NRR-PMDAPEm Resource

From: Beltz, Terry
Sent: Wednesday, December 18, 2013 7:52 AM
To: Adams, Glenn D.
Cc: Eckholt, Gene F. (Eugene.Eckholt@xenuclear.com); 'Fields, John S.'; Carlson, Robert; Jackson, Christopher; Parks, Benjamin; Dean, Jeremy; Panicker, Mathew
Subject: Monticello Nuclear Generating Plant - Request for Additional Information re: NRC Staff Review of AREVA Fuel Transition License Amendment Request (TAC No. MF2479)
Attachments: Monticello Nuclear Generating Plant - Request for Additional Information (SRXB) (TAC No. MF2479).docx

Dear Mr. Adams:

By letter dated July 15, 2013, 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. 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 of the Office of Nuclear Reactor Regulation has identified areas where additional information is needed to complete its review. The requests for additional information (RAIs) are attached.

Please provide a response to the RAIs by January 31, 2014. You may request to discuss the contents of these RAIs with the NRC staff in a conference call, including any change to the proposed response date.

Please let me know if you have any questions or concerns.

Sincerely,

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OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR SYSTEMS BRANCH

REQUEST FOR ADDITIONAL INFORMATION

REGARDING LICENSE AMENDMENT REQUEST

TO TRANSITION TO AREVA ATRIUM 10XM FUEL AND SAFETY ANALYIS PRODUCTS

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

(TAC NO. MF2479)

By letter dated July 15, 2013 (Agencywide Documents Access and Management System Accession No. ML13200A185), Northern States Power Company - a Minnesota corporation, doing business as Xcel Energy (the licensee), requested an amendment to the operating license and facility Technical Specifications (TSs) 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, as outlined in the following requests for additional information.

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 Clad 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.

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.[1]

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.

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."

[1] 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.

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.

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.

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.

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.

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

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.

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

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.

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.