument Setpoint Methodology," as guidance. Evaluation of the permissive setpoint has determined that an increase in the trip setting is appropriate to provide additional margin from the pressure which would provide a false low pressure ECCS pump permissive. In addition, changes are submitted for the Bases to provide a discussion as to the basis for the trip setting.

The proposed changes to the limiting conditions for operation for the low pressure ECCS pump permissive trip setting for actuation of the ADS provides the required level of assurance, consistent with the safety analyses, that a valid permissive signal is provided to the ADS. Thus the proposed change is acceptable as evaluated using the GE setpoint methodology guidance. The proposed change provides an enhancement to the current Technical Specification by ensuring the permissive trip setting is properly established to preclude the potential for a false pump running indication. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or a significant reduction in the margin of safety.

H. Containment Cooling/Containment Spray

Technical Specification Section 3.5, CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS, Specification 3.5.C, Containment Spray/Cooling System, page 104.

Changes are proposed to the plant Technical Specifications to modify the limiting conditions for operation for the containment spray/cooling subsystem.

The containment cooling and containment spray subsystem is an integral part of the Residual Heat Removal (RHR) system. Two redundant subsystems are provided with each subsystem comprised of two (2) RHR pumps, two (2) Residual Heat Removal Service Water (RHRSW) pumps, one (1) RHR heat exchanger, and the associated valves and piping to establish the subsystem flow path. Each subsystem is manually operated and individually controlled. A discussion of the RHR system and the containment cooling and containment spray subsystem is provided in section 6.2.3 of the USAR. Following a design basis Loss of Coolant Accident (LOCA), the containment cooling subsystem removes heat from the suppression pool.

The containment cooling and containment spray subsystem is placed in service by manual operator action subsequent to a design basis LOCA. The most limiting design basis LOCA case for MNGP is the double-ended break of the recirculation suction line. For this event, the reactor is depressurized rapidly and break flow is released to the containment. The Emergency Core Cooling System (ECCS) initiates to provide core cooling upon receiving applicable signals. With an assumed loss-of-offsite power and the limiting single failure for the containment of a diesel generator, one Core Spray (CS) pump and two Low Pressure Coolant Injection (RHR pumps in LPCI mode) pumps remain for core cooling. Upon actuation of the ECCS, reactor vessel water volume is rapidly restored. After approximately 10 minutes, one RHR pump in the LPCI mode (with one RHR heat exchanger) may be transferred to the suppression pool cooling mode. The other RHR pump in the LPCI mode is secured and replaced by one RHRSW pump. Refer to USAR section 5.2.3.3 for additional discussion concerning the containment cooling system response to the design basis LOCA event. No credit is taken in the analysis of the containment response to the design basis LOCA for the RHR containment spray mode.

The effect of MNGP Power Rerate to 1775 MWt on the design basis Loss of Coolant Accident containment response was evaluated using the most limiting inputs for the complement of containment cooling equipment (one RHR pump, one RHRSW pump and one RHR heat exchanger), suppression pool and ultimate heat sink temperature of 90°F, with no credit taken for the containment spray mode of RHR and assuming a bounding reactor power of 1880 MWt. Because there will be more residual heat with an increased reactor power level, the containment long term response will have slightly higher temperatures. The bulk suppression pool temperature was found to increase an additional 10°F to a maximum of 194°F and the suppression chamber gas space temperature was conservatively assumed to be in equilibrium with the suppression pool. See Section 4.1 of Exhibit E for a discussion of the containment response at rerate conditions.

Long term suppression chamber temperatures remain within the design temperature of the structure, thus factors of safety provided in the ASME Code are maintained and safety margin is not affected. It was confirmed that the long term containment response does not adversely affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell or suppression chamber room. The drywell long term temperature response is not significantly affected by a bounding increase in reactor power up to 1880 MWt. Thus factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt with the complement of containment cooling equipment as proposed for Technical Specification 3.5.C.

The MNGP Power Rerate analyses confirmed that ECCS pump Net Positive Suction Head (NPSH) is adequate given the containment temperature response and the limiting complement of containment cooling equipment (one RHR pump, one RHRSW pump and one RHR heat exchanger). NSP has determined that the peak bulk suppression pool temperature with a power rerate to 1775 MWt is acceptable for ECCS pump NPSH in accordance with Technical Specification 3.5.C. See NSP's response to Question 51 of the staff's rerate RAI dated September 5, 1997 and NRC SER dated July 25, 1997 for additional information on ECCS pump NPSH at rerate conditions.

The proposed changes to the limiting conditions for operation for the containment cooling subsystem maintains assurance that two (2) redundant subsystems are operable to provide the required redundancy to perform the post accident heat removal function of the containment cooling system, assuming the worst case single active failure coincident with a loss of offsite power. The required complement of containment cooling equipment as proposed has been demonstrated by the MNGP Power Rerate evaluation as fully capable of performing the required safety function for a bounding increase in the reactor power to 1880 MWt, thus this evaluation demonstrated MNGP Power Rerate to 1775 MWt with the proposed changes is acceptable. One RHR pump and one RHRSW pump are fully capable of removing the post accident containment heat loads, thus each of the independent containment cooling subsystems have additional installed redundancy.

Deletion of the limiting conditions for operation specified by Technical Specifications 3.5.C.2 and 3.5.C.3 has no adverse effect on the capability of the system to perform required functions. Under the worst case, with an RHRSW pump inoperable in each subsystem, the remaining required operable equipment is fully capable to perform the RHRSW heat removal function. Overall reliability is not reduced because two fully redundant containment cooling subsystems are maintained with the proposed changes. With one RHRSW pump inoperable in each subsystem, a single failure can be taken such as the loss of an emergency diesel generator, with the remaining operable pumps and flow paths fully capable of containment heat removal following a design basis accident. Capability of this configuration has been demonstrated in the containment long term response analysis, therefore the proposed changes are acceptable and assure the required complement of equipment is operable to mitigate the consequences of the worst case design basis accident.

The limiting conditions for operation for one containment cooling subsystem or both containment cooling subsystems inoperable and the bases for these specifications remain unchanged. With one containment cooling subsystem inoperable, the remaining operable containment cooling subsystem is adequate to perform the containment cooling function. However the overall reliability is reduced because of a potential single failure affecting the remaining operable subsystem. The seven (7) day out-of-service time is based on the 100% redundant capability of the operable containment cooling subsystem consisting of 1 RHR pump, 1 RHR heat exchanger and 1 RHRSW pump and the low probability of an event occurring requiring containment cooling during this period. With both containment cooling subsystems inoperable, the containment cooling function can not be performed, thus timely action must be taken to restore one subsystem or to place the plant in a condition for which containment cooling is not required.

The proposed changes to the MNGP Technical Specifications for the containment spray/cooling subsystems are consistent with the safety analysis for MNGP Power Rerate and remain consistent with the current plant safety analysis. The analysis has demonstrated that long term suppression chamber temperatures at power rerate conditions have no adverse effect on the containment structure or the environmental qualification of equipment located in the drywell or suppression chamber room. This analysis assumed the limiting complement of containment cooling equipment specified by the proposed change. Factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt with the most limiting complement of containment cooling equipment. The proposed changes to the limiting conditions for operation for the containment cooling equipment provide the required level of assurance, consistent with the safety analyses, that the required complement of equipment will be operable to perform the required safety function. Thus the proposed changes are acceptable. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or a significant reduction in the margin of safety.

I. Editorial Changes

Technical Specifications, Table of Contents, page ii.

Technical Specifications, Table of Contents, page iii.

Technical Specifications, Section 3.1, Bases, page 39

Technical Specifications, Section 3.2, Bases, page 65

Technical Specification Section 3.5/4.5, Section A, ECCS System, Bases, page 112.

Technical Specification Section 3.5/4.5, Section B, RHR Intertie Line, Bases, page 112.

Editorial corrections are proposed to correct inconsistencies within the Technical Specifications. Page number listings provided in the Technical Specification Table of Contents are proposed to be revised to reflect the actual page numbering established in the body of the specifications. In addition, corrections are proposed to establish consistency between the Table of Contents section headings and the section headings in the body of the specifications.

Bases revisions to Section 3.1 page 39 were made in accordance with NSP's response to Question 36 contained in NSP's rerate RAI response letter dated September 5, 1997. Bases revisions are provided on page 68 to delete reference to a report made obsolete by this license amendment request. Bases revisions are also provided to page 112 of the MNGP Technical Specifications to establish consistency between the Automatic Depressurization System (ADS) Bases and the ADS Technical Specifications for the out-of-service time. Finally, clarification is provided to the Bases concerning the discussion of the Residual Heal Removai (RHR) Intertie Line found on page 112 of the MNGP Technical Specifications.

The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

J. Changes to Reactor Water Level Instrument Setpoints

At rerate conditions, the increased steam flow results in a larger pressure drop across the steam dryer/separator. Consequently, the difference (depression) between the measured level obtained from the annulus region and the actual level inside the shroud increases. The depression at 100% power, based on calculations done for the rerate program, increases to about 9" at 1775 MWt.

In order to resolve this discrepancy, all low level trips will be shown as a function of measured level in the annulus of ≥ 7 inches which corresponds to ≥ 10'4" above the top of active fuel inside the shroud at 1775 MWt. Transients have been analyzed for rerate assuming a depression larger than 9" for conservatism. The transient analysis determined all applicable acceptance criteria were met.

Additional discussions of various systems, structures, and components that have been evaluated for MNGP Power Rerate but do not involve changes to the plant Technical Specifications, are provided in Exhibit E.

IV. Determination of Significant Hazards Considerations

The proposed change to the Operating License has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10CFR50.91 using standards provided in 10CFR50.92. This analysis is provided below.

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

The probability of occurrence and consequences of an accidents previously evaluated have been evaluated for MNGP Power Rerate. This evaluation has concluded that MNGP Power Rerate will not involve a significant increase in the probability of occurrence or consequences of previously evaluated accidents.

1. Evaluation of Accident Consequences

a) ECCS-LOCA Analysis

The Emergency Core Cooling System Loss of Coolant Accident (ECCS-LOCA) performance analysis has been evaluated for MNGP Power Rerate using methodology which has been approved by the NRC for 10CFR50.46 analyses. The current ECCS performance requirements were used in the power rerate analysis; no further parameter relaxations were included in the analysis. The ECCS-LOCA analysis was performed for MNGP Power Rerate for the existing licensed rated thermal power and at a bounding thermal power level of 1880 MWt that is approximately 6% greater than the proposed power rerate to 1775 MWt. In addition, the bounding thermal power level was increased by an additional 2% in accordance with regulatory guidance. The licensing peak clad temperature for the bounding analyzed thermal power level remains below the 10CFR50.46 required limit of 2,200°F. Therefore the analysis demonstrates that MNGP will continue to comply with 10CFR50.46 and 10CFR50, Appendix K at rerated conditions thus the consequences of a LOCA is not significantly increased for the proposed power rerate.

b) Abnormal Operating Transient Analysis

An evaluation of the Updated Safety Analysis Report (USAR) and reload transients has been performed for MNGP Power Rerate to demonstrate that the proposed power rerate has no adverse effect on plant safety. This evaluation was performed for a power level of 1775 MWt, with the exception that certain event evaluations were performed at 102% of the rerate power level. The transient analysis performed to demonstrate the acceptability of MNGP Power Rerate used the NRC approved methods identified in the MNGP Technical Specifications.

The limiting transient events at the power rerate conditions have been analyzed. This includes all events that establish the core thermal operating limits and the events that bound other transient acceptance criteria. These limiting transients were benchmarked against the existing rated thermal power level by performance of the event analysis at both the proposed rerate power level and the existing rated power level. In addition, an expanded group of transient events was evaluated to confirm that these events were less severe with the power rerate than the most limiting transients. The events included in the expanded group of transient events were chosen based on those events which have been demonstrated to be sensitive to initial power level. This evaluation confirmed that the existing set of limiting transient events remains valid for MNGP Power Rerate. The evaluation was performed for a representative core and demonstrated the overall capability to meet all transient safety criteria for the power rerate. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The results of the evaluation of transients demonstrate that the power rerate can be accomplished without a significant increase in the consequences of the transients evaluated. The fuel thermal-mechanical limits at the power rerate conditions are within the specific design criteria for the GE fuels currently loaded in the MNGP core. Also, the power-dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) methods developed as part of the core performance improvement program remain applicable to rerate conditions. The transient event evaluation confirmed that MNGP Power Rerate has no significant effect on the power-dependent and flow-dependent MCPR and MAPLHGR limits. The peak reactor pressure vessel bottom head pressure remains within the ASME requirement for reactor pressure vessel overpressure protection.

The effects of plant transients were evaluated by assessing a number of disturbances of process variables and malfunctions or failures of equipment consistent with USAR. The transient events were evaluated against the Safety Limit Minimum Critical Power Ratio, (SLMCPR). The SLMCPR is determined using NRC-approved methods. The limiting transient events are slightly more severe when initiated from the rerate power level. The power rerate transient evaluation results show a slightly more limiting event initial CPR ($\leq$ 0.02) than that initiated from the present rated power level for the near limiting transients However, for the most limiting transient, the evaluation of a representative core showed that no change is required to the Operating Limit MCPR for the power rerate and that the integrity of the SLMCPR is maintained. The margin of safety established by the SLMCPR is not affected and the event consequences are not significantly affected by the proposed power rerate to 1775 MWt. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The results demonstrate that the MNGP core thermal power output can be safely increased to the power rerate level without significant effect on the consequences of previously evaluated postulated transient events. The results of the rerate transient analysis are summarized as follows.

(1) Events Resulting in a Nuclear System Pressure Increase

(a) Main Generator Load Rejection with No Steam Bypass

At rerate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC accepted design criteria.

(b) Main Turbine Trip with No Steam Bypass

At rerate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC accepted design criteria.

(c) Main Steam Isolation Valve Closure, Flux Scram

The peak reactor pressure vessel bottom head pressure for rerate conditions is slightly higher than the reactor pressure vessel bottom head pressure at current conditions. However, the resultant pressure is still below the ASME overpressure limit of 1,375 psig.

(d) Slow Closure of a Single Turbine Control Valve

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

(2) Event Resulting in a Reactor Vessel Water Temperature Decrease

(a) Feedwater Controller Failure-Maximum Demand

The delta CPR calculated for this event at rerate conditions is about 0.01 higher than the corresponding value for the current rated power when the impact of the new condensate pumps is factored in. The trend for the Feedwater Controller Failure-Maximum Demand event is consistent with the analysis for the current rated power. The fuel thermal margin results are within the acceptable limits for the fuel types analyzed.

(b) Loss of Feedwater Heating

This event at the rerate conditions remains significantly less than the cycle operating MCPR limit. The results at low core flow conditions are actually slightly higher than for the high core flow condition because of increased inlet coolant subcooling into the reactor core. The calculated thermal and mechanical overpower limits at the power rerate conditions for this event also meet the fuel design criteria.

(c) Inadvertent HPCI Actuation

For the limiting condition analyzed, both the high water level setpoint and the high reactor pressure vessel steam dome pressure scram setpoints are not reached. Based on the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event with respect to delta Critical Power Ratio (ΔCPR) considerations. In addition, the fuel transient thermal and mechanical overpower limits remain within the NRC accepted design values.

(3) Event Resulting in a Positive Reactivity Insertion

(a) Rod Withdrawal Error (RWE)

The current Rod Block Monitor (RBM) system for MNGP with power dependent setpoints was analyzed for the rod withdrawal error event at the power rerate conditions using a statistical approach consistent with NRC approved methods. The analysis concluded that the transient is slightly more severe with a greater delta Critical Power Ratio (ΔCPR) from the initial most limiting CPR. However, the fuel and mechanical overpower results remain within the NRC accepted design criteria.

(4) Event Resulting in a Reactor Vessel Coolant Inventory Decrease

(a) Pressure Regulator Failure to Full Open

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

(b) Loss of Feedwater Flow

This transient event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. Water level declines rapidly and a low level causes a reactor scram. The closure of the main steam isolation valves and the actuation of High Pressure Coolant Injection and Reactor Core Isolation Cooling terminate the event. This event was included in the power rerate evaluation to provide assurance that sufficient water makeup capability is available to keep the core covered when all normal feedwater is lost. The generic analysis performed in support of the extended power uprate program shows that at the power rerate conditions a large amount of water remains above the top of the active fuel. These sequences of events do not require any new operator actions or shorter operator response times. Therefore, the operator actions for the event do not significantly change for the power rerate.

(5) Event Resulting in a Core Coolant Flow Decrease

(a) Recirculation Pump Seizure

The recirculation pump seizure assumes instantaneous stoppage of the pump motor shaft of one recirculation pump. As a result, the core flow decreases rapidly. The heat flux decline lags core power and flow and could result in a degradation of core heat transfer. At the power rerate conditions, the transient results confirmed that the consequences of the pump seizure event remain non-limiting.

(6) Event Resulting in a Core Coolant Flow Increase

(a) Recirculation Flow Controller Failure Increasing Flow

The results of this transient for the power rerate remain non-limiting as compared with other more severe pressurization events.

c) Design Basis Accident Challenges to the Containment

The primary containment response to the limiting design basis accident was evaluated for a bounding reactor power level approximately 6% greater than the proposed power rerate to 1775 MWt. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The effect of the power rerate on the short term containment response (peak values) as well as the long term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the bounding power level, thus the proposed power rerate to 1775 MWt is acceptable. Factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt.

Short-term containment response analyses were performed for the limiting design basis LOCA consisting of a, double-ended guillotine break of a recirculation suction line, to demonstrate that operation at a bounding reactor power will not result in exceeding the containment design limits. This limiting design basis LOCA event results in the highest short-term containment pressures and dynamic loads. The analysis determined that for a bounding reactor power the maximum drywell pressure values are bounded by the current USAR analysis value and by the containment design pressure. The power rerate to 1775 MWt has no adverse effect on the containment structural design pressure.

Because there will be more residual heat with increased thermal power, the containment long term response will have slightly higher temperatures. Long term suppression chamber temperatures remain within the design temperature of the structure, thus factors of safety provided in the ASME code are maintained and safety margin is not affected. Analysis confirmed that ECCS pump NPSH is adequate for this temperature response. It was confirmed that the long term response does not adversely affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell or suppression chamber room. The drywell long term temperature response is not adversely affected for a bounding reactor power. An analytical power level of 1880 MWt bounds the decay heat associated with the 1775 MWt power level with a one sided confidence interval of 95%. The containment long term response is therefore acceptable for the power rerate to 1775 MWt.

The impact of a reactor power increase on the containment dynamic loads have been determined, evaluated and found to have no adverse effects for conditions which well bound the proposed power rerate. Thus the containment dynamic loads were found to be acceptable for the power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the primary containment response to the design basis accident confirmed that the power rerate does not result in a significant increase in consequences for a bounding reactor power approximately 6% greater than the proposed power rerate to 1775 MWt.

d) Radiological Consequences of Design Basis Accidents

For MNGP Power Rerate, the radiological consequences of the limiting design basis accidents were re-evaluated. These evaluations included the effect of the power rerate on the radiological consequences of accidents presented in USAR Section 14.7.

This evaluation was performed using inputs and evaluation techniques consistent with the current regulatory guidance, the current GE analysis methods, and the appropriate plant design basis. The inputs and analysis methods used for MNGP Power Rerate differ from those utilized in the current licensing basis evaluation presented in the USAR and the AEC safety evaluation supporting plant initial licensing. The MNGP Power Rerate evaluations used the more contemporary staff approved methods. The inputs used in the MNGP Power Rerate evaluation provide a conservative assessment of the potential radiological consequences. The conclusions of these evaluations are consistent with the original licensing basis evaluations. The radiological consequences of the limiting design basis accidents remain well within 10CFR100 guidelines for a bounding thermal power approximately 6% greater than the proposed power rerate of 1775 MWt. In addition the bounding thermal power level was increased by an additional 2% in accordance with regulatory guidance.

To conservatively analyze the change in consequences, the evaluation of radiological consequences using the analysis inputs and methods was performed for the existing licensed rated thermal power and a thermal power bounding the proposed power rerate. This provides a conservative bounding change in consequences for the requested power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the radiological consequences of design basis accidents confirmed that the power rerate does not result in a significant increase in consequences for a bounding power level approximately 6% greater than the proposed power rerate. The results remain below the 10CFR100 guideline values as well as the licensing basis established in the March 18, 1970 AEC safety evaluation. Therefore, the postulated radiological consequences do not represent a significant change in accident consequences and are clearly within the regulatory guidelines for the proposed power rerate to 1775 MWt.

e) Other Evaluations

(1) Performance Improvements

The MNGP Power Rerate safety analysis has been performed taking into account the implementation of the following previously approved operational features.

(a) Maximum Extended Load Line Limit/Increase Core Flow (MELLL/ICF)

The safety analysis for rerate conditions shows that the extended operating domain as analyzed by MELLL/ICF remains valid for the power rerate conditions.

(b) Average Power Range Monitor/Rod Block Monitor Technical Specification (ARTS) Improvements

The safety analysis for rerate conditions shows that the ARTS improvements remain valid for the power rerate conditions.

(c) Single Loop Operation (SLO)

The safety analysis for rerate conditions shows that the single loop operating mode remains valid for the power rerate conditions. The MELLLA trip setpoints determined for two-loop operation were confirmed to be acceptable for single loop operation with a correction applied to account for the actual effective drive flow applied when operating in single loop. The single loop settings have been conservatively established to be consistent with the two loop settings while ensuring the appropriate corrections are applied to the MAPLHGR and the operating limit MCPR to account for single loop operation.

(2) Effect of Power Rerate on Support Systems

An evaluation was performed to address the effect of MNGP Power Rerate on accident mitigation features, structures, systems, and components within the balance of plant. The results are as follows:

- Auxiliary systems such as, building heating, Ventilation and Air Conditioning (HVAC) systems, reactor building closed cooling water, service water and emergency service water, spent fuel pool cooling, process auxiliaries such as instrument air and make up water and the post-accident sampling system were confirmed to operate acceptably under normal and accident conditions at rerate conditions.

- The secondary containment and standby gas treatment system were confirmed to be able to adequately contain, process, and control the release of normal and post-accident levels of radioactivity at rerate conditions.

- Instrumentation was reviewed and confirmed to be capable of performing its control and monitoring functions under rerate conditions. As required, analyses were performed to determine the need for setpoint changes for various functions (e.g., APRM neutron flux scram setpoints). In general, setpoints are to be changed only to maintain adequate difference between plant operating parameters and trip setpoints, while ensuring safety performance is demonstrated. The revised setpoints have been established using the NRC reviewed methodology as guidance.

- Electric power systems including the turbine generator and switchgear components were verified as being capable of providing the electrical load as a result of the rerate power levels. An evaluation of the auxiliary power system for the power rerate conditions confirmed that the system has sufficient capacity with the changes identified in Exhibit I to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the required engineered safeguards equipment following postulated accidents. No safety-related electrical loads were affected which would adversely impact the emergency diesel generators.

- Piping systems were evaluated for the effect of operation at higher power levels, including transient loading. The evaluation confirmed that, with few exceptions, piping and supports are adequate to accommodate the increased loading resulting from operation at rerate power conditions. In a few cases, piping supports will be modified to accept higher forces due to rerate conditions.

- The effect of rerate conditions on high energy line break (HELB) was evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement and blowdown forces and the environmental effects resulting from HELB events at rerate conditions.

- Control room habitability was evaluated. With the implementation of minor hardware and non-hardware changes to the control room ventilation system, Post-accident Control Room and Technical Support Center doses at rerate conditions were confirmed to be within the guidelines of General Design Criterion 19 of 10CFR50, Appendix A.

- The environmental qualification of equipment important to safety was evaluated for the effect on normal and accident operating conditions at rerate power levels. The equipment remains qualified for the new conditions. Minor adjustments will reflect some changes to maintenance frequencies. The preventative maintenance program will continue to provide for equipment maintenance or replacement to ensure equipment environmental qualification at rerate power conditions.

(3) Effect on Special Events

The consequences of special events (i.e., ATWS, 10CFR50 Appendix R, and Station Blackout) remain within NRC accepted criteria for rerate conditions. Concurrent malfunctions assumed to occur during accidents have been accounted for in the safety analyses for rerate conditions. The consequences of these equipment malfunctions does not change with implementation of the MNGP Power Rerate program. The generic ATWS analysis for operation at rerate conditions is being revised. The revision is not expected to affect MNGP compliance with NRC acceptance criteria.

f) Conclusion

The evaluation of the Emergency Core Cooling System performance has demonstrated the criteria of 10CFR50.46 are satisfied, thus the margin of safety established by the criteria is maintained. The analysis demonstrated that the ECCS will function with the most limiting single failure to mitigate the consequences of the accidents and maintain fuel integrity. The system will continue to perform as required under rerate conditions to mitigate the consequences of accidents and thus the power rerate does not adversely affect ECCS performance in a manner to increase the severity of consequences. Challenges to the containment have been evaluated and the integrity of the fission product barrier has been confirmed. The radiological consequences of design basis accidents have been evaluated and it was found that the effect of the proposed power rerate on postulated radiological consequences does not result in a significant increase in accident consequences. These evaluations have been performed for a bounding reactor power approximately 6% greater than the proposed power rerate. In addition the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. Thus the evaluations provide conservative bounding results for the proposed power rerate to 1775 MWt and demonstrate that the proposed power rerate does not result in significant increase in accident consequences.

The abnormal transients have been analyzed under the power rerate conditions, and the analysis has confirmed that the power rerate to 1775 MWt has only a minor effect on the minimum critical power ratio and that no change to the safety limit critical power ratio results, thus the margin of safety as assured by the safety limit critical power ratio is maintained. The effect of the power rerate on the consequences of abnormal transients which result from potential component malfunctions has been shown to be acceptable, thus the power rerate does not result in a significant increase in transient event consequences.

The spectrum of analyzed postulated accidents and transients has been investigated, and has been determined to meet the current regulatory criteria for the MNGP at rerate conditions. In the area of core design, the fuel operating limits will still be met at the rerate power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in the plant Technical Specifications. The evaluation of transient and accident consequences was performed consistent with the proposed changes to the plant Technical Specifications. Therefore, the proposed Operating License and Technical

Specification changes will not cause a significant increase in the consequences of an accident previously evaluated for the Monticello plant.

2. **Evaluation of the Probability of Previously Evaluated Accidents**

The proposed power rerate imposes only minor increases in the plant operating conditions. No changes are required to the rated core flow, rated reactor pressure, or turbine throttle pressure. The power rerate will result in moderate flow increases in those system associated with the turbine cycle (i.e., condensate, feedwater, main steam, etc.). For MNGP Power Rerate, the small increase in operating temperatures for balance of plant support systems has no significant effect on LOCA or other accident probabilities. The increase in flow rates in balance of plant systems is addressed by compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." The MNGP Power Rerate evaluations have confirmed that the power rerate has no significant effect on flow induced erosion/corrosion. The worst case limiting feedwater and main steam piping flow increases were evaluated to be approximately proportional to the power increase. The affected systems are currently monitored by the MNGP Erosion/Corrosion program. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

The occurrence frequency of accident precursors and transients have been addressed when required by applying the guidance of NRC reviewed setpoint methodology to insure that acceptable trip avoidance is provided during operational transients subsequent to implementation of rerate. The setpoint evaluation has confirmed that MNGP Power Rerate does not result in any increase in challenges to the plant protective instrumentation.

Plant systems, components, and structures have been verified to be capable of performing their intended functions under rerate conditions with a few minor exceptions. Where necessary, some components will be modified prior to implementation of the MNGP Power Rerate Program to accommodate the revised operating conditions (e.g., a limited number of pipe supports changes, instrumentation setpoint changes, control room habitability improvements). MNGP Power Rerate does not significantly affect the reliability of plant equipment. Where reliability effects have been identified, modifications and administrative controls will be implemented prior to the power rerate to adequately compensate. No new components or system interactions that could lead to an increase in accident probability are created due to the power rerate.

The probability (i.e., frequency of occurrence) of design basis accidents occurring is not affected by the increased power level, as the applicable criteria established for plant equipment (e.g., ANSI Standard B31.1, ASME Code,) will still be followed as the plant is operated at the rerate power level. The MNGP Power Rerate analysis basis assures that the power dependent margin prescribed by the Code of Federal

Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the Code design rules have been demonstrated to be maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant. Reactor scram setpoints as established are such that there is no significant increase in scram frequency due to rerate conditions. No new challenges to safety-related equipment will result from the power rerate. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant increase in the probability of an accident previously evaluated.

B. The proposed Operating License changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The basic Boiling Water Reactor configuration, operation and event response is unchanged by the power rerate. Analysis of transient events has confirmed that the same transients remain limiting and that no transient events result in a new sequence of events which could lead to a new accident scenario. The MNGP Power Rerate analyses confirmed that the accident progression is basically unchanged by the power rerate.

An increase in power level will not create a new fission product release path, or result in a new fission product barrier failure mode. The same fission product barriers such as the fuel cladding, the reactor coolant pressure boundary and the reactor containment, remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits and is demonstrated by the MNGP Power Rerate transient analysis and accident analysis. Similarly, analysis of the reactor coolant pressure boundary and primary containment have demonstrated that the power rerate has no adverse effect on these fission product barriers. The proposed changes to the plant Technical Specifications to support the power rerate implementation are consistent with the MNGP Power Rerate analyses and assure transient and accident mitigation capability in compliance with regulatory requirements.

The effect of MNGP Power Rerate on plant equipment has been evaluated . No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode resulting from the power rerate was identified. The full spectrum of accident considerations defined in the USAR have been evaluated and no new or different kind of accident resulting from the power rerate has been identified. MNGP Power Rerate uses already developed technology and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria which includes accepted codes, standards, and methods. GE has designed BWRs of higher power levels than the rerate power of any of the currently operating BWR fleet and no new power dependent accidents have been identified. In addition, MNGP Power Rerate does not create any new sequence of events or failure modes that lead to a new type of accident.

All actions to ensure that safety-related structures, systems, and components will remain within their design allowable values and ensure they can perform their intended functions under rerate conditions will be taken prior to implementation of the power rerate. MNGP Power Rerate does not increase challenges to or create any new challenge to safety-related equipment or other equipment whose failure could cause an accident. Plant modifications required to support implementation of MNGP Power Rerate will be made to existing systems (e.g., a limited number of pipe supports, instrumentation setpoints, control room habitability improvements), rather than by adding new systems of a different design which might introduce new failure modes or accident sequences. The Technical Specification changes required to implement the power rerate require little change to the plant's configuration, and all changes have been evaluated and are acceptable.

Therefore, the proposed Operating License and Technical Specification changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. **The proposed Operating License changes do not involve a significant reduction in a margin of safety.**

The accident analysis, as well as a majority of the plant specific evaluations performed in support of MNGP Power Rerate have been performed assuming a bounding steady state power level 112.6% of the existing licensed limit of 1670 MWt, and approximately 6% above the licensed maximum thermal power level of 1775 MWt proposed by MNGP Power Rerate. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance when applicable for the evaluation of accidents and transients. For plant conditions associated with a bounding analysis power level, the analyses demonstrated operating margin to criteria establishing margins of safety, thus additional operating margin is demonstrated and assured for the proposed power rerate to 1775 MWt and added confidence is established in the integrity of criteria establishing margin to safety.

The cycle specific transient analysis, as well as the analysis to establish plant instrumentation set points have been performed assuming a plant steady state power level of 1775 MWt. This analysis approach was taken in order to demonstrate safety and equipment margins while ensuring appropriate cycle specific operating limits. The evaluation of transient events and instrument setpoints demonstrated operating margin to criteria establishing margins of safety for the proposed power rerate conditions.

The MNGP Power Rerate analysis basis assures that the power dependent safety margin assuring criteria prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the code design rules have been maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant.

1. **Fuel Thermal Limits**

No change is required in the basic fuel design to achieve the rerate power levels or to maintain the margins as discussed above. No increase in the allowable peak bundle power is requested for the power rerate. The abnormal transients have been evaluated under the power rerate conditions for a representative core configuration. The analysis has confirmed that the power rerate has no adverse effect on the operating limit Minimum Critical Power Ratio (MCPR) and that no change to the safety limit MCPR results, thus the margin of safety as assured by the safety limit MCPR is maintained. The fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and the operating limit MCPR will still be met at the rerate power level. The MNGP Power Rerate analyses have confirmed the acceptability of these operating limits for the power rerate without an adverse effect on margins to safety. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

2. **Design Basis Accidents Challenges to Fuel**

The evaluation of the Emergency Core Cooling System performance has demonstrated the criteria of 10CFR50.46 are satisfied, thus the margin of safety established by the criteria is maintained. This evaluation was performed for a bounding reactor power level approximately 6% greater than the proposed power rerate. In addition the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The analysis demonstrates that MNGP will continue to comply the 10CFR50.46 at the rerate conditions and that the margin of safety established by the regulation is maintained for the proposed power rerate.

3. **Design Basis Accident Challenges to Containment**

The primary containment response to the limiting design basis accident was evaluated for a bounding reactor power level approximately 6% greater than the proposed power rerate to 1775 MWt. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The effect of the power rerate on the short term containment response (peak values) as well as the long term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the bounding power level, thus the proposed power rerate to 1775 MWt is acceptable. Factors of safety provided in the ASME Code are maintained and safety margin is not affected for the power rerate to 1775 MWt.

Short-term containment response analyses were performed for the limiting design basis LOCA consisting of a, double-ended guillotine break of a recirculation suction line, to demonstrate that operation at a bounding reactor power will not result in exceeding the containment design limits. The analysis determined that for a bounding reactor power the maximum drywell pressure values are bounded by the current USAR analysis value and by the containment design pressure. The power rerate to 1775 MWt has no adverse effect on the containment structural design pressure.

Long term suppression chamber temperatures remain within the design temperature of the structure, thus factors of safety provided in the ASME code are maintained and safety margin is not affected. An analytical power level of 1880 MWt bounds the decay heat associated with the 1775 MWt power level with a one sided confidence interval of 95%. Analysis confirmed that ECCS pump NPSH is not adversely affected with this temperature response. It was confirmed that the long term response does not significantly affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell or suppression chamber room.

The impact of a reactor power increase on the containment dynamic loads have been determined, evaluated and found to have no adverse effects for conditions which well bound the proposed power rerate. Thus the containment dynamic loads were found to be acceptable for the power rerate to 1775 MWt.

The MNGP Power Rerate evaluation of the primary containment response to the design basis accident confirmed that the power rerate does not result in a reduction in margins of safety for a bounding reactor power approximately 6% greater than the proposed power rerate to 1775 MWt.

4. Design Basis Accident Radiological Consequences

The Updated Safety Analysis Report (USAR) provides the radiological consequences for each of the design basis accidents. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. For power rerate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor which will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of design basis accidents have been evaluated, and it was found that the consequences did not result in a significant increase in consequences for a bounding reactor power level approximately 6% greater than the proposed power rerate. In addition, the bounding reactor power level was increased by an additional 2% in accordance with regulatory guidance. The results remain below the 10CFR100 guideline values as well as the licensing basis established in the March 18, 1970 AEC safety evaluation. Therefore, the postulated radiological consequences are clearly within the regulatory guidelines and all radiological safety margins are maintained for the power rerate to 1775 MWt.

5. Transient Evaluations

The effects of plant transients were evaluated by assessing a number of disturbances of process variables and malfunctions or failures of equipment consistent with USAR. The transient events were evaluated against the Safety Limit Minimum Critical Power Ratio, (SLMCPR). The SLMCPR is determined using NRC-approved methods. The Power Rerate transient analyses were performed using the approved methodology specified in the plant Technical Specifications. The limiting transient events are slightly more severe when initiated from the rerate power level. The power rerate transient evaluation results show a slightly more limiting transient initial CPR ($\leq$ 0.02) than that initiated from the present rated power level for the near limiting transients. However, for the most limiting transient, the evaluation of a representative core showed that no change is required to the Operating Limit MCPR for the power rerate and that the integrity of the SLMCPR is maintained. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and to establish cycle specific operating limits.

The fuel thermal-mechanical limits at the power rerate conditions are within the specific design criteria for the GE fuels currently loaded in the MNGP core. Also, the power dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) methods developed as part of the core performance improvement program remain applicable to rerate conditions. The transient event evaluation confirmed that MNGP Power Rerate has no significant effect on the power-dependent and flow-dependent MCPR and MAPLHGR limits. The peak reactor pressure vessel bottom head pressure remains within the ASME requirement for reactor pressure vessel over pressure protection.

The margin of safety established by the SLMCPR is not affected by the proposed power rerate to 1775 MWt.

6. Technical Specification Changes

The Technical Specifications ensure that the plant and system performance parameters are maintained at the values assumed in the safety analysis. The Technical Specification (setpoints, trip settings, etc.) are selected such that the actual equipment is maintained equal to or conservative with respect to the inputs used in the safety analysis. Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. The Technical Specifications address equipment availability and limit equipment out-of-service to assure that the plant can be expected to have at least the complement of equipment available to deal with plant transients as that assumed in the safety analysis. The evaluations and analyses performed to demonstrate the acceptability of MNGP

Power Rerate were performed using inputs consistent with the proposed changes to the plant Technical Specifications.

The events that form the Technical Specification Bases were evaluated for the power rerate conditions using inputs and initial conditions consistent with the proposed Technical Specification changes. Although some changes to the Technical Specifications are required for the power rerate, no NRC acceptance limit will be exceeded. Therefore, the margins of safety assured by safety limits and other Technical Specification limits will be maintained. The changes to the Technical Specification Bases proposed by this submittal are consistent with the evaluations which demonstrated acceptability of the power rerate.

7. **Conclusion**

The spectrum of postulated accidents, transients, and special events has been investigated and have been determined to meet the current regulatory criteria for the MNGP at the power rerate conditions. In the area of core design, the fuel operating limits will still be met at the rerate power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in the plant Technical Specifications. Challenges to fuel or ECCS performance were evaluated and shown to meet the criteria of 10CFR50.46 and 10CFR50, Appendix K. Challenges to the containment have been evaluated and the integrity of the fission product barrier has been confirmed. Radiological release events have been evaluated and shown to meet the guidelines of 10CFR100. The proposed Operating License and Technical Specification changes are consistent with the MNGP Power Rerate evaluation performed. The evaluations demonstrated compliance with the margin assuring acceptance criteria contained in applicable codes and regulations. Therefore, the proposed Operating License and Technical Specifications changes will not involve a significant reduction in the margin of safety.

V. Environmental Evaluation Summary

Northern States Power has performed an evaluation to determine the environmental effects which the proposed licensing action might cause. The evaluation is provided in Exhibit F, "MNGP Power Rerate Environmental Evaluation." The information provided in Exhibit F demonstrates that the proposed changes will have no significant effect on the human environment. This information is provided to aid the Commission in complying with its statutory obligations under section 102(2) of the National Environmental Policy Act (NEPA).

The environmental effects of the proposed power rerate were reviewed. Based on this review, it is concluded that the proposed rerate will have an insignificant effect on the environment and the plant will be operated in an environmentally acceptable manner as established by the Atomic Energy Commission, predecessor agency to the NRC, in its Final Environmental Statement (FES). Except for administrative changes, existing Federal, State, and local regulatory permits presently in effect will accommodate the power rerate without modification to permit requirements. Effects to air, water, and land resources will be essentially non-existent. Exhibit F includes additional informat on concerning the environmental effect of the power rerate.

The NRC has stated in 10CFR51.22(c)(9) that certain licensing actions are eligible for a categorical exclusion from NEPA review where the NRC has found that the category of actions does not individually or cumulatively have a significant effect on the human environment. Given the above, NSP submits that the proposed power rerate meets the eligibility criteria for the categorical exclusion from environmental review set forth in 10CFR51.22(c)(9).

1. **Issuance of an amendment to a permit or license for a reactor pursuant to 10CFR Part 50.**

   The proposed power rerate would be implemented in the form of an amendment to the Monticello Part 50 License.

2. **Changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10CFR Part 20, or which changes an inspection or a surveillance requirement.**

   The proposed power rerate would change requirements with respect to uses of facility components located in the restricted area as defined in 10 CFR Part 20. As described in Exhibit A, NSP proposes to make changes to various Operating License and Technical Specifications requirements associated with the planned implementation of the power rerate. None of the requirements which are proposed to be changed involve facility components located outside the restricted area. In addition, the proposed amendment would change certain inspection requirements found in plant Technical Specifications.

3. **The amendment involves no significant hazards consideration.**

   As discussed in Exhibit A, the proposed amendment involves no significant hazards consideration. First, the proposed power rerate will not involve a significant increase in the probability of occurrence or consequences of previously evaluated accidents. Second, the proposed power rerate does not create the possibility of a new or different kind of accident from any accident previously evaluated. Third, the proposed power rerate does not involve a significant reduction in a margin of safety.

4. **There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.**

   As demonstrated in Sections 6.0 and 7.0 of Exhibit F, the proposed power rerate involves no significant change in the types of effluents that may be released offsite. In addition, the proposed power rerate involved no significant increase in the amounts of the effluents that may be released offsite.

5. **There is no significant increase or cumulative occupational radiation exposure.**

   As demonstrated in Exhibit F, Section 7.2.1 with respect to in plant radiation conditions, Section 7.3, "Radiological Consequences of Accidents," and Section 8.0 concerning the environmental effects of uranium fuel cycle activities, the proposed power rerate does not involve significant increases in individual or occupational radiation exposure.

Given the above, NSP believes that the proposed amendment meets the criteria specified in 10CFR51.22(c)(9) for a categorical exclusion from the requirement to perform an Environmental Assessment or Environmental Impact Statement.

VI. Conclusion

The requested license amendment proposes an increase in the MNGP Operating License maximum power level from 1670 megawatts thermal to 1775 megawatts thermal and includes the supporting changes to the MNGP Technical Specifications. The proposed changes have been evaluated against the criteria of 10CFR50.92, and the proposed changes involve no significant hazards considerations. In addition, an environmental evaluation has been performed to evaluate the effect of the proposed license amendment on the human environment. The environmental evaluation has found that the proposed action does not have any individual or cumulative adverse effect on the human environment and that the proposed action may satisfy the criteria of 10CFR51.22(c)(9) for categorical exclusion from environmental review.

The evaluations performed to support MNGP Power Rerate have found that the power rerate implementation can be performed within the existing capabilities of the installed facility. MNGP Power Rerate implementation does not require the construction of any new systems, buildings, roadways, railroad spurs or transmission lines. Minor system modifications are to be performed to enhance the capacities and capabilities of installed plant systems. Exhibit [illegible] "Summary of Plant Modifications for Power Rerate Implementation," provides a description of the planned plant modifications to be performed with implementation of MNGP Power Rerate. None of these planned modifications represent significant material changes to the facility. These modifications are to be implemented under the provisions of 10CFR50.59.