ENCLOSURE 2

AREVA REPORT NO. ANP-3286NP, Revision 0

RESPONSES TO RAI FROM SRXB ON MNGP TRANSITION TO AREVA FUEL

NON-PROPRIETARY

18 pages follow

ANP-3286NP
Revision 0

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January 2014

AREVA Inc.

AREVA Inc.

ANP-3286NP
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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial issue

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Figures

1.0 Introduction

In Reference 1, Northern States Power Company - a Minnesota corporation, doing business as Xcel Energy, requested an amendment to the operating license and facility Technical Specifications for the Monticello Nuclear Generating Plant (MNGP). The amendment, if approved, would allow for a transition to the AREVA ATRIUM 10XM fuel design. The amendment would also allow the implementation of AREVA safety analysis methods.

The U.S. Nuclear Regulatory Commission (NRC) staff in the Reactor Systems Branch (SRXB) is reviewing the safety analyses for anticipated operational occurrences (AOOs), design basis accidents (DBAs), and special events. The SRXB staff has determined that additional information is required to complete its review (Reference 2). The Requests for Additional Information (RAI) and the AREVA responses are attached.

These responses are provided so Xcel Energy can provide a complete set of responses to the NRC by combining the AREVA responses with the responses being prepared by Xcel Energy.

2.0 RAIs and Responses

SRXB RAI-1) *In document ANP-3211(NP), "Monticello EPU LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel," Figure 6.22, "Limiting TLO Recirculation Line Break Cladding Temperatures," the trace for Peak Centerline Temperature (PCT) rod indicates a temperature excursion (i.e., a "blip") that occurs between 125-150 seconds.*

Please explain the cause for this excursion in sufficient detail to confirm the validity of the result.

AREVA Response

The "blip" occurs at 129.9 seconds. This sudden increase in clad temperature is due to clad failure of the PCT rod at the plane of interest. The heat generated during metal water reaction is calculated using the Baker Just model described in 10 CFR 50 Appendix K. The reaction rate is assumed not to be steam limited. The heat generated by the metal-water reaction increases as the clad temperature increases and decreases as the oxide thickness increases. The inside of the clad is assumed to have no oxide when the clad fails, so the increase in heat generated is maximum at the time of failure and decreases as the oxide thickness increases.

SRXB RAI-2) *Table 6.3 of ANP-3211(P) provides break spectrum results for the MNGP Emergency Core Cooling System (ECCS) evaluation performed using AREVA's EXEM BWR-2000 evaluation model. The break spectrum results are shown for the assumption that the low pressure coolant injection (LPCI) injection valve is the single failure. However, the failure of a low-pressure injection system may not be the most limiting for smaller breaks, where low pressure injection systems would not provide the most significant sources of emergency core coolant early in the transient. In fact, Table 6.4 of ANP-3211(P) identifies different limiting single failures and PCTs for, among others, small break analyses.*

*Please provide break spectrum results for the limiting small break single failure.**

AREVA Response

As mentioned in the RAI, for small breaks the low pressure injection systems are not the most important sources of emergency core coolant. Instead, the high-pressure coolant injection system (HPCI) is more important for small breaks. Table 5.1 of ANP-3211(P) lists the available ECCS for assumed single failures. Cases designated SF-BATT and SF-HPCI are potentially limiting single failures for small breaks since HPCI is disabled by the postulated failures. Since SF-BATT has fewer Low Pressure Core Spray (LPCS) and LPCI pumps than SF-HPCI, only SF-BATT is analyzed.

The PCT results in Tables 6.3 and 6.4 of ANP-3211(P) show that for Monticello the AREVA LOCA methodology calculates a second peak in PCT for small breaks of approximately 0.1 ft^2. The break spectrum results for this small break are summarized below:

* RAI was formulated largely from information contained in ANP-3211(NP). The referenced tables, however, were redacted from the non-proprietary report. NRC staff verified non-proprietary nature of information not contained in non-proprietary copy of report.

TLO Results for 102% Power [ ]				
Single Failure	PCT (°F) for 0.10 ft^2 Break			
	Pump Suction		Pump Discharge	
	Mid-Peaked	Top-Peaked	Mid-Peaked	Top-Peaked
SF-LPCI	[			
SF-BATT				
SF-ADS				]

TLO Results for 102% Power [ ]				
Single Failure	PCT (°F) for 0.10 ft^2 Break			
	Pump Suction		Pump Discharge	
	Mid-Peaked	Top-Peaked	Mid-Peaked	Top-Peaked
SF-LPCI	[			
SF-BATT				
SF-ADS				]

The above results established that for small breaks, SF-BATT was the limiting single failure and the discharge side of the recirculation pump was the limiting break location. Based on these results, additional small break areas were analyzed for SF-BATT and pump discharge and the most limiting small break areas are reported in Table 6.4 of ANP-3211(P). The break spectrum calculations performed to identify the limiting small break size are summarized below:

SF-BATT				
Break Size and Type	[ ] Pump Discharge		[ ] Pump Discharge	
	Mid-Peaked	Top-Peaked	Mid-Peaked	Top-Peaked
[				]

SRXB RAI-3) *A top-peaked axial power shape places the hot node at a higher elevation. Although, during a large break with a relatively fast blowdown, the time of hot node uncovery may be insignificant as a function of height, it should take longer to achieve a stable quench in a hot node at a higher elevation.*

Please explain why the mid-peaked power shape is limiting in terms of PCT.

AREVA Response

We agree; a node higher in the core should uncover sooner and reflood later than a node lower in the core. Our experience is that these characteristics can be offset by other factors and therefore AREVA analyzes both mid-peaked and top-peaked axial power profiles. One reason mid-peaked breaks may have a higher PCT is the power in the hot assembly is usually higher for a mid-peaked axial than for a top-peaked axial when both assemblies are operating at the same MCPR.

Table 6.3 in ANP-3211(P) shows mid-peaked axial shapes are limiting for some break characteristics and top-peaked axial shapes are limiting for other break characteristics. The results for SF-BATT provided in response to RAI-2 show the top-peaked axial shapes are often more limiting than the mid-peaked axial shapes.

SRXB RAI-4) *In document ANP-3213(NP), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)," Section 4.2, "Safety Limit MCPR Analysis," states, "The radial power uncertainty used in the analysis includes the effects of up to 1 traversing incore probe (TIP) machine out-of-service or the equivalent number of TIP channels and/or up to 50% of the LPRMs [local power range monitors] out-of-service and a 1200 effective full-power hour (EFPH) LPRM calibration interval." Currently, MNGP TS Surveillance Requirement (SR) 3.3.1.1.6 requires LPRM calibration every 2000 EFPH. According to the NRC's records, this surveillance interval will be revised to 1770 EFPH upon implementation of the requested extended power uprate amendment.*

4.a) *Please explain how the assumed calibration interval and the SR align in compliance with 10 CFR 50.36(c)(3), "Surveillance Requirements," which states that, "surveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."*

4.b) *Please explain how the radial power uncertainty assumed in the SLMCPR analysis accounts for SR 3.0.2, which states, in part, that "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the frequency is met."*

AREVA Response

Xcel Energy will provide this response.

SRXB RAI-5) *Enclosure 1, "Evaluation of the Proposed Change," to the July 15, 2013, request letter, states, in Section 2.3, that "TS 5.6.3 will be revised to add appropriate NRC-approved AREVA analytical methods." As proposed, a specific applicability is provided neither for methods that are proposed for retention, nor for those proposed for addition.*

Please provide additional information justifying the retention of the existing references, including the applicability of these references and the purpose that they will continue to serve in developing the cycle-specific Core Operating Limits Report.

AREVA Response

Xcel Energy will provide this response.

SRXB RAI-6) *The initial dome pressure for the ASME overpressure analysis was assumed to be at its maximum value. Justify the acceptability of this assumption in light of the fact that, at a lower pressure condition at the same power level, the initial steady state void fraction could be higher, leading to a greater void collapse and resultant flux spike.*

AREVA Response

We agree, for the reasons mentioned in the RAI, a lower initial dome pressure may experience a larger pressure increase (peak pressure – initial pressure) during the event. However, a lower initial dome pressure also has more margin to the pressure limit. AREVA calculations have shown the increase in the pressure rise during the event does not offset the increase in initial pressure margin. For example, analyses for another BWR show an initial pressure 20 psi lower had a 15 psi larger pressure rise but the peak pressure was still 5 psi lower when compared to the case initiated at a higher initial pressure. Experience with AREVA methods indicates this trend is applicable for Monticello.

SRXB RAI-7) *The discussion regarding the ASME Overpressure Analysis contained in Section 7.1 of ANP-3213(NP) indicates that the effects of various assumptions to increase the overall conservatism of the analysis have been approximated using single effect sensitivity studies, as described in Appendix E to ANP-3224(NP), and added to the total result for the predicted peak pressure. The discussion in ANP-3224(NP) then refers to AREVA letter NRC:12:023, for justification that separate consideration of the effects of the conservative assumptions is more conservative than an integral analysis. The NRC:12:023 letter is based on a study that was performed using, apparently, some type of representative plant.*

Please demonstrate that this study is applicable to Monticello by providing information that shows that the sequence of events between the two plants is sufficiently similar as to capture similar effects from the phenomena for which the COTRANSA2 models have been corrected. Key parameters to consider may include the time, following the initiating event, of (1) key equipment initiation, (2) maximum neutron flux, (3) reactor trip, (4) peak heat flux, and (5) minimum critical power ratio.

AREVA Response

The justification that separate consideration of the effects is more conservative than an integral analysis was performed for a representative plant at EPU conditions. The initial operating conditions were 102% of EPU rated power and 105% of rated core flow. The timing of the requested key parameters for the representative analysis in NRC:12:023 is compared to Monticello analyses (provided in Reference 3). MCPR is not compared since the ASME event is a special analysis performed to evaluate peak pressure criteria and MCPR is not calculated. The timing for Monticello is summarized in the following table for the same initial operating conditions as the representative analysis and for the initial conditions that were slightly more limiting for Monticello and therefore reported in Reference 3 (102% of EPU rated power and 99% of rated core flow). The comparison of key parameters demonstrates that Monticello timing is sufficiently similar for the conclusion to be applicable to Monticello.

Event description	Event time (sec.)		
	NRC:12:023 102%P / 105%F	Monticello 102%P / 105%F	Monticello 102%P / 99%F
Start of MSIV closure	0.0	0.0	0.0
Reactor scram on high neutron flux	3.147	2.937	2.941
Peak neutron flux	3.275	3.264	3.267
First SRU actuation	4.177	4.141	4.141
Peak heat flux	4.405	4.366	4.363

SRXB RAI-8) *Confirm that the ATWS Overpressurization Analyses discussed in Section 7.2 of ANP-3213(NP) were analyzed using COTRANSA2, in a manner largely accordant with NRC-approved methodology.*

If the analysis was not performed using COTRANSA2, please describe the codes and methods used to analyze the event in sufficient detail to permit the NRC staff to verify their acceptability.

<u>AREVA Response</u>

The ATWS Overpressurization analyses were performed using the COTRANSA2 code (Reference 3, Table 2.3). This code was approved for use to analyze pressurization type events similar to ATWS overpressure analysis. Since this event was not specifically mentioned in the COTRANSA2 topical report (Reference 4), NRC clarified that it is applicable for calculating the peak pressures (which occur early during an ATWS event) in Reference 5.

SRXB RAI-9) *Please provide Reference 35, "Potential Violation of Low Pressure Technical Specification Safety Limit," to ANP-3213(NP).*

<u>AREVA Response</u>

Xcel Energy response (this is Reference 1 in L-MT-13-10, March 11, 2013, "Licensing Amendment Request: Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits").

SRXB RAI-10) *Section 7.3, "Reactor Core Safety Limits – Low Pressure Safety Limit, Pressure Regulator Failed Open Event (PRFO)," of ANP-3213(NP) concludes that "The results of the analyses at various power/flow statepoints and cycle exposures showed that the lowest steam dome pressure that was reached before thermal power was ≤ 25% thermal power was 665 psia (650 psig)." However, the NRC staff determined that the basis for the pressure applicability of TS 2.1.1.1 is the applicability of the critical power correlation in use, and not necessarily the result of a system analysis.*

Please provide information that will permit the NRC staff to verify the assertion, per ANP-3213(NP), that "…this event poses no threat to thermal limits."

10.a) *Identify the lower pressure applicability bounds of the critical power correlations proposed for use.*

AREVA Response

Separate AREVA critical power correlations will be used for the ATRIUM 10XM fuel and the co-resident GE14 fuel.

The ATRIUM 10XM fuel has been analyzed and will be monitored using the ACE correlation (References 6 and 7). The low pressure bound is 290.8 psia, Reference 7, page 2-3.

The GE14 fuel has been analyzed and will be monitored with the SPCB/GE14 correlation. The SPCB/GE14 correlation is based on the SPCB correlation (Reference 8) using the methodology described in Reference 9. The low pressure bound for SPCB is 571.4 psia (Reference 8, page 1-2). The range of data used to construct additive constants for the Monticello GE14 fuel did not extend below 800 psia. However, the report addressing the applicability of AREVA method for Monticello (Reference 10, Appendix G) provides justification for a conservative implementation of SPCB/GE14 to a low pressure limit of 571.4 psia.

10.b) *Identify the statepoint for the limiting event, and for that event, provide plots of reactivity, core power, system pressure, and heat flux.*

AREVA Response

The limiting state point was identified to be 60% of rated power and 44% of rated core flow when the event is initiated from early exposures in the cycle.

The requested plots are provided below.

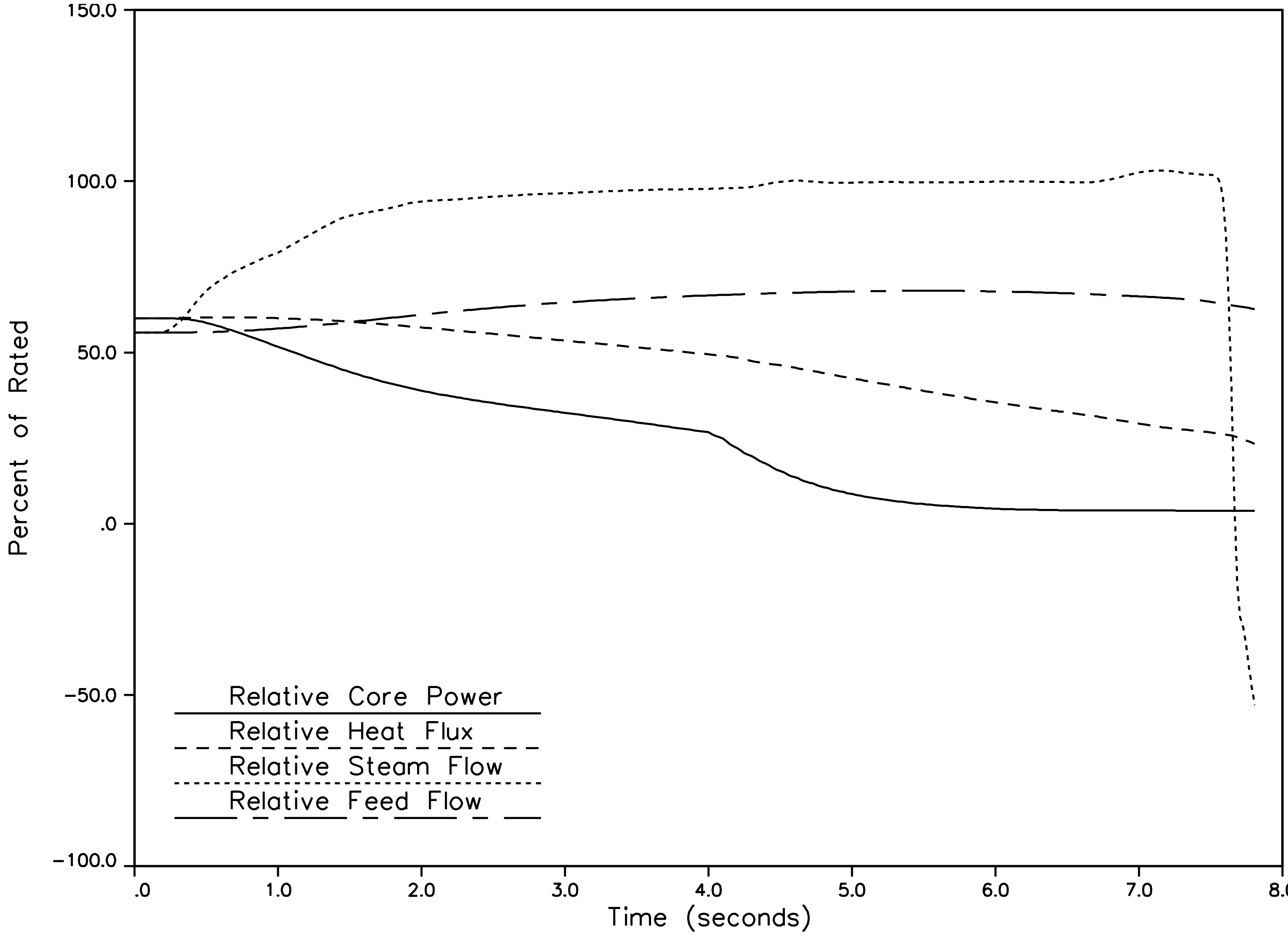

Figure 1 Key Parameters for Monticello Cycle 28 PRFO 60P/44F

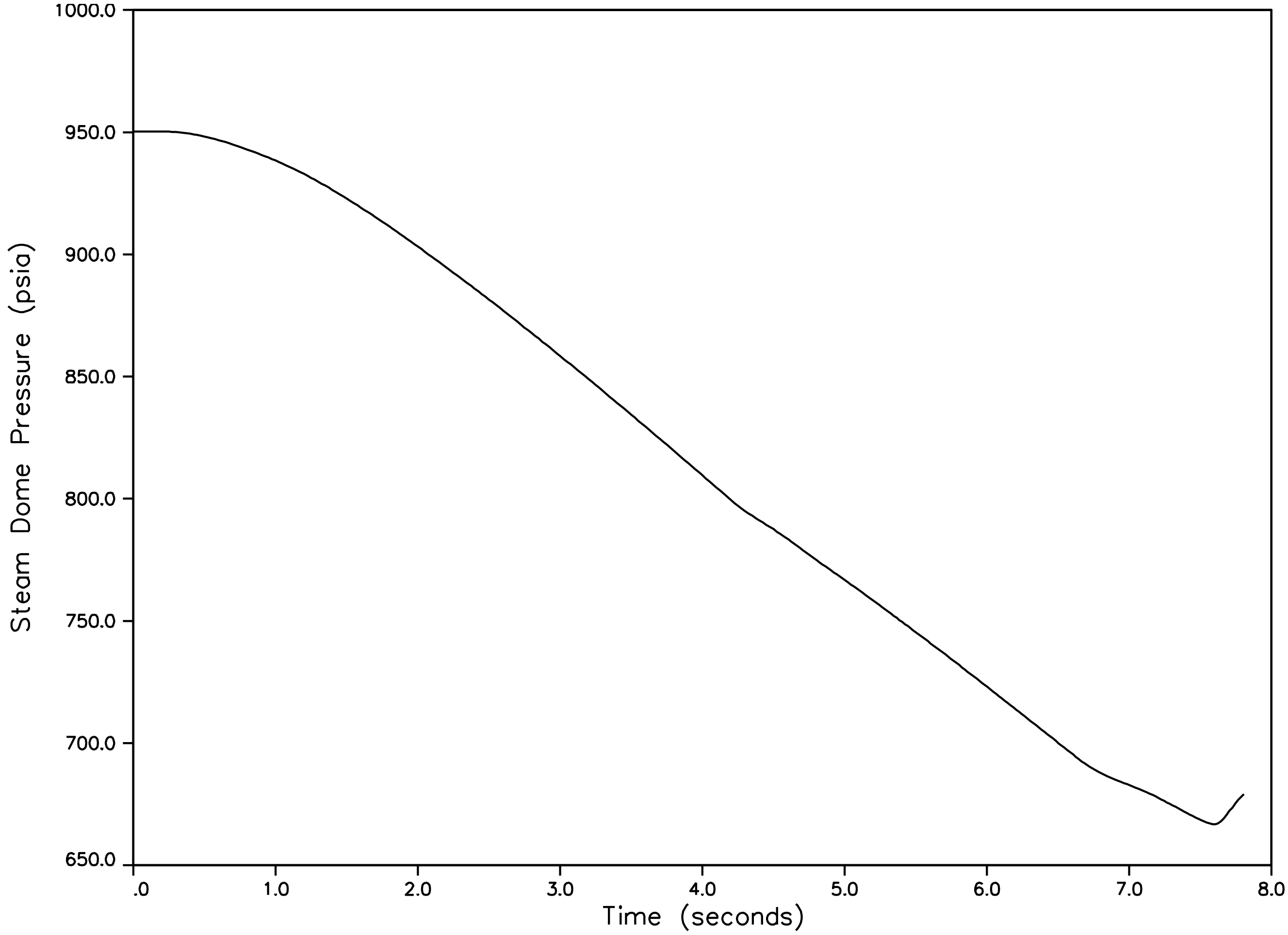

Figure 2 Dome Pressure for Monticello Cycle 28 PRFO 60P/44F

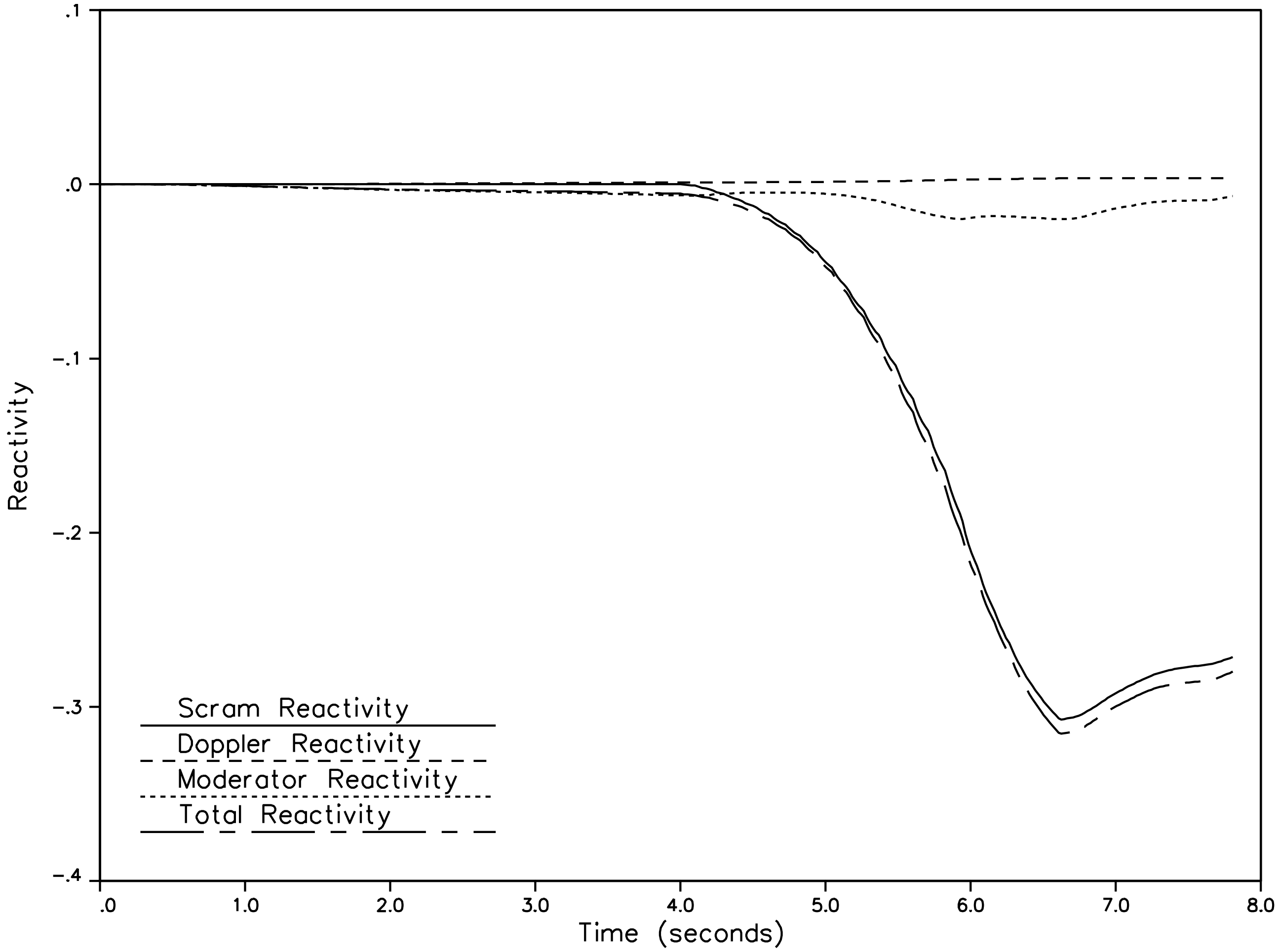

Figure 3 Reactivities for Monticello Cycle 28 PRFO 60P/44F

Notice that both the core power and the heat flux are decreasing throughout the depressurization event. The margin to thermal limits increases throughout the PRFO event.

10.c) *If the results show that the reactor coolant system tends to a state in which the critical power correlations are not valid and the core power exceeds 25%, explain how it was determined the event poses no threat to thermal limits.*

AREVA Response

The proposed change to Technical Specification 2.1 for the reactor low pressure limits reflects the low pressure bounds for the AREVA correlations detailed in response to RAI 10.a. Therefore, based on the transient results the thermal power will drop below 25% before the pressure drops below the correlation's low pressure bounds.

SRXB RAI-11) *The NRC has reviewed the CPR results for the core-wide transients provided in Chapter 5 of ANP-3213(NP), and determined that the results from the THERMEX methodology appear to identify a different set of limiting events than those determined using previous methodology and documented in Cycles 25 and 26 Supplemental Reload Licensing Reports.*

Please explain why this is the case.

AREVA Response

In support of the introduction of AREVA ATRIUM 10XM fuel and AREVA methods, AREVA reviewed the current licensing basis for Monticello. This is described in Section 2.0 of Reference 3. Therefore, the potentially limiting events that were considered were not limited to the events that were discussed in the THERMEX methodology. The AREVA review included a review of recent Monticello Supplemental Reload Licensing Reports. The set of limiting events identified in previous GNF analyses was included in the events AREVA analyzed in support of the fuel transition and the introduction of AREVA methodology. The results are slightly different due to different analysis inputs and the differences between AREVA and GNF methodologies. Overall, the limiting events are the same (inadvertent high-pressure coolant injection, feedwater controller failure, turbine trip with bypass and degraded scram, turbine trip without bypass, load rejection without bypass and single loop pump seizure).

3.0 References

1. License Amendment Request for Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methodology, July 15, 2013, MNGP L-MT-13-055, ML13200A185.

2. Monticello Nuclear Generating Plant – Request for Additional Information re: NRC Staff Review of AREVA Fuel Transition Licensing Amendment Request (TAC No. MF2479), December 18, 2013, ML13353A366.

3. ANP-3213(P) Revision 1, *Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)*, AREVA NP, June 2013.

4. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.

5. Letter, NRC to Siemens Power Corporation, "Siemens Power Corporation RE: Request for Concurrence on Safety Evaluation Report Clarifications (TAC NO. MA6160), May 31, 2000, ML003719373.

6. ANP-3138(P) Revision 0, *Monticello Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, August 2012.

7. ANP-10298PA Revision 0, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, March 2010.

8. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

9. EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.

10. ANP-3224P Revision 2, *Applicability of AREVA NP BWR Methods to Monticello*, AREVA NP, June 2013.

ENCLOSURE 3

AREVA ATRIUM 10XM FUEL TRANSITION

RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION

This enclosure provides a response from the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, to a request for additional information (RAI) provided by the Nuclear Regulatory Commission (NRC) in Reference 1. The NRC RAIs are in italics font and the NSPM responses are in normal font.

NRC Request SRXB RAI-4

In document ANP-3213(NP), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)," Section 4.2, "Safety Limit MCPR Analysis," states, "The radial power uncertainty used in the analysis includes the effects of up to 1 traversing incore probe (TIP) machine out-of-service or the equivalent number of TIP channels and/or up to 50% of the LPRMs [Local Power Range Monitors] out-of-service and a 1200 effective full-power hour (EFPH) LPRM calibration interval." Currently, MNGP [Monticello Nuclear Generating Plant] TS [Technical Specifications] Surveillance Requirement (SR) 3.3.1.1.6 requires LPRM [Local Power Range Monitor] calibration every 2000 EFPH [Effective Full Power Hours]. According to the NRC's records, this surveillance interval will be revised to 1770 EFPH upon implementation of the requested extended power uprate amendment.

4.a) Please explain how the assumed calibration interval and the SR [Surveillance Requirement] align in compliance with 10 CFR 50.36(c)(3), "Surveillance Requirements," which states that, "surveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

4.b) Please explain how the radial power uncertainty assumed in the SLMCPR [Safety Limit Minimum Critical Power Ratio] analysis accounts for SR 3.0.2, which states, in part, that "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the frequency is met."

NSPM Response

The RAI requires the following clarification:

The MNGP TS, SR 3.3.1.1.6, Frequency (surveillance interval) was previously defined as "2000 EFPH," but was subsequently revised under the extended power uprate (EPU) amendment (LA 176) to be "1000 megawatt days per ton,"(MWD/T) not "1770 EFPH" as indicated in the RAI. Therefore, the responses below use "1000 MWD/T" as the current required surveillance interval.

4.a The SR 3.3.1.1.6 calibration is performed by obtaining LPRM gain settings. LPRM gain settings are determined from the local flux profiles measured by the TIP System. This establishes the relative local flux profile for appropriate representative input to the Average Power Range Monitor (APRM) System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

Based on the full EPU power level conditions, an interval of 1000 MWD/T is equivalent to ~1115 EFPH. Therefore, the analyzed 1200 EFPH LPRM calibration interval bounds the TS SR calibration interval of 1000 MWD/T. Therefore, the LPRM calibration frequency of 1000 MWD/T meets the requirements of 10 CFR 50.36(c)(3) (Surveillance Requirements).

4.b SR 3.3.1.1.6 requires the LPRMs to be calibrated on frequency of 1000 megawatt days per ton. SR 3.0.2 applies to this SR and permits the surveillance frequency to be extended by up to 25% of the specified frequency.

The value of uncertainty in the SLMCPR analysis has not been explicitly related to the LPRM surveillance interval, so the analysis described in RAI response 4.a above, does not include the 25% grace period that is provided by SR 3.0.2 for the LPRM surveillance interval. The effect of this 25% grace period will be evaluated and supplemental information will be provided to the NRC in approximately three months. This supplement will include any changes to the SLMCPR analysis or other related analyses, as necessary.

NRC Request SRXB RAI-5

Enclosure 1, "Evaluation of the Proposed Change," to the July 15, 2013, request letter, states, in Section 2.3, that "TS 5.6.3 will be revised to add appropriate NRC-approved AREVA analytical methods." As proposed, a specific applicability is provided neither for methods that are proposed for retention, nor for those proposed for addition.

Please provide additional information justifying the retention of the existing references, including the applicability of these references and the purpose that they will continue to serve in developing the cycle-specific Core Operating Limits Report.

<u>NSPM Response</u>

The first element of this question deals with a concern that the license amendment request (LAR) did not provide justification to defend the "specific applicability" of the newly-proposed AREVA analytical methods listed in TS 5.6.3. As stated in the Section 1.0, Introduction, of ANP-3224P (Enclosure 6 of the LAR), the purpose of this report is to demonstrate that the licensing methodologies are applicable to operation of the MNGP including EPU conditions. The specific applicability of each document proposed for addition to TS 5.6.3 is addressed in ANP-3224P.

The second element of this question deals with a concern that additional justification is needed to defend the specific applicability of the legacy analytical methods that are proposed for retention in TS 5.6.3. In general terms, NSPM did not justify the applicability of the legacy analysis methodologies listed in TS 5.6.3 because the conditions and limitations of those methods inherently speak for themselves in terms of applicability, no matter what changes evolve in plant design, plant operations, or fuel design. Prior to use, the core reload evaluation process requires the confirmation that any analytical code used for the reload analysis be reviewed for applicability. Thus, no attempt was made to screen these legacy methods for the change to ATRIUM 10XM fuel design.

Please note that the scope of legacy analytical methods listed in TS 5.6.3 is reduced by the following factors:

- Two of the legacy methods shown in the LAR's TS markup (and now listed as "not used" in TS 5.6.3.b), were removed from the TS in MNGP license amendment 175. Thus, no further discussion of their applicability to AREVA ATRIUM 10XM fuel is required.
- Two of the legacy methods shown in the LAR's TS markup (listed as TS 5.6.3.b.4 and b.5) are generic to boiling water reactor fuels and do not exclude any particular fuel designs. Furthermore, these two methods are subject to removal pending approval of the license amendment for Maximum Extended Load Line Limit Plus (MELLLA+). Thus, no further discussion of their applicability to AREVA ATRIUM 10XM fuel is necessary.

Based on the above, the scope of concern is reduced to the analytical method described in TS 5.6.3.b.1, known as "General Electric Standard Application for Reactor Fuel (GESTAR)". This document contains a wide array of analytical methods and evaluations that embody General Electric fuels and other fuels in some cases. As discussed above, the applicability of any section of GESTAR speaks for itself, and will be confirmed each cycle reload that employs GESTAR. NSPM intends to use applicable portions of GESTAR as necessary to develop the Core Operating Limits Report (COLR), and expects to do so for the two or three operating cycles associated with transitioning to the ATRIUM 10XM fuel design. NSPM expects, at a minimum, that GESTAR will be necessary to evaluate the following during transition core reloads:

- Emergency Core Cooling System (ECCS) Loss of Coolant Accident (LOCA) analyses for legacy GE14 fuel, including break spectrum analysis and maximum axial planar linear heat generation rate (MAPLHGR) analysis.
- Fuel rod design limits for GE14 fuel (linear heat generation rate vs. exposure), including Mechanical Over Power (MOP) and Thermal Over Power (TOP).
- Anticipated Transient Without Scram (ATWS) analysis is performed using GE methodology, which AREVA shows to be bounding for operations with ATRIUM 10XM fuel.

Therefore, the retention of TS 5.6.3.b.1 provides the means to evaluate the GE14 fuel type for transition (or mixed) cores and generate core operating limits. It also serves as a means to retain certain analyses and evaluations that are performed on a generic basis to bound the ATRIUM 10XM fuel type.

NRC Request SRXB RAI-9

Please provide Reference 35, "Potential Violation of Low Pressure Technical Specification Safety Limit," to ANP-3213(NP).

NSPM Response

Reference 35 from ANP-3213(NP) is General Electric 10CFR Part 21 Communication, *Potential Violation of Low Pressure Technical Specification Safety Limit,* SC05-03, March 29, 2005. This document is included in Attachment 1 to this Enclosure.

References

1. Email from T. Beltz (NRC) to G. Adams (NSPM), "Monticello Nuclear Generating Plant - Request for Additional Information re: NRC Staff Review of AREVA Fuel Transition License Amendment Request (TAC MF2479)," dated December 18, 2013.

Attachment 1

SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," March 29, 2005

14 pages follow

10 CFR Part 21 Communication

GE Energy- Nuclear

SC05-03 **March 29, 2005**

To: **Identified Plants (Attachment 1)**

Subject: **Potential to Exceed Low Pressure Technical Specification Safety Limit**

X	**Reportable Condition [21.21(d)]**		**60 Day Interim Report [21.21(a)(2)]**
	Transfer of Information [21.21(b)]		**Safety Information Communication**

Summary:

GE has continued to improve the methodology used for licensing basis transient analyses. The approved model has evolved from REDY, to ODYN, to TRACG. Reactor depressurization transients, such as Pressure Regulator Failure-Maximum Demand (Open) (PRFO), are non-limiting for fuel cladding integrity because Critical Power Ratio (CPR) increases during the event, and they are not typically included in the scope of reload evaluations. Recent investigations by GE have determined that even though REDY, ODYN, and TRACG all show CPR increasing during the transient, the difference in reactor level swell predicted by REDY, vs. ODYN and TRACG, can impact the predicted plant response to the PRFO.

Technical Specification (TS) Safety Limits (SL) are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor Core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded. The standard Improved Technical Specifications (ITS) specify SL 2.1.1.1 to require that thermal power shall be ≤ [25]% rated (the value is a plant-specific number), when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. Many plants have implemented the ITS or have TS that contain a similar SL. This SL was introduced to preclude

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the need for CPR calculations when reactor steam dome pressure is less than 785 psig. The power value in ITS SL 2.1.1.1 is selected to ensure that power remains well below the fuel assembly critical power for the conditions in which CPR calculations are not performed.

Previous evaluations by GE using the REDY model predicted that reactor water level would swell during a PRFO transient; the depressurization would be terminated by a high level turbine trip. However, level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. Recent evaluations by GE with the improved transient models have determined that the reactor level swell may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by MSIV closure at the low-pressure isolation setpoint (LPIS). Depending upon the plant-specific response to a PRFO, 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% of rated, which would exceed the conditions in ITS SL 2.1.1.1. This indicates that ITS SL 2.1.1.1 is overly conservative, because an event that causes CPR to increase and does not threaten fuel cladding integrity can result in exceeding this reactor core SL.

Absent a plant-specific analysis that demonstrates otherwise, GE is identifying plants that have the potential for reactor dome pressure to drop below 785 psig for a PRFO as those with a LPIS-Analytical Limit (AL) of less than 785 psig. These plants are listed as Affected in Attachment 1. Sufficient evaluations to determine the plant specific conditions (for example, off-rated conditions) for which the PRFO will not cause reactor dome pressure to drop below 785 psig have not been completed and are judged to not be necessary since this is a non-limiting transient for challenges to fuel cladding integrity. Therefore, all other plants identified in Attachment 1 are considered to be potentially vulnerable to this condition.

The 10 CFR 21.3 definition of a *Defect* includes, "a condition or circumstance involving a basic component that could contribute to the exceeding of a safety limit, as defined in the technical specifications." Section 21.21(d) requires that the NRC be notified upon determination of the existence of a defect. Even though this condition does not threaten fuel cladding integrity or produce a significant safety hazard, it is defined as a *Defect* under 10 CFR 21 because the conditions in plant TS corresponding to ITS SL 2.1.1.1 may be exceeded, which requires NRC notification as a Reportable Condition.

10 CFR 50.36(c)(1) provides requirements for Limiting Safety System Settings (LSSS). Subparagraph (ii)(A) states, "*LSSS for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded*." Plants may have identified the LPIS as a LSSS. However, in this instance, the LPIS does not have a "significant safety function" since its operation is not required to protect fuel cladding integrity, and so the LPIS should not be specified as a LSSS.

TS Bases for the main steam line (MSL) low pressure isolation state that MSIV closure ensures that the RPV temperature change limit of 100°F/hr is not reached, and that it supports actions to ensure SL 2.1.1.1 is not exceeded. The TS Bases states clearly that the LPIS-Allowable Value is based on preventing excessive RPV depressurization. While low-pressure MSIV closure supports actions to ensure SL 2.1.1.1 is not exceeded, this is not a “significant safety function” of the LPIS.

Background

Most operating BWRs have a TS SL that requires the core thermal power to be < [25]% of rated when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. This was intended to provide fuel cladding integrity protection during start-up conditions since the GEXL correlation (used by GE to perform CPR calculations) is not approved as a licensing model for pressures < 785 psig (~800 psia). GE has tested some fuel designs over an extended pressure range, in some cases as low as 600 psia. The critical power continues to increase monotonically as pressure decreases below the bottom of the approved range, and results in CPR increasing as pressure decreases. However, the GEXL correlation is not approved by the NRC for licensing calculations for pressures below 800 psia.

Licensees are required to demonstrate that no AOO will cause a SL to be exceeded, as documented in plant Safety Analysis Reports. AOOs that cause reactor depressurization are not routinely evaluated because thermal margins increase during these events, and therefore, they are non-limiting. The PRFO is the only AOO capable of depressurizing the reactor vessel enough to reach the MSIV low-pressure isolation setpoint (LPIS). PRFO evaluations with REDY showed that vessel level swell caused by the depressurization resulted in a high water level turbine trip, which resulted in the event being terminated before the pressure in the main steam line (MSL) reached the LPIS. Therefore, GE did not perform an evaluation of the PRFO or specify a LPIS-AL to demonstrate compliance with the conditions of ITS SL 2.1.1.1.

Evaluation

An evaluation of the PRFO with ODYN shows that the vessel level swell may not be sufficient to cause a high water level trip and the depressurization transient may not be terminated until the MSIV closes due to the low MSL pressure. The scenario for this event, is (1) a pressure regulator failure results in the maximum steam demand (specified by the value of the Maximum Combined Flow Limiter (MCFL), typically between 110 and 130%), (2) the turbine control valves (TCV) and the turbine bypass valves (TBV) open as required to meet the steam demand, (3) the steam flow increases and the pressure in the reactor steam dome and MSL decreases and (4) once the turbine inlet pressure (Pturb) reaches the LPIS, a MSIV trip occurs. The MSIV closure terminates the depressurization and scrams the reactor.

GE analysis assumes that the MSIV is tripped at the turbine inlet pressure corresponding to the LPIS-Analytical Limit (AL). The trip may occur significantly sooner since the Allowable Value and Nominal Trip setpoint implemented by the licensees may significantly exceed the LPIS-AL.

A typical PRFO response for a LPIS-AL of 720 psig is shown in Figure 1. Reactor dome pressure decreases until terminated by MSIV closure. This results in dome pressure (Pdome) dropping below the SL 2.1.1.1 value (785 psig = ~800 psia on Figure 1), while heat flux, which is indicative of thermal power, is still in excess of 25% of rated power. The CPR (not shown) continues to increase during the depressurization, so that the initial CPR is the limiting CPR condition during the entire transient. The MSIV closure signal is generated when the turbine inlet pressure (Pturb) reaches the LPIS-AL (shown as 735 psia in Figure 1). The conditions that exceed SL 2.1.1.1 exist for only a few seconds, and as stated previously, CPR increases during the event relative to the initial CPR value, so fuel cladding integrity is not threatened. Nonetheless, this is now a known AOO that could contribute to the exceeding of a safety limit, as defined in the technical specifications.

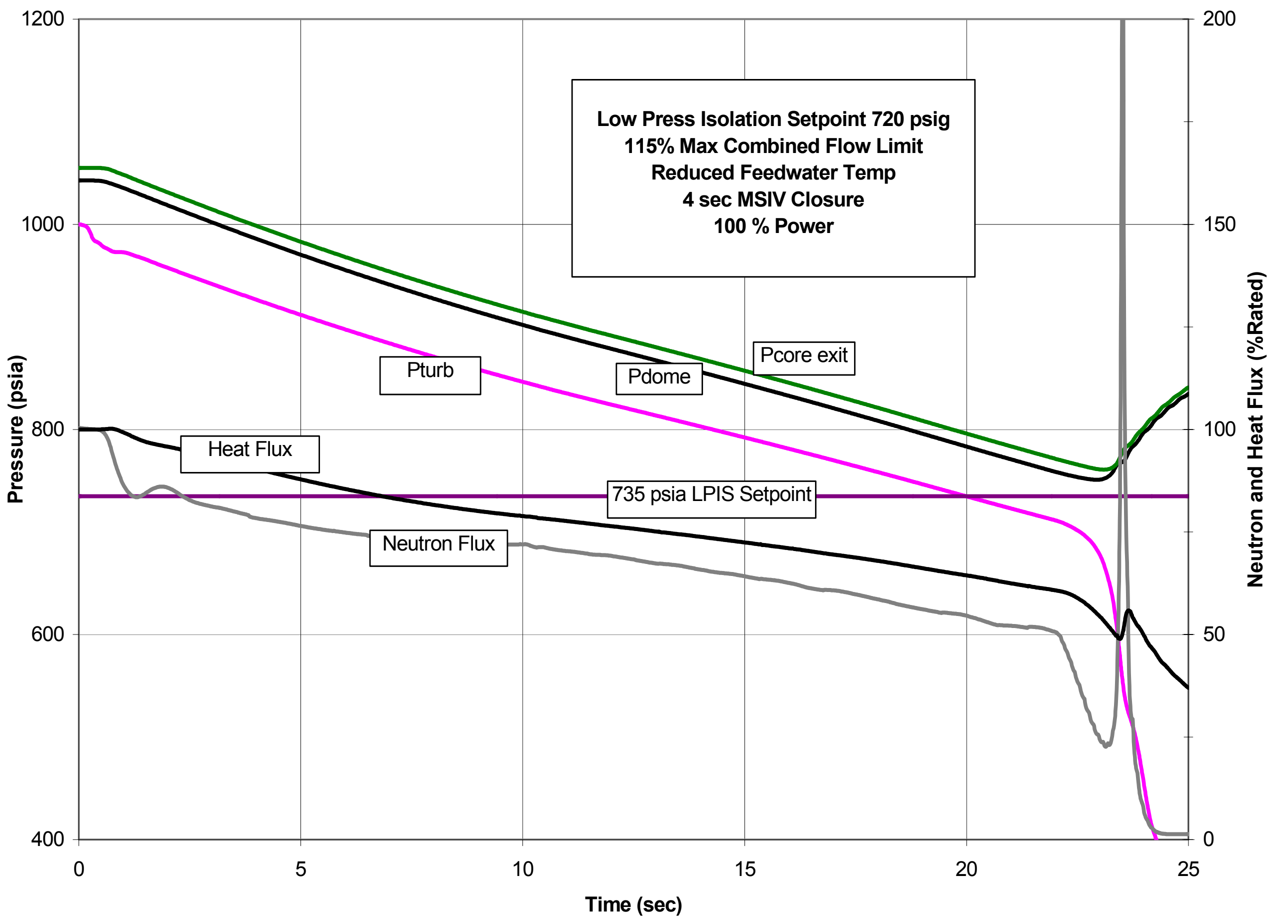

Figure 1. Typical PRFO Response

Important parameters which impact plant response to a PRFO event include (1) LPIS, (2) MCFL, (3) feedwater temperature, (4) steam line pressure drop, (5) MSIV closure time, (6) turbine inlet pressure sensor delay, and (7) initial core power. Table 1 lists values contained in GE records for the LPIS-AL (which is a turbine inlet pressure value) and MCFL. Actual plant-specific values may deviate from the values contained in GE records. The MCFL, feedwater temperature and initial core power impact the depressurization rate. The MCFL impacts depressurization rate because it determines the TCV and TBV opening. The reduced feedwater temperature and initial power less than rated are conditions that have less than rated steam flow. A PRFO from these conditions results in a larger steam flow mismatch than if initiated from rated conditions. This results because the TCV and TBV are assumed to open to the maximum demand regardless of initial conditions. The LPIS-AL determines the turbine inlet pressure at which the isolation is initiated. The steam line pressure drop maintains the reactor steam dome pressure above the turbine inlet pressure. The steam line pressure drop ranges from approximately 30 to 100 psi. The MSIV closure time and the turbine inlet pressure sensor delay determine how long the depressurization continues after the LPIS-AL has been reached.

The ranges (approximate) in these parameters is large; 720 to 850 psig for LPIS-AL, 108 to 130% for MCFL, 0 to 170°F for feedwater temperature reduction (most plants fall in the range of 70 to 105°F), 30 to 100 psi for steam line pressure drop, MSIV closure time is typically 3 sec (minimum) and 5 sec (maximum), and variation on the turbine inlet pressure sensor delay is unknown. The potential for operation at off-rated core powers must also be considered.

Several evaluations were performed within the range of parameters identified above, including off-rated initial operating conditions, to determine the possibility that the dome pressure would drop below the SL 2.1.1.1 value of 785 psig while thermal power exceeded [25]% of rated. From this limited set of evaluations, it was apparent the plants with LPIS-AL below 785 psig have the potential to experience a PRFO that results in the dome pressure dropping below 785 psig while thermal power exceeds [25]% of rated. Evaluations of the PRFO over a wide range of conditions led to the conclusion that all plants are considered potentially vulnerable to exceeding SL 2.1.1.1. While GE has not evaluated every possible plant combination to determine the full extent of which plants are/are not affected by this condition, GE believes that the evaluation is sufficient in this instance since the recommendation is that no compensatory actions are appropriate, and that SL 2.1.1.1 should be corrected so that it provides the intended fuel cladding integrity protection for startup conditions and does not provide unnecessary and overly conservative restrictions for events that do not threaten fuel cladding integrity.

Safety Basis

Technical Specification SL 2.1.1.1 requires that reactor power shall be ≤ [25]% of rated when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated. SLs must be met for both normal operation and for anticipated operational occurrences (AOO). A Pressure Regulator Failure – Maximum Demand (Open) (PRFO) is defined as an AOO and an analysis using GE transient analysis methods and licensing basis assumptions has demonstrated that at least for some plants, this event can cause SL 2.1.1.1 to be exceeded, and is therefore, a reportable condition under 10 CFR 21. GE is not aware of any AOO that has caused SL 2.1.1.1 to be exceeded, and plant-specific MSIV low-pressure isolation setpoints that are higher than the LPIS-AL make it less likely that this would occur. The LPIS should not be considered a LSSS for protection of SL 2.1.1.1 since it does not provide a "significant safety function" with regard to protecting fuel cladding integrity. Critical power ratio continues to increase with decreasing reactor pressure (to at least pressures as low as 600 psia based on historical GE critical power testing), so there is not threat to fuel cladding integrity for this event. Even though there is no safety significance to this condition, it is a condition that may cause SL 2.1.1.1 to be exceeded.

Justification for Continued Operation

The purpose of reactor core SLs is to protect fuel cladding integrity such that significant fuel damage is not calculated to occur for any steady-state operation condition, normal operation transient, or AOO. The PRFO does not threaten fuel cladding integrity. Specifically, the Minimum Critical Power Ratio Safety Limit (SLMCPR) is defined to ensure fuel cladding integrity is protected. This event causes the Critical Power Ratio (CPR) to increase. With an initial condition that is restricted by the MCPR Operating Limit (OLMCPR) and an event that causes the CPR to increase, the margin to the SLMCPR increases during the event and therefore no threat to fuel cladding integrity exists.

Even if a PRFO were to occur in the interim before the overly conservative condition in SL 2.1.1.1 is addressed by a TS modification, SL 2.1.1.1 may not be exceeded due to the actual plant conditions when the transient occurred, including an actual LPIS value that may be significantly above the LPIS-AL used in the GE analysis.

An immediate plant change such as increasing the LPIS-AL is unfounded and could potentially degrade plant safety. Licensees have set the LPIS at the current plant values for a variety of reasons. Raising the LPIS-AL may cause unnecessary safety system challenges by increasing the incidence of plant isolations when, as described above, the PRFO event does not challenge fuel cladding integrity. Increasing the possibility of safety system challenges when not necessary could adversely affect plant safety and, therefore, is not appropriate.

Corrective/Preventive Actions

The GE evaluation has shown a vulnerability to exceeding a reactor core SL, for a condition that does not threaten fuel cladding integrity in any way. In this case, there is no clear compensatory action that can be defined to appropriately mitigate this vulnerability. Since the condition does not challenge the physical barrier that the SL is intended to protect, the LPIS should not be considered as a LSSS for SL 2.1.1.1 and there is no need for a compensatory action.

The BWROG Technical Specification Improvement Coordination Committee (TSICC) is evaluating this condition and the specification of SL 2.1.1.1. GE and the BWROG will develop a proposed resolution to this issue and interact with the NRC to obtain approval of the proposed TS modification. GE and the BWROG are willing to meet with the NRC at their convenience to discuss the situation and the potential approaches for a TS modification. Plant Technical Specifications and Bases will be developed as necessary, depending upon the outcome of the TS modification activity.

Table 1. LPIS-AL* and MCFL Values Contained in GE Records

Utility	Plant	Cycle	LPIS-AL (PSIG)	MCFL (%)
AmerGen Energy Co.	Clinton	10	825	115
AmerGen Energy Co.	Oyster Creek	20	825	110
Carolina Power & Light Co.	Brunswick 1	15	785	115
Carolina Power & Light Co.	Brunswick 2	16	785	115
Constellation Nuclear	Nine Mile Point 1	19	834.2	110
Constellation Nuclear	Nine Mile Point 2	10	720	115
Detroit Edison Co.	Fermi 2	11	720	115
Energy Northwest	Columbia			
Entergy Nuclear Northeast	FitzPatrick	17	825	115
Entergy Nuclear Northeast	Pilgrim	16	782.3	109
Entergy Nuclear Northeast	Vermont Yankee	24	765	
Entergy Operations, Inc.	Grand Gulf	11	837	
Entergy Operations, Inc.	River Bend	10	825	
Exelon Generation Co.	Dresden 2	19	785	110
Exelon Generation Co.	Dresden 3	19	785	110
Exelon Generation Co.	LaSalle 1	11	825	120
Exelon Generation Co.	LaSalle 2	11	825	120
Exelon Generation Co.	Limerick 1	11	720	115
Exelon Generation Co.	Limerick 2	9	720	115
Exelon Generation Co.	Peach Bottom 2	16	750	115
Exelon Generation Co.	Peach Bottom 3	15	750	115
Exelon Generation Co.	Quad Cities 1	19	785	110
Exelon Generation Co.	Quad Cities 2	18	785	110
FirstEnergy Nuclear Operating Co.	Perry 1	11	782.3	123
Nebraska Public Power District	Cooper	23	825	110
Nuclear Management Co.	Duane Arnold	20	800	125
Nuclear Management Co.	Monticello	23	809	108
PPL Susquehanna LLC.	Susquehanna 1			
PPL Susquehanna LLC	Susquehanna 2			
PSEG Nuclear	Hope Creek	13	720	130
Southern Nuclear Operating Co.	Hatch 1	22	795	115
Southern Nuclear Operating Co.	Hatch 2	19	795	115
Tennessee Valley Authority	Browns Ferry 1	6	825	
Tennessee Valley Authority	Browns Ferry 2	13	825	
Tennessee Valley Authority	Browns Ferry 3	11	825	
BKW-FMB Energy, Ltd.	Muehleberg	32	783.2	
Comision Federal de Electricidad	Laguna Verde 1	11	825	115
Comision Federal de Electricidad	Laguna Verde 2	8	825	115
Iberdrola SA	Cofrentes	11	825	
Japan Atomic Power Co.	Tokai 2	7	850	
Japan Atomic Power Co.	Tsuruga	14	825	
Kernkraftwerk Leibstadt AG	Leibstadt	7	837	
Nuclenor SA	Santa Maria de Garoña	24	800	110
Taiwan Power Company	Chinshan 1	19	850	
Taiwan Power Company	Chinshan 2	18	850	
Taiwan Power Company	Kuosheng 1			

Utility	Plant	Cycle	LPIS-AL (PSIG)	MCFL (%)
Taiwan Power Company	Kuosheng 2	3	833	
Taiwan Power Company	Lungmen 1	1	750	
Taiwan Power Company	Lungmen 2	1	750	
Tokyo Electric Power Company	Fukushima Daini 2			
Tokyo Electric Power Company	Fukushima Daiichi 6	3	825	
Tokyo Electric Power Company	Kashiwazaki 6			
Tokyo Electric Power Company	Kashiwazaki 7			

*LPIS-AL values are turbine inlet pressure

Attachment 1 - Notified Plants

Affected	Potentially Affected	Utility	Plant
	X	AmerGen Energy Co.	Clinton
	X	AmerGen Energy Co.	Oyster Creek
	X	Carolina Power & Light Co.	Brunswick 1
	X	Carolina Power & Light Co.	Brunswick 2
	X	Constellation Nuclear	Nine Mile Point 1
X		Constellation Nuclear	Nine Mile Point 2
X		Detroit Edison Co.	Fermi 2
		Dominion Generation	Millstone 1
		Energy Northwest	Columbia
	X	Entergy Nuclear Northeast	FitzPatrick
X		Entergy Nuclear Northeast	Pilgrim
X		Entergy Nuclear Northeast	Vermont Yankee
	X	Entergy Operations, Inc.	Grand Gulf
	X	Entergy Operations, Inc.	River Bend
		Exelon Generation Co.	CRIT Facility
	X	Exelon Generation Co.	Dresden 2
	X	Exelon Generation Co.	Dresden 3
	X	Exelon Generation Co.	LaSalle 1
	X	Exelon Generation Co.	LaSalle 2
X		Exelon Generation Co.	Limerick 1
X		Exelon Generation Co.	Limerick 2
X		Exelon Generation Co.	Peach Bottom 2
X		Exelon Generation Co.	Peach Bottom 3
	X	Exelon Generation Co.	Quad Cities 1
	X	Exelon Generation Co.	Quad Cities 2
X		FirstEnergy Nuclear Operating Co.	Perry 1
	X	Nebraska Public Power District	Cooper
	X	Nuclear Management Co.	Duane Arnold
	X	Nuclear Management Co.	Monticello
		Pooled Equipment Inventory Co.	PIM
		PPL Susquehanna LLC	Susquehanna 1
		PPL Susquehanna LLC	Susquehanna 2
X		PSEG Nuclear	Hope Creek
	X	Southern Nuclear Operating Co.	Hatch 1
	X	Southern Nuclear Operating Co.	Hatch 2
	X	Tennessee Valley Authority	Browns Ferry 1*
	X	Tennessee Valley Authority	Browns Ferry 2
	X	Tennessee Valley Authority	Browns Ferry 3

*The plant is in an extended shutdown

Non-US BWR Plants

Affected	Potentially Affected	Utility	Plant
	X	BKW-FMB Energy, Ltd.	Muehleberg
	X	Comision Federal de Electricidad	Laguna Verde 1-2
	X	Iberdrola SA	Cofrentes
	X	Japan Atomic Power Co.	Tokai 2
	X	Japan Atomic Power Co.	Tsuruga 1
	X	Kernkraftwerk Leibstadt AG	Leibstadt
	X	Nuclenor SA	Santa Maria de Garoña
	X	Taiwan Power Company	Chinshan 1-2
	X	Taiwan Power Company	Kuosheng 1-2
	X	Taiwan Power Company	Lungmen 1-2*
	X	Tokyo Electric Power Company	Fukushima Daiichi 6
		Tokyo Electric Power Company	Fukushima Daini 2
		Tokyo Electric Power Company	Kashiwazaki 6-7

* Under construction

<u>Attachment 2</u> - Reportable Condition per §21.21(d)

(i) Name and address of the individual informing the Commission:

J. S. Post, Manager, Engineering Quality & Safety Evaluations, GE Energy - Nuclear, 3901 Castle Hayne Road, Wilmington, NC 28401.

(ii) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect:

The affected and potentially affected plants are identified in Attachment 1.

(iii) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect:

GE Energy – Nuclear, Wilmington, NC.

(iv) Nature of the defect or failure to comply and safety hazard which is created or could be created by such defect or failure to comply:

The defect is a calculation of an anticipated operational occurrence (AOO), which predicts that the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient will be terminated by a high water level trip as a result of level swell in the reactor. An improved (and approved) model predicts that MSIV closure will occur when steam line pressure reaches the low-pressure isolation setpoint (LPIS), rather than terminate due to a high water level trip. Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig while thermal power exceeds 25% of rated, which would be a violation of SL 2.1.1.1. This constitutes a *Defect* as defined in 10 CFR 21.3, even though there is no safety hazard created. SL 2.1.1.1 was intended to protect fuel cladding integrity during startup conditions without the need to perform a Critical Power Ratio (CPR) calculation. The AOO of concern is a transient from normal operating conditions that causes CPR to increase, so the event produces additional margin to the Minimum Critical Power Ratio Safety Limit (SLMCPR) and does not threaten fuel cladding integrity. The LPIS should not be considered as a Limiting Safety System Setting (LSSS) for SL 2.1.1.1 since it does not provide a “significant safety function” with regard to protecting fuel cladding integrity. This indicates that SL 2.1.1.1 is overly conservative because an event that causes CPR to increase and does not threaten fuel cladding integrity, may result in exceeding a reactor core SL.

(v) The date on which the information of such defect or failure to comply was obtained:

This was identified as a potentially reportable condition by GE on January 28, 2005.

(vi) In the case of a basic component which contains a defect or failure to comply, the number and locations of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part:

The basic component which contains the defect is a licensing analysis for a Pressure Regulator failure – Maximum Demand (Open) (PRFO) that shows reactor water level swell caused by the event will lead to turbine trip and the event will be terminated by the pressurization associated with the turbine trip. Updated GE models indicate that reactor pressure will decrease for this event until the MSIV low-pressure isolation setpoint (LPIS) is reached. GE analysis of the PRFO assuming MSIV trip is initiated at the LPIS-Analytical Limit, indicates that this can lead to exceeding Technical Specification SL 2.1.1.1.

(vii) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action (note, these are actions specifically associated with the identified Reportable Condition):

The GE evaluation has shown a vulnerability to exceeding a reactor core SL, for a condition that does not threaten fuel cladding integrity in any way. In this case, there is no clear compensatory action that can be defined to appropriately mitigate this vulnerability, and since the condition does not challenge the physical barrier that the SL intends to protect (i. e., the fuel cladding integrity), there is no safety basis for a compensatory action.

GE and the BWROG Technical Specification Improvement Coordination Committee (TSICC) is evaluating this condition and the specification of SL 2.1.1.1. GE and the BWROG are willing to meet with the NRC at their convenience to discuss the situation and the potential approaches for a TS modification. The BWROG will develop a proposed resolution to this issue and interact with the NRC to obtain approval of the proposed TS modification.

(viii) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees:

The purpose of reactor core SLs is to protect fuel cladding integrity such that significant fuel damage is not calculated to occur for any steady-state operation condition, normal operation transient, or AOO. The PRFO does not threaten fuel cladding integrity. Specifically, the Minimum Critical Power Ratio Safety Limit (SLMCPR) is defined to ensure fuel cladding integrity is protected. This event causes the Critical Power Ratio

(CPR) to increase. With an initial condition that is restricted by the MCPR Operating Limit (OLMCPR) and an event that causes the CPR to increase, there is increased margin to the SLMCPR during the event and there is no threat to fuel cladding integrity.

Even if a PRFO were to occur in the interim before the overly conservative condition in SL 2.1.1.1 is addressed by a TS modification, SL 2.1.1.1 may not be exceeded due to the actual plant conditions when the transient occurred, including an actual LPIS value that may be significantly above the LPIS-AL used in the GE analysis.

An immediate plant change such as increasing the LPIS-AL is unfounded and could potentially degrade plant safety. Licensees have set the LPIS at the current plant values for a variety of reasons. Raising the LPIS-AL may result in unnecessary safety system challenges by increasing the incidence of plant isolations when, as described above, this event does not challenge fuel cladding integrity. Increasing the possibility of safety system challenges when not necessary could adversely affect plant safety, and therefore, is not appropriate.

Attachment 3 – Recent GENE 10 CFR Part 21 Communications

The following is a list of recent 10 CFR Part 21 communications that GE Energy-Nuclear has provided to affected licensees as Reportable Conditions (RC), Transfers of Information (TI), 60 Day Interim Reports (60 Day) or Safety Information Communications (SC).

Number	Ref.	Subject	Date
SC04-01	PRC 03-64	Missing Rebound Hook Buffer AK-25 Mechanism Assemblies (TI)	1/12/04
SC04-02	PRC 03-69	ABWR Main Steam Line Break Containment Analysis Methods (TI)	2/2/04
SC04-03	PRC 03-68	Impact of Hydrogen-Oxygen Recombination on BWR/2 LOCA Analysis (SC)	4/27/04
SC04-04	PRC 04-05	SRV Cycling Impact on Containment Load Evaluations (TI)	4/30/04
SC04-05	PRC 04-06	Loose Internal Terminal Strip, Barksdale Pressure Switch (TI, SC)	5/7/04
SC04-06	PRC 04-10	Spent Fuel Storage Sub-criticality Confirmation (TI, SC)	6/24/04
SC04-07	PRC 04-10	Non-US Plant Communication; Spent Fuel Storage Sub-criticality (TI, SC)	6/25/04
SC04-08	PRC 04-12	AKR-30/-50 Circuit Breaker Latch Roller (TI)	7/7/04
SC04-09	PRC 04-13	SBM Switch Coined Contact Tip Assembly (SC)	7/12/04
SC04-10	PRC 04-16	SRM/IRM Preamps (SC)	8/12/04
SC04-11	PRC 04-15	Narrow Range Water Level Instrument Level 3 Trip (SC, 60-Day)	8/16/04
SC04-12	PRC 04-18	Non-conservative SLMCPR (RC, 60-Day)	8/24/04
SC04-13	PRC 04-18	Non-Conservative SLMCPR Final Report (SC)	9/29/04
SC04-14	PRC 04-15	Narrow Range Water Level Instrument Level 3 Trip Final Report (TI, SC)	10/11/04
SC04-15	PRC 04-23	Turbine Protection System Impact in Transient Analysis (TI, SC)	10/31/04
SC04-16	PRC 04-27	Shutdown Margin Calculation Process (SC)	12/9/04
SC04-17	PRC 04-29	HMA Relay Coil Failures (SC)	12/14/04
SC05-01	PRC 04-24	Excessive Current Leakage in Zener Diodes Used for RMCS and RC&IS Serial Word Transmissions (SC)	2/17/05
SC05-02	PRC 05-02	Potential Spurious Recirculation Pump Downshifts from High to Low Speed; High Level Optical Isolator Cards (SC)	3/29/05
SC05-03	PRC 05-01	Potential to Exceed Low Pressure Technical Specification Safety Limit (RC)	3/29/05