ENCLOSURE 2

AREVA REPORT NO. ANP-3289NP, REVISION 0

RESPONSES TO RAI FROM SNPB ON MNGP TRANSITION TO AREVA FUEL

NON-PROPRIETARY

36 pages follow

ANP-3289NP
Revision 0

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February 2014

AREVA Inc.

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Nature of Changes

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

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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 Nuclear Performance and Code Review branch (SNPB) is reviewing the safety analyses for anticipated operational occurrences (AOOs), design basis accidents (DBAs), and special events. The SNPB 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

SNPB RAI-1: ANP-3221P, Section 3.2.2

Please provide a detailed description of the statistical method used in the analysis for cladding creep collapse that yielded the best-estimate results for the creep collapse of the cladding.

AREVA Response

The creep collapse criterion is evaluated by calculating the formation of an axial gap in the fuel column due to fuel densification for comparison to the design limit on gap size. The design criterion is as follows.

Clad creep collapse shall be prevented. [

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The RODEX4 application methodology is a statistical uncertainty propagation method that uses a Monte Carlo random sampling of relevant input parameters to evaluate the propagation of the uncertainties to the design analysis results. [

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The uncertainties used in the analysis are categorized as [

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Figure 1 – Calculated Cladding Collapse Margin to Fuel Column Axial Gap Limit

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The above description of the methodology and criteria are consistent with the approved RODEX4 topical report, BAW-10247P (Reference 3). More specifically, details on the methodology can be found in an RAI response during the review of RODEX4, Response 18 on page 84 of BAW-10247Q4P that is contained within the approved topical report.

SNPB RAI-2: ANP-3221P, Section 3.2.3

In the methodology for analysis of overheating of fuel pellets, it is stated that linear heat generation rate (LHGR) margins are provided along with LHGR uncertainties due to channel bow input to the statistical analysis.

Please provide details of how the channel bow uncertainties are developed and how they are incorporated in to the statistical analysis.

Note: This RAI is similar to the one staff made for the review of ANP-3159P which was part of the Browns Ferry units' fuel transition to ATRIUM 10XM fuel design.

AREVA Response

The uncertainty in the calculated channel bow leads to an associated uncertainty in the fuel rod power level. This uncertainty in power is taken into account as part of the RODEX4 statistical application methodology. A series of steps are carried out to assess the effect of channel bow and its associated model uncertainty on the fuel rod thermal-mechanical behavior by accounting for channel bow in the generation of the fuel rod power histories.

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The above description is consistent with the methodology described in Reference 3. Additional information can be found in the third round of RAI responses, BAW-10247Q3(P), that is contained in the approved RODEX4 topical report.

The RODEX4 results presented in ANP-3221P for the Monticello Cycle 28 ATRIUM 10XM fuel include the adjustments as described above to account for power uncertainties from channel bow. The method is identical to that used for the RODEX4 calculations in support of the Browns Ferry LAR and the Brunswick reload licensing calculations.

<u>SNPB RAI-3: ANP-3221P, Section 3.2.7</u>

Please provide details of how RODEX4 treats heat transfer coefficient to account for the presence of crud at normal, low level and abnormal level, and explain how this is applied to crud measurements at MNGP.

<u>AREVA Response</u>

The RODEX4 code includes a provision to input a crud thickness layer and the associated thermal conductivity of the crud. This feature was briefly described in one of the RAI responses in the RODEX4 topical report (BAW-10247Q4P, Response 15, p77, Reference 3). If input, the crud layer is modeled as an additional thermal resistance term between the coolant film and the corrosion layer. The input to RODEX4 for a crud layer depends on whether the plant is considered to have low, normal levels of crud or if higher levels of crud are indicated.

The RODEX4 model for calculating corrosion is described as an oxidation model and it calculates a corrosion layer using a relation based on the kinetics of oxidation. However, it would be more accurate to characterize the model as predicting "liftoff" because the model is benchmarked to liftoff measurement data and not solely corrosion data. The term liftoff refers to the separation or liftoff of the eddy current measurement probe from the metallic surface of the fuel rod due to the presence of the insulating corrosion and crud layers. The liftoff measurement method cannot discern between oxide and crud. Visual examination of the fuel rod surfaces, more recent hot cell data, and crud scrape measurements confirm the presence of the thin, tenacious crud layers.

BAW-10247PA SER restriction 5 on crud says:

> "RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal crud/corrosion layer is defined by a formation that increases the calculated fuel average temperature by more than 25 °C beyond the design basis calculation. ..."

Additional information was included in response to the RODEX4 SER restriction on addressing crud. The information states "As our clad measurements [

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The RODEX4 SER restriction specifies that specific analyses would be required if an abnormal crud or corrosion layer is observed at any given plant. The SER restriction defines an abnormal crud or corrosion layer by a formation that increases the calculated fuel average temperature by more than 25 °C beyond the design basis calculation. As part of the approved RODEX4 methodology, analyses are already performed for each cycle. Where plant specific measurements indicating abnormal crud are obtained, analyses for that plant are based on the plant specific data. If liftoff levels are found to be greater than those used in the RODEX4 corrosion model benchmark, then a plant-specific crud thickness will be input to encompass the total liftoff thickness. The crud input will serve to satisfy the SER restriction on the design basis crud layer in cases where abnormal crud is encountered.

Since the corrosion model was benchmarked to measurements that include some amounts of existing crud, there is an inherent assumption [

] The small temperature difference is otherwise not significant to fuel rod performance or safety.

If abnormal levels of crud are encountered, the selection of the crud thermal conductivity input becomes more important. The thermal conductivity of a crud layer depends on the composition, porosity, structure, and operating conditions and it continues to be a subject of development. Values of crud thermal

conductivity obtained from literature range from approximately 0.7 W/(m*K) up to nearly a factor of ten greater. Thermal conductivity values are more typically greater than 0.8 and less than that of oxide (i.e., less than 2.0 W/(m*K)). Unless specific crud characterization and/or measurements are available for a plant, a value of [ ] will be used. Note that the combined layer of oxide and crud includes the selection of a conservative oxide thermal conductivity that contributes to the composite thermal resistance.

In the transition to the ATRIUM 10XM design at Monticello, [

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Crud is considered primarily a part of the plant operating environment that is mainly a function of the plant water chemistry conditions.

To assess the levels of crud at Monticello, Xcel provided AREVA with a prior water chemistry evaluation which contained the conclusion that current water chemistry conditions do not reduce fuel reliability margins according to industry water chemistry guidelines.

Visual examination data of fuel from past operating cycles in Monticello were provided to AREVA for review. One examination was on two fuel assemblies that had operated for four cycles. Photos revealed crud levels that visually appear consistent (i.e., no worse) in comparison to visuals at other plants with AREVA fuel that had previously been judged to have normal (i.e., low) crud levels as based on liftoff measurement data.

During the re-channeling of eight fuel assemblies over the past year, Xcel characterized the appearance of the fuel as having normal, uniform crud loading with no indications of abnormal crud layers.

Based on the above information available from the Monticello plant, the crud conditions were taken to be within normal levels experienced by plants used for the RODEX4 corrosion model benchmarking. Therefore, no additional crud input was necessary to account for a design basis crud level.

SNPB RAI-4: ANP-3092P, Section 2.0

It has been stated in Section 2.0 that the ATRIUM 10XM fuel assemblies are hydraulically compatible with the co-resident GE14 fuel design for the entire range of licensed power-to-flow operating map. Please clarify this statement and elaborate whether the hydraulic compatibility between ATRIUM 10XM and GE14 fuel designs for all combinations of power-to-flow operating map for MELLLA+ and extended flow window (EFW) during transition cycles as well as for full-core ATRIUM 10XM at MNGP.

AREVA Response

Tables 3.7 and 3.8 of ANP-3092P provide thermal-hydraulic results for transition cycles at 100% core power / 100% core flow (rated core flow at rated EPU power) and 59.2% core power / 43.3% core flow (minimum pump speed on MELLLA line), respectively. Table 1, Table 2, and Table 3 provide similar thermal-hydraulic results at 100% core power / 105% core flow (maximum core flow at rated EPU power), 100% core power / 80% core flow (minimum core flow at rated EPU power on MELLLA+ line) and 82.5%

core power / 57.4% core flow (minimum core flow at the MELLLA+ boundary). The results presented in Table 1, Table 2, and Table 3 demonstrate the thermal-hydraulic design criteria are satisfied for full-core and transition core configurations. Differences in core average results (core pressure drop and core bypass flow) between ATRIUM 10XM and GE14 results are within the range considered compatible. [

] The Critical Power Ratio (CPR) results of the ATRIUM 10XM and GE14 indicate ATRIUM 10XM fuel will not cause thermal margin problems for the coresident fuel design. The results presented in ANP-3092P and the additional results provided in Table 1, Table 2, and Table 3 demonstrate the hydraulic compatibility between ATRIUM 10XM and GE14 fuel designs for the licensed power/flow map for MNGP during transition cycles as well as for full-cores of ATRIUM 10XM fuel.

Table 1 Monticello Thermal-Hydraulic Results at Rated Conditions (100%P / 105%F) for Transition to ATRIUM 10XM Fuel

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Table 2 Monticello Thermal-Hydraulic Results at Rated Conditions (100%P / 80%F) for Transition to ATRIUM 10XM Fuel

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Table 3 Monticello Thermal-Hydraulic Results at Off-Rated Conditions (82.5%P / 57.4%F) for Transition to ATRIUM 10XM Fuel

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SNPB RAI-5: ANP-3092P, Section 3.1

Please explain the details of test data reduction process and its modification to account for [[

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AREVA Response

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SNPB RAI-6: ANP-3092P, Sections 3.2 and 3.3

6.a) *It appears that the NRC staff safety evaluation (SE) report for Revision 3 of EMF-2209(P)(A), "SPCB Critical Power Correlation" is a combination of the SE for EMF-2209(P) Revision 2 Addendum 1. In the Revision 1 of EMF-2209 safety evaluation, the NRC staff has imposed four conditions when the SPCB correlation is used for licensing applications.*

Describe how these conditions are satisfied when the SPCB correlation is used to compute the thermal margin performance for GE14 fuel at MNGP.

AREVA Response

The four conditions and how they are satisfied when the SPCB correlation is applied to AREVA fuel are addressed in Section 2-22 of ANP-3224P (which was provided with the fuel transition LAR, Reference 1 Enclosure 6). The application of the SPCB correlation to Monticello GE14 fuel (SPCB/GE14) follows topical report EMF-2245(P)(A). The ranges of applicability identified in the first three conditions in EMF-2209(P)(A) represent the ranges of parameters available to AREVA when SPCB was developed for AREVA fuel. The ranges of some parameters for which GE14 critical power was available were reduced. Therefore, when applying SPCB to GE14 fuel, the ranges of applicability are equal to or more restrictive than the ranges of applicability identified in the four conditions in EMF-2209(P)(A).

Condition 1 states the SPCB correlation is applicable with a design local peaking factor no greater than 1.5. SPCB/GE14 is applicable with a design local peaking factor no greater than 1.427.

Condition 2 states "If in the process of calculating the MCPR safety limit, the local peaking factor exceeds 1.5, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod by rod basis." For SPCB/GE14, an additional uncertainty of 0.068 is imposed when the local peaking factor exceeds 1.427 for the MCPR safety limit.

Condition 3 states "The SPCB correlation range of applicability is 571.4 to 1432.2 psia for pressure, 0.087 to 1.5 Mlb/hr-ft^2 for inlet mass velocity and 5.55 to 148.67 Btu/lbm for inlet subcooling." Based on the range of parameters for GE14 critical power information that were available to AREVA, the ranges of applicability for SPCB/GE14 are reduced to 800 to 1300 psia for pressure, 0.18 to 1.5 Mlb/hr-ft^2 for inlet mass velocity and 5.55 to 100 Btu/lbm for inlet subcooling. Appendix G of ANP-3224P presents a conservative method for extending the low pressure boundary for SPCB/GE14 to 571.4 psia.

6.b) *EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" describes the processes for the application of approved SPC (AREVA) BWR critical power correlations to the co-resident fuel remaining from prior reloads (such as GE14 fuel at MNGP). The two processes that are presented in the topical report are the indirect process and the direct process.*

Please identify which process is used at MNGP and describe the steps used in the process implemented.

AREVA Response

The indirect approach was used. AREVA previously used the indirect approach to develop additive constants and uncertainties in order to apply the SPCB correlation for GE14 fuel. For Monticello, GNF provided correspondence stating that the critical power performance previously furnished for GE14 fuel is applicable for the GE14 fuel with shorter heated length in Monticello without adjustment. Therefore, the SPCB correlation is being applied to the MNGP GE14 fuel with the same additive constants and uncertainties that were applied to GE14 fuel during the AREVA fuel transition for a previous BWR. The only difference is the conservative method for extending the low pressure boundary as described in Appendix G of ANP-3224P.

Development of the SPCB/GE14 correlation followed the steps described in Section 3.1 of EMF-2245(P)(A). The following overview is provided.

1. The SPCB correlation was used to predict critical power for each set of condition for which GE14 critical power was known. The range of these conditions established the ranges of applicability mentioned in response to RAI-6.a.
2. The calculated critical power data were then used to establish the appropriate additive constants for the co-resident fuel using the approved procedures for the critical power correlation being used.
3. The additive constant uncertainty (i.e., standard deviation) for the GE14 fuel was determined using the equations identified on page 3-2 of EMF-2245(P)(A). The additive constant uncertainty for SPCB/GE14 was calculated to be **[** **]** for the

MCPR safety limit. As mentioned in response to RAI-6a, [ ] for the MCPR safety limit.

6.c) *From Tables 3-5 and 3-6 of ANP-3092(P), it can be seen that for the transition core off-rated conditions (59.2%P/43.3%F) thermal-hydraulic results critical power ratio (CPR) has a higher margin than the CPR for rated conditions.*

Please explain the reason for this higher margin for off-rated conditions of power and flow.

AREVA Response

Since the calculations in Tables 3-5 and 3-6 use the same assembly peaking factor, the assembly power is Table 3-6 is 59.2% of the assembly power in Table 3-5. CPR is the ratio of the assembly power which would result in boiling transition at some location (Critical Power) to the actual assembly power. The CPR margin increased in Table 3-6 because the reduction in critical power resulting from the reduction in assembly flow, along with the changes in pressure, subcooling etc, was less than 59.2%. In other words, the reduction in assembly power increased the margin to boiling transition more than the reduction in assembly flow and other factors decreased the margin to boiling transition.

Similar results are seen for actual operating conditions of GE14 fuel in MNGP. One example is when the core maneuvered from 100% pre-EPU core power and 88.3% core flow to 78% pre-EPU core power and 66.6% core flow. At the initial conditions the CPR of the limiting GE14 assembly was 1.846 and at the next steady state conditions the CPR of the same assembly was 2.080.

SNPB RAI-7: ANP-3092P, Section 3.4

Please provide details of the analysis to determine the impact of rod bow on thermal margin at lower and higher exposures of ATRIUM 10XM fuel at MNGP.

AREVA Response

The approach described below was used in the MNGP analysis for the impact of rod bow on thermal margin over the entire range of exposures. AREVA uses the NRC approved correlation described in topical report XN-75-32(P)(A) Supplement 1 (Reference 7). The correlation was developed [ ] at the request of the NRC as discussed in Reference 7. [

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AREVA's BWR rod bow CPR penalty was derived using open literature data, Based on this data, it was concluded that thermal margins were not substantially reduced for closures as low as 0.03 inch.

AREVA's model application for ATRIUM 10 type fuel was presented in an informational submittal to the NRC (Reference 11).

The MCPR penalty (decrease in MCPR) versus rod bow (% closure) for the ATRIUM 10 fuel design is presented in the Figure 2. To assure that this model is conservative, AREVA ran a CHF test on an ATRIUM-10 bundle in which two rods were welded together. The measured MCPR penalty is also shown in Figure 2. This shows that the measured MCPR penalty for 100% closure was less than the MCPR penalty predicted by the model.

The impact of rod bow on thermal margin of ATRIUM 10XM fuel at MNGP is provided in response to SNPB RAI-10.

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Figure 2 – MCPR Penalty Model vs. Test Data

SNPB RAI-8: ANP-3119P, Table 2-1

Table 2-1 of ANP-3119P lists the active fuel length for full length rods and part length rods of ATRIUM 10XM design as 145.24 inches and 75.0 inches, respectively. The active fuel length for full length rods for ATRIUM 10XM fuel design approved for an earlier plant application and for fuel rods for another plant currently under review is 150 inches.

Please provide a summary of the impact on mechanical, thermal and neutronics characteristics of the reduced active full length designed for MNGP unit as compared to AREVA NP's previous designs, both approved and under review at the agency.

AREVA Response

The 3% difference in full-length fuel rod length between 145.24 inches and 150 inches is considered to be a minor difference. There is not expected to be any significant impact on mechanical, thermal, and neutronic characteristic associated with this difference in full-length fuel rod length. Assemblies with 145.24 inch full-length fuel rods and assemblies with 150 inch full-length rods, meet all requirements of the NRC approved Generic Mechanical Design Criteria (ANF-89-98(P)(A)) and are, therefore, fully qualified for use in Boiling Water Reactors. All AREVA Analyses supporting the Monticello LAR are performed explicitly modeling the full-length rods as 145.24 inches. There are no evaluations that extrapolate performance based on analyses of 150 inch full-length rods.

SNPB RAI-9: ANP-3119P, Sections 3.3.1 and 3.3.9

9.a) *Please provide a summary of the stress evaluation analysis performed to confirm the design margin and the establishment of a baseline that added accident loads. Also provide the results of the evaluation analysis that show that the assembly structural component criteria are maintained under normal and faulted conditions.*

AREVA Response

As discussed in ANP-3119P Section 3.3.1, [ ] the fuel assembly structural components do not receive significant loads during normal and AOO conditions. [

] No analyses are performed to confirm design margin under normal operating and AOO conditions [ ] The following text describes how AREVA's approved methodology was conservatively applied for the Monticello ATRIUM 10XM stress evaluation.

To ensure the structural integrity of [ ] Section III of the ASME Boiler and Pressure Vessel code (Reference 13) is used to establish acceptable design limits. To evaluate the stresses under normal operating conditions, [

] The maximum normal operation [
] for MNGP is then compared against the limit to ensure that adequate margin is maintained.

To evaluate the stress under AOO and accident conditions, [

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For the [ ] the normal operating stresses [ ] The design margin is confirmed by comparing the resulting stress to the design limit as defined by Section III of the ASME Boiler and Pressure Vessel code (Reference 13).

Information on the stress evaluation results and comparison to the load limits that show that the assembly structural component criteria are maintained under faulted conditions can be found in Table 3-1, Section 3.4.4 of ANP-3119P.

9.b) *Please provide supporting analysis and results to show that structural integrity of the water channel is maintained under loads generated during normal operation and anticipated operational occurrences (AOO).*

AREVA Response

The answer to this RAI was included in the response to SNPB RAI-9 (9.a).

SNPB RAI-10: ANP-3119P, Section 3.3.5

Please provide the details of the analysis and results from the analysis that assure that the lateral creep bow of the fuel rods is not of sufficient magnitude to impact on thermal margins for the ATRIUM 10XM fuel design at MNGP.

AREVA Response

There is no CPR penalty due to lateral creep bow of the fuel rods before an assembly average exposure of 34.7 GWd/MTU. ATRIUM 10XM fuel will not reach this exposure during their first cycle of irradiation. The rod bow CPR penalty reaches 0.01 at 42.8 GWd/MTU. By this assembly exposure in the equilibrium cycle, for all three of the ATRIUM 10XM neutronic designs, the CPR margin of the cycle limiting ATRIUM 10XM assembly had increased to 0.40 or larger. This illustrates that rod bow does not affect thermal margins due to the lower powers achieved by high exposure assemblies (as mentioned on page 4-7 of Reference 11).

Additional information about rod bow is provided in SNPB RAI-7.

SNPB RAI-11: ANP-3119P, Section 3.3.8

Please provide typical calculations that show that there are large margins to assembly lift-off under normal operating conditions and faulted conditions for the transition cycles at MNGP mixed core conditions.

AREVA Response

A liftoff calculation was performed for the ATRIUM 10XM under the reactor conditions of Monticello Nuclear Generating Plant, to ensure that the following design criteria established in ANF-89-98(P)(A) (Reference 14) are met:

- For normal operation and anticipated operational occurrences (AOO), the submerged fuel assembly weight, including the channel must be greater that the hydraulic loads.
- For accident (faulted) condition the normal hydraulic plus additional accident loads shall not cause the assembly to become disengaged from the fuel support, to assure that control blade insertion is not impaired.

The calculation [

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Table 4 MNGP ATRIUM 10XM Fuel Assembly

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SNPB RAI-12: ANP-3119, Table 3-2

Please provide a description of the analysis and its results how oxidation and hydriding were accounted for in the stress and fatigue analyses.

AREVA Response

Corrosion of structural components must be conservatively bounded in strength and fatigue calculations to account for the material loss that occurs during oxidation. This is [

] were used to show that the fuel channel can withstand the duty cycle loads. There are no hydrogen uptake limits for structural components. However, AREVA performed calculations to ensure that the ductility of the water channel could support a 1% strain limit (or higher) consistent with the requirements on fuel rod cladding. Calculations have estimated the hydrogen absorption in a water channel to be at 286 ppm at the end of life. Measurements on unirradiated cladding have shown that 1% strain is achievable even with hydrogen levels in excess of 500 ppm. A large margin is therefore maintained.

SNPB RAI-13: ANP-3224P Section 2-10

NRC staff approved Topical Report, EMF-93-177(P)(A) Revision 1 with three SER restrictions and with three more restrictions carried over from Revision 0 of the TR.

Please provide details of how the SER restrictions Numbers 1, 2, 4, 5 and 6 are met for the implementation of EMF-93-177(P)(A) at MNGP.

AREVA Response

The response below provides the details showing how the SER restrictions numbers 1, 2, 4, 5 and 6 are met for the implementation of EMF-93-177(P)(A) at MNGP.

Restriction number 1:

> "The fuel channel TR (Technical Report) methods and criteria may be applied to fuel channel designs similar to the configuration of a square box with radiused corners open at the top and bottom ends. The wall thickness shall fall within the range of current designs. The channels shall be fabricated from either Zircaloy-2 or Zircaloy-4. AREVA will not use Zircaloy material for channels which has less strength than specified in the TR, and if the strength of material is greater than that in the TR, AREVA will not take credit for the additional strength without staff review."

The ATRIUM 10XM delivered to MNGP used the advanced fuel channel with 0.1 inch thick corners and 0.075 inch thick side walls made from Zircaloy-4 sheets. These [

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Restriction number 2:

> "Updates to channel bulge and bow data are permitted without review by the NRC staff; however, AREVA shall resubmit the channel bulge and bow data statistics if the two-sigma upper and lower bounds change by more than one standard deviation."

AREVA [ ] The D-lattice plants approved models were used in the analyses.

Restriction number 4:

> "The allowable differential pressure loads and accident loads should bound those of the specific plant."

AREVA performed [

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Results were summarized in Tables 3-1, 3-2 and 3-3 of Reference 12.

Restriction number 5:

> "Lattice dimensions should be compatible to those used in the analyses reported such that the minimum clearances with control blades continue to be acceptable."

AREVA's analyses used the MNGP lattice dimensions.

Restriction number 6:

> "Maximum equivalent exposure and residence time should not exceed the values used in the analyses."

AREVA's analyses [

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SNPB RAI-14: ANP-3224P Section 2-11

NRC staff approval of Topical Report BAW-10247PA was subject to five SER restrictions.

Please provide details of how the SER restriction Number 5 regarding crud deposition is handled in the case of fuel transition to ATIUM 10XM at MNGP.

AREVA Response

Please see the response to SNPB RAI-3. The topic is similar so a consolidated response was provided.

SNPB RAI-15: ANP-3224P Section 2-15

Topical report EMF-2158(P)(A) was approved by the NRC staff subject to six SER restrictions of which number 6 is "AREVA shall notify any customer who proposes to use the CASMO-4/MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1 through 4 above must be met. AREVA's notification shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system."

Please explain how this SER restriction is implemented for the ATRIUM 10XM fuel transition process at MNGP.

AREVA Response

SER Restriction 6 of EMF-2158(P)(A) is applicable when the CASMO-4/MICROBURN-B2 code system is supplied independent of an AREVA fuel contract. In the case of the MNGP LAR the CASMO-4/MICROBURN-B2 code system is supplied integral to an AREVA fuel contract. Therefore, SER Restriction 6 does not apply.

SNPB RAI-16: ANP-3224P, Sections 2-20, 2-21 and 2-22

ANF-524(P)(A) was approved by the NRC staff subject to four SER restrictions. The SPCB correlation replaced the ANFB correlation.

16.a) *Is the SPCB correlation applied to the co-resident GE14 fuel in the MNGP core during the transition to ATRIUM 10XM?*

AREVA Response

Yes, please refer also to the response to SNPB RAI-6.

16.b) *Please explain how the SER restriction number 3 regarding the CPR channel bowing penalty for non-ANF fuel (co-resident GE14 fuel) is applied for the MNGP core.*

AREVA Response

The fuel channel bow model in ANF-524(P)(A) was based on assembly exposure. SER Restriction 3 is "The CPR penalty bowing penalty for non-ANF fuel should be made using conservative estimates of the sensitivity of local power peaking to channel bow". The sensitivity of local peaking to channel bow was

calculated with a 4 bundle CASMO "colorset" calculation. Monticello MCPR Safety Limit calculations use the methodology described in Reference 16 (this was mentioned at the end of Section 2-20 in ANP-3224P).This methodology implements a model which calculates channel bow based on the difference in fluence between opposite sides of the fuel channel. The sensitivity of local peaking to channel bow is calculated with MICROBURN-B2 3D nodal powers for the bowed condition. This change in the modeling of channel bow is summarized in Reference 16 (Figure 2-1 in ANP-10307Q1P, AREVA MCPR Safety Limit Methodology Responses to RAIs).

AREVA requested and GNF provided information about channel bow for the GE14 fuel design in MNGP. This was compared to the channel bow predicted by applying the AREVA fluence gradient model to the GNF fuel assemblies. Based on this comparison, [

] These adjustments were determined and applied so that the predicted channel bow for the GNF fuel using the AREVA bow model was either in alignment with, or conservative relative to the GE channel bow information for all exposures.

SNPB RAI-17: ANP-3224P, Section 2-25

17.a) *One of the modifications made with regards to ANP-10307PA Revision 0 is to address a concern with the application of the fuel channel bow standard deviation when the fluence gradient is computed to exceed the bound of the channel measurement database. As a consequence, the fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA to determine the Safety Limit Minimum Critical Power Ratio was increased by the ratio of channel fluence gradient to the channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty was determined.*

Please explain the details of the above modification that was implemented in ANP-10307PA in connection with the fuel transition at MNGP unit.

AREVA Response

The maximum fluence gradient for each assembly in Cycle 28 was calculated by MICROBURN-B2. A few assemblies were predicted to experience a fluence gradient slightly larger than the fluence gradient used in the model benchmark. The approach previously developed to address this NRC concern (refer to RAI-17.b) was applied to MNGP. Specifically,

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17.b) *During review of Brunswick Steam Electric Plant (BSEP) ATRIUM 10XM fuel transition LAR, the NRC staff determined that the predictive model for channel bow was validated against an empirical data that was not bounding of BSEP's expected performance. To resolve this issue, the licensee for BSEP agreed to increase the channel bow uncertainty in the SLMCPR calculation for the most severely deflected fuel channels. In view of the excessive channel bow that occurred at BSEP, a license condition was proposed for BSEP Units 1 and 2 in connection with the use of AREVA channel bow model outside the range of the channel bow measurement database from which its uncertainty was quantified (Reference: Letter, BSEP 13-0002, from Michael J. Annacone (Duke Energy) to NRC, "Supplement to License Amendment Request for Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, CORE OPERATING LIMITS REPORT (COLR), and Revision to Technical Specification 2.1.1.2 Minimum Critical Power Ratio Safety Limit," Duke Energy, January 22, 2013).*

Confirm whether a similar license condition is required for the MNGP unit for the fuel transition.

AREVA Response

By virtue of its inclusion in the analysis submitted in the LAR (i.e., Section 2-25 of ANP-3224), the licensee has accepted the channel bow uncertainty in the SLMCPR calculation as an element of the methodology. Therefore, no license condition is required for MNGP.

SNPB RAI-18: ANP-3224P, Section 2-29

Please provide a latest revision of the methodology and code manual for XCOBRA-T code that is currently used in the thermal hydraulic core analysis for the fuel transition in the MNGP core.

AREVA Response

The latest revision of the XCOBRA-T methodology is identified in Section 2-29 of ANP-3224P (Reference 17). During a conference call with the NRC reviewer on January 27, 2014 regarding this RAI, the NRC reviewer determined that the XCOBRA-T methodology did not need to be provided.

The latest revision of the XCOBRA-T code manual (Reference 18) is being provided with these RAI responses. This is being provided for information only and is proprietary in its entirety.

SNPB RAI-19: ANP-3224P, Section 2-15 and Appendix A

It appears from the above mentioned sections of your submittal for ATRIUM 10XM fuel transition at MNGP unit that the licensee has used Topical Report, EMF-2158(P)(A) to calculate radial and axial power distribution measurement uncertainties.

Please provide details of analyses, calculations, and the database information used to establish these uncertainties. Also, please confirm that the uncertainties (including those for TIP distribution uncertainties) calculated for MNGP unit is in line with the uncertainties that are listed in Chapter 9 of EMP-2158(P)(A).

AREVA Response

Radial and axial power distribution measurement uncertainties used in the MNGP safety limit analyses are not taken from EMF-2158(P)(A). These uncertainties are based upon the GARDEL core monitoring uncertainties. The GARDEL core monitoring power distribution measurement uncertainties were provided in Reference 19. In Enclosure 3 of Reference 21, Xcel Energy discussed its approach to evaluate the impact of explicitly accounting for the 25% grace period for the LPRM calibration interval.

SNPB RAI-20: ANP-3224P, Appendix A

It is stated in Section A1 of the Appendix A that the methods used in CASMO-4 are state of the art. The methods used in MICROBURN-B2 are state of the art. The methodology accurately models a wide range of thermal hydraulic conditions including EPU and extended power/flow operating map conditions.

Please explain how the CASMO-4 and MICBURN-B2 calculations are applied to the extended power/flow operating map conditions.

AREVA Response

CASMO-4 calculations are performed for various void fractions and fuel temperatures. This data is used by MICROBURN-B2 to construct cross sections consistent with the operating state-point conditions of power, flow and pressure. The specific values of power, flow and pressure are input values used in the calculation. The core design engineer utilizes values appropriate for the state-point to be analyzed within the constraints of the licensing restrictions of the power/flow map. Mutual solutions of the power distribution and flow distribution are used to determine the conditions of each node in the core.

SNPB RAI-21: ANP-3224P, Appendix D

In page D-2 it is stated that the multi-rod database used in the [[

]]. As a result, the multi-rod database and prediction uncertainties are not available to AREVA. However, the correlation has been independently validated by AREVA against public domain multi-rod data and proprietary data collected for prototypical ATRIUM-10 and ATRIUM 10XM test assemblies. Selected results for the ATRIUM-10 test assembly are reported in the public domain in Reference 42.

The NRC staff would like to review the data that has validated the correlation and requests a copy of the Reference 42 that is listed in ANP-3224P.

AREVA Response

The requested reference is provided as an enclosure to the Xcel Energy response.

SNPB RAI-22: ANP-3138P

NRC staff has performed the review of the supplemental topical report, ANP-10298PA Revision 0, Supplement 1P Revision 0, "Improved K-Factor Model for ACE/ATRIUM 10 XM Critical Power Correlation," December 2011. If the final approval is done before the implementation of the fuel transition at MNGP unit, it may be advantageous to list the approved supplement to the MNGP COLR/TS.

Note: This RAI is advanced information for the licensee to monitor development of the approval process for the supplemental TR.

AREVA Response

When NRC approval of ANP-10298PA Revision 0, Supplement 1P Revision 0, "Improved K-Factor Model for ACE/ATRIUM 10 XM Critical Power Correlation," December 2011 is obtained, Xcel Energy will provide revised documentation to incorporate this document in the license basis for the Fuel Transition LAR.

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 Regarding License Amendment Request to Transition to AREVA ATRIUM 10XM Fuel and Safety Analysis Products (TAC No. MF2479), ML13200A185.

3. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP Inc., February 2008.

4. XN-NF-79-59(P)(A), *Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies*, Exxon Nuclear Company, November 1983.

5. XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.

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

7. XN-75-32(P)(A) Supplements 1 through 4, *Computational Procedure for Evaluating Fuel Rod Bowing,* Exxon Nuclear Company, October 1983. (Base document not approved.)

8. XN-NF-82-06(P)(A) Supplement 1 Revision 2, *Qualification of Exxon Nuclear Fuel for Extended Burnup,* Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel", May 1988.

9. EMF-85-74(P), Revision 0, Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,* Siemens Power Corporation, February 1998.

10. "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (Revision 1)," NRC Report dated February 16, 1977.

11. EMF-95-52(P) Revision 1, *Fuel Design Evaluation for Siemens Power Corporation ATRIUM™-10 BWR Reload Fuel*, April 1998, transmitted to the NRC by Siemens Power Corporation Letter, "Design Evaluations for SPC ATRIUM™-9B and ATRIUM™-10 Fuel", April 8, 1998, (NRC:98:021).

12. ANP-3119P Revision 0, *"Mechanical Design Report for Monticello ATRIUM 10XM Fuel Assemblies",* AREVA Inc., October 2012.

13. *ASME* Boiler and Pressure Vessel Code, Section III, Division 1, American Society of Mechanical Engineers.

14. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation,* May 1995.

15. EMF-93-177(P)(A), Revision 1, *Mechanical Design for BWR Fuel Channels,* August 2005.

16. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.

17. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2 Revision 0, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Code Analysis*, Exxon Nuclear Company, February 1987.

18. EMF-CC-167(P) Revision 8, *XCOBRA-T Theory, Programmer's and User's Manual*, AREVA NP, November 2011.

19. Response to Requests for Additional Information (RAI) for the License Amendment Request to Revise the Minimum Critical Power Ratio Safety Limit in Reactor Core Safety Limit 2.1.1.2 (TAC No. ME4790), February 8, 2011, MNGP L-MT-11-009, ML110450240.

20. EMF-95-52(P) Revision 1, *Fuel Design Evaluation for Siemens Power Corporation ATRIUM™-10 BWR Reload Fuel*, April 1998, (transmitted to the NRC by Siemens Power Corporation Letter, "Design Evaluations for SPC ATRIUM™-9B and ATRIUM™-10 Fuel", April 8, 1998, NRC:98:021).

21. AREVA ATRIUM 10XM Fuel Transition – Responses to Request for Additional Information (TAC MF2479), January 31, 2014, MNGP L-MT-14-003, ML14035A297.