Xcel Energy®

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

March 11, 2013

L-MT-13-010
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License No. DPR-22

License Amendment Request: Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

Pursuant to 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM), doing business as Xcel Energy, Inc., proposes to reduce the reactor steam dome pressure specified within Reactor Core Safety Limits Specification 2.1.1, in the Technical Specifications (TSs). This change resolves a 10 CFR Part 21 condition concerning a potential to momentarily violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) transient.

U.S. Nuclear Regulatory Commission (NRC) approved critical power correlations have evolved. Some have a lower-bound pressure, of the approved pressure range for the correlation, that are significantly below the 785 psig reactor steam dome pressure specified in the safety limits. NSPM proposes to reduce the reactor steam dome pressure to align with the lower-bound pressure associated with the General Electric (GE) GEXL14 critical power correlation. This correlation applies to the GE14 fuel comprising the Monticello Nuclear Generating Plant (MNGP) core.

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the Technical Specification pages indicating the proposed changes. Enclosure 3 provides a copy of the associated draft marked-up Technical Specification Bases pages for information.

The NSPM requests approval of the proposed license amendment by one year from the date of submittal, with an implementation period of 90 days.

The MNGP Plant Operations Review Committee has reviewed this application. In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Minnesota Official.

A001
NRR

Should you have questions regarding this letter, please contact Mr. Richard Loeffler at (763) 295-1247.

This license amendment request has been evaluated and has no impact on the pending Extended Power Uprate and Maximum Extended Load Line Limit Analysis Plus (MELLLA+) license amendment requests currently under NRC review.

Summary of Commitments

This letter proposes no new commitments and does not revise any existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 11, 2013.

John Grubb

John C. Grubb
Plant Manager, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

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1.0 INTRODUCTION

In 2005, General Electric (GE) submitted a 10 CFR Part 21 notification, SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit" (Reference 1) to the U.S. Nuclear Regulatory Commission (NRC). This Part 21 notification identified that applying newer computer analysis codes that a Pressure Regulator Failure Maximum Demand (Open) transient could result in a condition where the reactor steam dome pressure could potentially, momentarily, decrease below 785 psig while thermal power was above 25 percent of rated thermal power (RTP). This condition would violate Technical Specification (TS) Reactor Core Safety Limit 2.1.1.1.

Some advanced fuel designs have an NRC approved critical power correlation with a lower-bound pressure significantly below the 785 psig reactor steam dome pressure specified in TS Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. Northern States Power Company – Minnesota (NSPM) proposes to utilize this fact and reduce the reactor steam dome pressure consistent with the approved lower-bound pressure for the critical power correlation for the GE14 fuel comprising the Monticello Nuclear Generating Plant (MNGP) core. GE14 fuel utilizes the GEXL14 critical power correlation, with an approved pressure range from 700 to 1400 psia. Revising the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to 686 psig resolves this 10 CFR Part 21 condition concerning the potential to violate a safety limit during a Pressure Regulator Failure Maximum Demand (Open) transient.

2.0 PROPOSED CHANGE

Reduce the reactor steam dome pressure specified within Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 from 785 to 686 psig. The Reactor Core Safety Limits would then read:

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

A mark-up of the proposed changes to the TS Reactor Core Safety Limits is provided in Enclosure 2. Enclosure 3 provides a copy of the associated draft TS Bases pages. The Bases changes will be issued in accordance with MNGP Specification 5.5.9, "Technical Specification TS Bases Control Program," following approval of the LAR.

3.0 HISTORY

On March 29, 2005, GE submitted a 10 CFR Part 21 notification (Reference 1) identifying that as a result of applying improved methodologies for licensing basis transient analyses it had been identified that an anticipated operational occurrence, the Pressure Regulator Failure Maximum Demand (Open), had been identified as an event in which Reactor Core Safety Limit 2.1.1.1 could potentially be violated. During the transient the expected sequence of events predicted by the computer models could potentially change, and based upon this, as described later herein, the reactor steam dome pressure could potentially, momentarily, decrease below 785 psig while thermal power was above 25 percent of RTP, violating Reactor Core Safety Limit 2.1.1.1.

GE indicated that the approved model has evolved from REDY, to ODYN, to the TRACG computer codes. Reactor depressurization transients, such as Pressure Regulator Failure Maximum Demand (Open) are non-limiting for fuel cladding integrity because Critical Power Ratio (CPR) increases during the event, and are not typically included in the scope of reload evaluations. Recent investigations by GE have determined that REDY, ODYN, and TRACG all show the CPR increasing during the transient and hence fuel cladding integrity is not threatened,[1] and that the difference in reactor level swell predicted by REDY, versus ODYN and TRACG, can impact the predicted plant response to the Pressure Regulator Failure Maximum Demand (Open).

GE indicated within the 10 CFR Part 21 notification letter that compensatory actions that would have appropriately mitigated this condition, but did not have other potentially deleterious effects, had not been identified. While this condition had been determined by GE to not involve an actual safety hazard, the potential for violation of a Reactor Core Safety Limit had been identified and restoration to comply with the safety limit is required.

1. The Minimum Critical Power Ratio (MCPR) Safety Limit specified in Reactor Core Safety Limit 2.1.1.1 is established to protect fuel cladding integrity.

On July 18, 2006, the Technical Specifications Task Force (TSTF) submitted TSTF-495, Revision 0, "Bases Change to Address GE Part 21 SC05-03," (Reference 2) proposing a modification to the "Applicable Safety Analysis" portion of the Reactor Core Safety Limit TS Bases (B 2.1.1). This change proposed to clarify that the safety limit was considered not to apply to momentary depressurization transients. On August 27, 2007, the NRC issued a letter to the TSTF (Reference 3) denying this proposed change to the TS Bases.

Since then, several approaches to resolve this situation had been considered at periodic Boiling Water Reactor Owners Group (BWROG) Licensing Committee meetings, but not adopted because a generic approach, applicable to all BWROG members and fuel vendors, could not be identified. It was recognized, however, that the approach described herein, to revise the reactor steam dome pressure consistent with the NRC approved pressure range of CPR correlations for advanced fuel designs would become increasingly viable as plants transitioned to these newer designs.

4.0 BACKGROUND

A discussion providing background on the Reactor Core Safety Limits and a summary of the Pressure Regulator Failure Maximum Demand (Open) transient scenario considering the change in computer analysis codes is provided below.

4.1 Background on the Reactor Core Safety Limits

TS Safety Limits are specified to ensure that specified acceptable fuel design limits (SAFDLS) are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. The Reactor Core Safety Limits are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the Safety Limits are not exceeded.

The Boiling Water Reactor (BWR) core is protected from the type of fuel failure that could occur during the Onset of Transition Boiling (OTB) by a combination of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. Reactor Core Safety Limit 2.1.1.1 states when the reactor steam dome pressure is less than 785 psig or when core flow is less than 10 percent of rated core flow, reactor thermal power shall be less than a specified value, usually 25 percent of RTP. When reactor pressure and core flow are greater than these specified values, Reactor Core Safety Limit 2.1.1.2 prohibits operation with a MCPR Safety Limit less than the value specified to prevent fuel cladding damage that could occur when a fuel assembly experiences the OTB.

For operation at low pressures or low flows, such as during startup, an alternate basis is used, as discussed in Section B 2.1.1 of the TS Bases, to provide fuel cladding integrity protection. Reactor Core Safety Limit 2.1.1.1 was introduced to preclude the need for CPR calculations when reactor steam dome pressure is less than 785 psig. Twenty-five (25) percent of RTP[2] in Reactor Core Safety Limit 2.1.1.1 was selected to ensure that reactor power would remain well below the fuel assembly critical power for the conditions in which CPR calculations are not performed. Consequently, below this value for reactor steam dome pressure and core flow, Reactor Core Safety Limit 2.1.1.1 restricts operation to less than or equal to 25 percent of RTP where the OTB conditions should not occur.

4.2 Pressure Regulator Failure Maximum Demand (Open) Transient Analysis Background

The GE Part 21 report describes a revised accident analysis scenario for the Pressure Regulator Failure Maximum Demand (Open) transient. A change in the predicted series of events for this transient was identified based upon a change in computer codes and the predicted results of this event.

Previous evaluations using the REDY computer code model indicated the transient would be terminated by direct turbine trip and subsequent reactor scram resulting from the reactor water level swell following the event. Specifically, for the postulated event the pressure regulator system fails in such a manner that a demand occurs to open the turbine steam admission valves, i.e., turbine stop valves (TSVs), turbine control valves, and turbine bypass valves. As a result, the reactor depressurization causes the formation of voids within the reactor core. The core voiding increases the reactor water level until the level reaches the main turbine trip (level) setpoint. The turbine trips, in turn sending a direct signal (via the TSV position switches) to the reactor protection system (RPS) resulting in the reactor automatically shutting down, terminating the transient.

A somewhat different series of events is predicted when the event was analyzed with improved transient models. The transient occurs as before and the reactor depressurizes; however, the reactor level does not swell to the setpoint to cause a main turbine trip. Level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. In which case the depressurization could be terminated by Main Steam Isolation Valve (MSIV) closure at the low-pressure isolation setpoint (LPIS).[3] This results in the transient not being terminated as quickly as the earlier

2. This is a plant-specific value. Plants that have received a power uprate may have a value specified less than 25 percent of RTP.
3. The Main Steam Line Pressure – Low Function (Function 1.b in TS Table 3.3.6.1-1) for MNGP corresponds to LPIS in the 10 CFR Part 21 notification.

methods predicted. Reactor depressurization continues to occur until the pressure decreases to the MSIV closure (in MODE 1) containment isolation signal setpoint. The MSIV closure is a direct input, via position switches, to the RPS. The reactor scrams and the transient is terminated.

However, under this series of events, the delay in termination of the transient introduces the possibility for reactor pressure to dip below 785 psig before the reactor has been shutdown. Depending upon the plant specific response to a Pressure Regulator Failure Maximum Demand (Open) event, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 25 percent of RTP, which would exceed the conditions in Reactor Core Safety Limit 2.1.1.1. This indicates that Reactor Core Safety Limit 2.1.1.1 is overly conservative with respect to this event, because during this event CPR continues to increase and therefore does not threaten fuel cladding integrity, but the pressure decrease could result in exceeding the value specified in the safety limit specification, while having no actual safety significance.

5.0 TECHNICAL ANALYSIS

The purpose of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is to protect fuel cladding integrity. The fuel cladding integrity safety limit (MCPR Safety Limit) is defined as the CPR in the limiting fuel assembly for which more than 99.9 percent of the fuel rods in the core are expected to avoid the OTB, considering the power distribution within the core and all uncertainties. The safety limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. It is determined using a statistical model that combines the uncertainties in operating parameters and procedures used to calculate critical power.

The probability of the occurrence of OTB is determined using approved critical power correlations. Each fuel vendor has developed correlations valid over specified pressure and flow ranges (mass flow rates) that are approved by the NRC. Some advanced fuel designs critical power correlations have received NRC approval down to a lower pressure than those approved previously. The lower-bound of the extended pressure ranges for these advanced fuel designs can be used to establish a lower reactor steam dome pressure than the 785 psig value currently specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 as described in the following sections. NSPM proposes to utilize the fact that the GE14 fuel, comprising the MNGP core, utilizes the GEXL14 critical power correlation, which has an approved pressure range from 700 to

1400 psia.[4] Revising the Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 reactor steam dome pressure from 785 to 686 psig resolves the reported 10 CFR Part 21 condition concerning the potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) transient.

5.1 Discussion of the Applicability of Various BWR Critical Power Correlations

Two critical power correlations are recognized in the BWR TS NUREGs, i.e., NUREG-1433 and NUREG-1434 (References 4 and 5, respectively). Since their establishment, critical power correlations have continued to evolve, and have been approved by the NRC. Some of these correlations represent extensions to existing correlations, e.g., GEXL14, and are only applicable to specific fuel designs, others are more generally applicable. Any of these NRC approved critical power correlations could have been used to establish a different reactor steam dome pressure than the 785 psig specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. While not intended to be inclusive, the applicability of some of these other critical power correlations is summarized below.

5.1.1 TS Bases Discussion of the BWR Fuel Cladding Integrity Safety Limit

NUREGs-1433 and 1434 contain and summarize two fuel vendors critical power correlations for BWR fuel. The analysis basis for these fuel vendor critical power correlations, i.e., the General Electric GEXL and Areva XN-3[5] correlations, are valid for performance of critical power calculations within the regions stated below:

- 2.1.1.1a – Fuel Cladding Integrity General Electric Company Fuel

  The GE critical power correlations are applicable for all critical power calculations at pressures greater than or equal to 785 psig[6] and core flows greater than or equal to 10 percent of rated flow.

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4. In accordance with 10 CFR 50.59, only fuel which has an NRC approved CPR correlation with a lower-bound pressure less than or equal to the reactor steam dome pressure specified in the safety limit may be loaded into the core.
5. The ANF XN-3 correlation listed in the BWR TS NUREGs is now Areva.
6. The GEXL critical power correlation was established as applicable for the stated pressure and flows due to limited test data then available at low pressure and flow conditions.

- 2.1.1.1b – Fuel Cladding Integrity Advanced Nuclear Fuel Corporation Fuel

  The XN-3 correlation is valid for critical power calculations at pressures greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 lb/hr-ft^2.

It has not been necessary for licensees to change the value of the reactor steam dome pressure, 785 psig, specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 in the past. This is because use of Areva XN-3 critical power correlation (580 psig) provided additional margin to the reactor steam dome pressure of 785 psig specified in the safety limit.

5.1.2 <u>SPCB Critical Power Correlation</u>

The analysis basis for the Areva SPCB critical power correlation is valid for performance of critical power calculations within the region stated below:

- The SPCB correlation (Reference 6) is valid for critical power calculations at pressures greater than 571.4 psia and bundle mass fluxes greater than 0.087×10^6 lb/hr-ft^2.

As indicated previously, it has not been necessary for licensees to change the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. This is due to the fact that use of a critical power correlation such as SPCB (571.4 psia -14.7 psia = 556.7 psig) provided additional margin to the reactor steam dome pressure of 785 psig specified in the safety limit.

5.1.3 <u>Application of the NRC Approved GEXL14 Critical Power Ratio Correlation</u>

On August 3, 2007, a final safety evaluation (SE) (Reference 7) was issued approving the GEXL14 critical power correlation which applies to GE14 fuel. As described in the associated licensing topical report and NRC SE, the pressure range over which the GEXL14 correlation is NRC approved for performance of critical power calculations is from 700 psia to 1400 psia. The reactor steam dome pressure of 686 psig is established from the lower-bound pressure (686 psig ≅ 685.3 psig = 700.0 psia - 14.7 psia).

- The GEXL14 correlation for GE14 fuel is valid for critical power calculations at pressures from 700 psia to 1400 psia and bundle mass flux from 0.1 x 10^6 to 1.8 x 10^6 lb/hr-ft^2.

As discussed herein, NSPM proposes to revise the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to reflect application of the GEXL14 critical power correlation to resolve the 10 CFR Part 21 condition.

5.2 No Impact on the Main Steam Line Pressure – Low Function

The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure. The Allowable Value for Main Steam Line Pressure – Low (TS Table 3.3.6.1-1, Function 1.b) for the MNGP is greater than or equal to 815 psig. The Nominal Trip Setpoint for this function is 840 psig and the Analytical Limit is 809 psig.

For the pressure regulator failure event, closure of the Main Steam Isolation Valves (MSIVs) ensures that the reactor pressure vessel temperature change limit (100°F/hr) is not reached. Also, as discussed in the TS Bases, this Function supports actions to ensure that Reactor Core Safety Limit 2.1.1.1 is not exceeded. This Function is described as closing the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to less than 25 percent of RTP. The proposed change to reduce the reactor steam dome pressure from 785 psig to 686 psig does not change the function of the Main Steam Line Pressure – Low scram, but does increase the margin between the setpoint and the safety limit the setpoint is protecting.

No changes are required or proposed to any instrumentation settings associated with the Main Steam Line Pressure – Low Function, including the TS Allowable Value. The TS Bases description for this function is revised to indicate the change in reactor steam dome pressure.

The trip on low main steam line pressure will occur as previously specified, at the assumed instrument settings discussed above. These Main Steam Line Pressure – Low Function setpoints provide added assurance that with the revised reactor steam dome pressure of 686 psig, that Reactor Core Safety Limit 2.1.1.1 would not be violated under realistic conditions.

5.3 Application of these Extended Pressure Ranges to Resolution of the 10 CFR Part 21 Concerning the Potential for Violation of Reactor Core Safety Limit 2.1.1.1 for a Pressure Regulator Failure Maximum Demand (Open) Transient

As discussed previously, each fuel vendor has critical power correlation(s) which are valid over established pressure ranges and flows (mass flow rates), approved by the NRC, which may or may not be fuel design specific. These critical power correlations have become increasingly fuel design dependent as advanced fuel designs evolved. This has resulted in an extension of the NRC approved pressure range to lower pressures as additional test data became available to demonstrate the validity of revised or new correlation(s) for performance of critical power calculations.

These reduced lower-bound pressures associated with the newer critical power correlations, as discussed previously, can be utilized to reduce the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2, consistent with the NRC approved pressure range for these correlation(s). Lowering the reactor steam dome pressure in this fashion provides margin to ensure Reactor Core Safety Limit 2.1.1.1 is not violated and resolves this 10 CFR Part 21 issue involving a potential to exceed the low pressure TS Safety Limit during a Pressure Regulator Failure Maximum Demand (Open) transient.

This method maintains the same approach currently reflected in the TS to establish the reactor steam dome pressure within the Safety Limits specification. As implied above, a different correlation, e.g., the Areva XN-3, Areva SPCB, or the GEXL14, could have been used to establish a different reactor steam dome pressure from the 785 psig value initially established by GEXL in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2.

This proposed change to the Reactor Core Safety Limits continues to ensure that a valid CPR calculation is performed for the AOOs described in the USAR, including the Pressure Regulator Failure Maximum Demand (Open) transient. NSPM has determined that with the value of 686 psig proposed for the reactor steam dome pressure, that a Pressure Regulator Failure Maximum Demand (Open) transient would not result in a violation of Reactor Core Safety Limit 2.1.1.1. Since this approach follows, and is consistent with, the way the reactor steam dome pressure has been established, and valid CPR calculations will continue to be performed, it is a safe and appropriate method to address the 10 CFR Part 21 condition.

5.4 Conclusion

In summary, it has been identified that a Pressure Regulator Failure Maximum Demand (Open) transient could potentially violate Reactor Core Safety Limit 2.1.1.1. Reducing the reactor steam dome pressure, specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2, from the current value of 785 psig to 686 psig in accordance with the NRC approved lower-bound pressure of the GEXL14 critical power correlation (GE14 fuel) eliminates the potential for this event. This TS change resolves a long-standing industry issue, the 10 CFR Part 21 condition, which identified that during a Pressure Regulator Failure Maximum Demand (Open) transient Reactor Core SL 2.1.1.1 might be violated.

6.0 REGULATORY ANALYSIS

6.1 Applicable Regulatory Requirements

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36, the TSs will include Safety Limits for nuclear reactors which are stated to be "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down."

The proposed TS change revises the reactor steam dome pressure stated in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to remove the potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) transient. The MNGP was designed largely before the publishing of the 70 General Design Criteria (GDC) for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. As such, the MNGP was not licensed to the 10 CFR 50 Appendix A, General Design Criteria (GDC).

The MNGP USAR, Section 1.2, lists the Principal Design Criteria (PDCs) for the design, construction and operation of the plant. MNGP USAR Appendix E provides a plant comparative evaluation with the proposed AEC 70 design criteria. It was concluded that the plant conforms to the intent of the GDCs. The applicable GDC and PDC are discussed below.

- PDC 1.2.2 – Reactor Core

  i. The reactor core and associated systems are designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

The applicable 70 draft AEC General Design Criterion (AEC-GDC) is:

- AEC-GDC 6 – Reactor Core Design (Category A)

  The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of off-site power.

Compliance with the fuel licensing criteria of 10 CFR 50 Appendix A, General Design Criteria (GDC) 10, "Reactor design," is achieved by preventing the violation of fuel design limits. GDC 10 states:

- GDC 10 – Reactor Design

  The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The NSPM has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. As long as the core pressure and flow are within the range of validity of the specified critical power correlation, in this case the GEXL14 correlation, the proposed reactor steam dome pressure change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 will continue to ensure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC 10 regarding acceptable fuel design limits and continues to assure that the underlying criteria of the safety limit is met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, is unaffected.

6.2 Precedent

On December 19, 2012, PPL Susquehanna, LLC submitted a license amendment request to revise the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 for the Susquehanna Steam Electric Station, Units 1 and 2 (Reference 8) to resolve this 10 CFR Part 21 condition.

6.3 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM) requests an amendment to Renewed Facility Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP) to reduce the reactor steam dome pressure specified within Technical Specification Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to resolve a 10 CFR 21 condition in which a postulated Pressure Regulator Failure Maximum Demand transient might result in a momentary violation of Reactor Core Safety Limit 2.1.1.1.

The NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the MNGP in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change to the reactor steam dome pressure in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 does not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. The proposed change is in accordance with an NRC approved critical power correlation methodology and as such maintains required safety margins. The proposed change does not adversely affect accident initiators or precursors nor does it alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained.

The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not require any physical change to any plant SSCs nor does it require any change in systems or plant operations. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

There are no hardware changes nor are there any changes in the method by which any plant systems perform a safety function. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

The proposed change does not introduce any new accident precursors, nor does it involve any physical plant alterations or changes in the methods governing normal plant operation. Also, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis.

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

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. Evaluation of the 10 CFR Part 21 condition by General Electric determined that there was no decrease in the safety margin, the Minimum Critical Power Ratio improves during the transient, and therefore is not a threat to fuel cladding integrity.

The proposed change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is consistent with, and within the capabilities of the applicable NRC approved critical power correlation, and thus continues to ensure that valid critical power calculations are performed. No setpoints at which protective actions are initiated are altered by the proposed change. The proposed change does not alter the manner in which the safety limits are determined. This change is consistent with plant design and does not change the TS operability requirements; thus, previously evaluated accidents are not affected by this proposed change.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

7.0 ENVIRONMENTAL EVALUATION

The NSPM has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, or (ii) authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, the NSPM concludes pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

8.0 REFERENCES

1. GE letter to the NRC GENE SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," to the NRC informing them of this reportable condition pursuant to 10 CFR 21, dated 3/29/2005.

2. Technical Specifications Task Force letter (TSTF-06-20) transmitting TSTF-495, Revision 0, "Bases Change to Address GE Part 21 SC05-03," dated July 18, 2006.

3. NRC letter to the TSTF, Denial of TSTF-495, Revision 0, "Bases Change to Address GE Part 21 SC05-03, Docket No: PROJ0753 (TAC MD2672)," dated August 27, 2007.

4. NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4."

5. NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6."

6. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003 [now Areva].

7. NRC letter to GNF, "Final Safety Evaluation for Global Nuclear Fuel (GNF) Topical Report (TR) NEDC-32851P, Revision 2, "GEXL14 Correlation for GE14 Fuel" (TAC No. MD5486), dated August 3, 2007" (ADAMS Accession No. ML072080365).

8. Letter from J. M. Helsel, PPL Susquehanna, LLC, to the NRC, "Susquehanna Steam Electric Station, Proposed Amendment No. 312 to Unit License NPF-14 and Proposed Amendment No. 284 to Unit License NPF-22: Low pressure Safety Limit and Reference Changes," dated December 19, 2012.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

MARKED-UP PROPOSED TECHNICAL SPECIFICATION CHANGES

(1 page follows)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1332 psig.

2.2 SL VIOLATIONS

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

DRAFT TECHNICAL SPECIFICATION BASES PAGES

(FOR INFORMATION)

(5 pages follow)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

USAR Section 1.2.2 (Ref. 1) requires the reactor core and associated systems to be designed to accommodate plant operational transients or maneuvers that might be expected without compromising safety and without fuel damage. Therefore, SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

No Changes for Information

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical

BASES

BACKGROUND (continued)

reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this SL provides margin such that the SL will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

The approved pressure range (700 to 1400 psia) of the GEXL14 critical power correlation is applied to resolve a 10 CFR Part 21 condition concerning a potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) transient (Reference 5). Application of this correlation, which applies to the GE14 fuel in the core, allows reduction of the reactor steam dome pressure from 785 to 686 psig, precluding violation of the safety limit for this event. This change in reactor steam dome pressure was approved in Amendment __ (Reference 7).

2.1.1.1 Fuel Cladding Integrity

The GEXL14 critical power correlation is applicable for all critical power calculations at pressures ≥ 686 psig and core flows ≥ 10% of rated flow (Reference 6). For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.56 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test

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APPLICABLE SAFETY ANALYSES (continued)

data taken at pressures from 0 psig to 785 psig indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig or < 10% core flow is conservative.

2.1.1.2 MCPR

No Changes for Information

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 3 includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to

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APPLICABLE SAFETY ANALYSES (continued)

provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. USAR, Section 1.2.2.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).
3. NEDE-31152P, "General Electric Fuel Bundle Designs," Revision 8, April 2001.
4. 10 CFR 100.
5. GE Part 21 Notification SC05-03, “Potential to Exceed Low Pressure Technical Specification Safety Limit,” dated March 29, 2005.
6. NRC Letter to A. Lingenfelter (GNF), ‘Final Safety Evaluation for Global Nuclear Fuel (GNF) Topical Report (TR) NEDC-32851P, Revision 2, “GEXL14 Correlation for GE14 Fuel” (TAC No. MD5486),’ dated August 3, 2007.
7. Amendment __, “Issuance of Amendment Re: Reducing the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits [or whatever NRC amendment title is],” dated [Month] XX, 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 686 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure switches that are connected to the MSL header close to the turbine stop valves. The pressure switches are arranged such that, even though physically separated from each other, each pressure switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 3).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is one of the Functions assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.