Controlled Document

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Revision 0

AREVA Responses to RAIs on the ATRIUM 10XM Compliance Audit and Brunswick LARs

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AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.		This is a new document.

Contents

Figures

This document contains a total of 19 pages.

Nomenclature

ASTM	American Society for Testing and Materials
BWR	Boiling water reactor
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CRWE	Control rod withdrawal error
FDL	Fuel design limit (LHGR limit)
LAR	Licensing Amendment Request
LHGR	Linear heat generation rate
LOCA	Loss-of-coolant accident
NRC	U.S. Nuclear Regulatory Commission
OFU	Operating flexibility uncertainty
PWR	Pressurized water reactor
RAI	Request for Additional Information
TMOL	Thermal-mechanical operating limit

1.0 Introduction

The NRC (Nuclear Regulatory Commission) provided RAIs (Requests for Additional Information) following a regulatory audit on November 2-5, 2010 (Reference 1). The questions relate to AREVA NP Inc. (AREVA) analytical methodology addition LARs (Licensing Amendment Requests) for the Brunswick Units 1 & 2 Technical Specifications (References 2 and 3) and ATRIUM™* 10XM fuel introduction. Responses to the RAIs are provided below in Section 2.0.

This is the second set of RAIs associated with the LARs. Responses to the first set of RAIs were provided by Reference 4.

* ATRIUM is a trademark of AREVA NP Inc.

2.0 NRC Questions and AREVA Responses

The NRC RAIs are listed below and each numbered question is followed by a response.

1. NUREG-0800 SRP-4.2 provides guidance on the treatment of corrosion, crud, and hydrogen content. Specifically, limits for each assembly component should be specified based on "mechanical testing to demonstrate that each component maintains acceptable strength and ductility." Furthermore, the thermal effects of both corrosion and crud need to be specifically accounted for within fuel rod thermal-mechanical design analyses and inputs to downstream safety analyses (e.g., LOCA stored energy). The maximum cladding oxidation design limit for the ATRIUM 10XM fuel design (ANP-2899P) is listed as [ ]

   a. The staff has adopted a corrosion/crud design limit of 100 microns. The corrosion/crud data are based on averaged values axially and circumferentially along a section of fuel rod. Please justify the [ ] design limit.

   b. Formation of localized cladding corrosion defects (e.g., nodular corrosion, oxide spallation, hydride lenses) promotes non-uniform cladding mechanical properties and may invalidate fuel performance models. Design limits on cladding oxide thickness are based on avoiding these localized defects and are based on pool-side inspections and measurements. Please demonstrate that localized cladding corrosion defects are not present at an oxide thickness of [ ]

   c. The level of cladding hydrogen content is limited to ensure the continued applicability of the cladding strain SAFDL. Design limits on cladding hydrogen content are based on mechanical testing (e.g., burst testing) performed on irradiated cladding. Please justify the design limit on cladding hydrogen corresponding to [ ] and the applicability of the cladding strain SAFDL.

   d. Explain the statement in Section 3.3.4 of ANP-2948P, Rev. 0 that there is no specific corrosion limit.

AREVA Response:

a. The [ ] originally was based on acceptable rod performance up to [ ] in PWRs. The criterion was conservatively set at [ ] The limit was established using poolside measurement data and it is associated with the data measurement and data reduction techniques along with the methodology for calculating the maximum corrosion. The criterion was developed independently of the approved limits for other fuel vendors during a time when no defacto industry limit existed. A comparison of the [ ] limit and the 100 μm limit adopted by the staff cannot be made without considering the definition of the oxide measurement and the analytical approach to calculate the oxide thickness. It is believed that the AREVA method of measurement in conjunction with the statistical analysis leads to a more conservative limitation on corrosion than the approach adopted by the NRC.

The [ ] was first approved by the NRC in 1991 for PWR fuel supplied by Advanced Nuclear Fuels Corporation (now AREVA NP Inc.) for application to rod exposures of 62 MWd/kgU (Reference 5). The PWR limit continues to exist for AREVA fuel licensed under the generic design criteria, EMF-92-116(P)(A) (Reference 6). Until 2008, no corrosion limit was specified for the AREVA BWR fuel. The allowance for not having a corrosion limit on BWRs was approved on the basis that corrosion levels exhibited by the AREVA BWR fuel were low such that a specific corrosion limit was not necessary (Reference 7). Instead, it was required that [ ] In 2004, AREVA submitted the RODEX4 topical report, BAW-10247PA that included the [ ] for BWR fuel rods. This limit was approved for BWR applications using RODEX4 in 2008 (Reference 8).

The limit is associated, in part, with the method of liftoff measurement and data reduction. The liftoff measurements originally used in deriving the [ ] Please see the prior RAI response for BAW-10247PA (first set of questions, RAI #31, Reference 8) where this same subject was examined as part of the RODEX4 review and approval. The liftoff measurement method is described in detail in Section 4.3 of Reference 5.

Rather than calculating a best-estimate corrosion thickness for comparison to a limit of 100 µm, the [ ]

[

] is described in RAI #17c in the same set of RAI questions referenced above (i.e., first set of questions, RAI #17c, Reference 8). Had a best-estimate calculation been performed instead in the RODEX4 methodology, the calculated oxide thickness would be approximately [

]

b. RODEX4 does not model localized effects such as nodular corrosion, oxide spallation or the presence of hydride lenses. Minor changes made in the heat treatment of the cladding and optimization of alloying elements within the ASTM limits for Zircaloy-2 have led to significant improvements in the [

]

Oxide spallation also could be considered a local phenomenon [

]

[

]

Hydride lenses have [

]

A 100 μm limit will [

]

c. Based on the Brunswick RODEX4 analyses and [

]

[

]

d. ANP-2948P (Reference 9) covers the fuel assembly structure including the fuel channel, water channel and other components besides the fuel rods. [

]

2. 10 CFR 50.46 requires maximum cladding oxidation not to exceed 17% of the total cladding thickness. Please explain how the [ ] will be able to meet the requirement considering that the fuel rod reaches the limit before LOCA occurs.

AREVA Response:

The [ ] is on the thickness of the oxidation layer while the 10 CFR 50.46 criterion is relative to the reduction in the cladding wall thickness. [

] Therefore, when the oxidation thickness is [ ] The cladding wall thickness for the ATRIUM 10XM fuel is [ ] An oxidation layer thickness at the design limit would correspond to a cladding wall reduction of [ ] Adding the peak metal water reaction of 0.99% from the Brunswick ATRIUM 10XM LOCA analysis reported in ANP-2943P (Reference 12) results in a total local oxidation of [ ] It is noted that the metal water reaction occurring during the LOCA event is [ ] In addition, the Brunswick peak local metal water reaction result occurs at beginning of life when the pre-accident oxide layer thickness is at a minimum.

3. The COLR Thermal Mechanical Operating Limits (TMOL) provides the limits on fuel rod power history used in the fuel design analyses. These rod power limits are surveilled by plant operators during normal operation, power maneuvering, and control blade movements.

    a. Best-estimate FRAPCON-3 calculations (nominal parameters and no modeling uncertainties) performed by the staff predict UO_2 fuel rod internal pressure above 1800 psia at end of life based on segmented power histories (rod power at 90% of TMOL and adjusted to 100% of TMOL for 1/6 exposure). Please justify the UO_2 fuel rod TMOL. Include a plot of peak local rod power (KW/ft) versus local burnup which includes the 10 highest axial nodal power histories in each time step used in the statistical analysis.

    b. Please discuss the design power histories and surveillance of (UO_2, Gd_2O_3) fuel rods.

AREVA Response:

a. It is very likely that the difference in results from FRAPCON and RODEX4 is due to the power history input. [ ] at the Brunswick plants are used in the RODEX4 analyses according to the approved methodology described in Reference 8. The power histories' levels are [ ] The TMOL (Thermal-Mechanical Operating Limit), or the FDL (Fuel Design Limit) given in the AREVA reports, is used as part of the [ ] to the thermal-mechanical analyses. According to the approved RODEX4 methodology, [

-]

[

]

[

]

Figure 2-1 Maximum LHGR Values from RODEX4 Power Histories Compared to the Fuel Design Limit

b. As previously discussed in RAI response 11 (Reference 4), the urania and gadolinia fuel within a fuel assembly are analyzed and monitored to the same FDL. Plant- and cycle-specific analyses are performed on the specified nuclear design to ensure that the fuel melting temperature criterion along with other fuel rod design criteria are satisfied. [

] If necessary to satisfy the design criteria, the urania or gadolinia rods may have enrichment setbacks to limit their local power peaking within the fuel assembly. Alternately, the FDL itself can be reduced until all design criteria have been met.

During operation, the core monitoring software calculates the LHGR for each fuel rod axial segment for comparison to the corresponding exposure-dependent FDL. As described above, [

] all rods in any given fuel assembly are monitored for compliance with the FDL that is validated by RODEX4. Thus, the fuel rod thermal-mechanical analyses take into account the anticipated operation of both the urania and gadolinia fuel rods.

4. Describe the control rod withdrawal error and flow runup events, and how these events are chosen for slow AOO transients as the limiting cases.

AREVA Response:

Slow AOO transients are those that occur over a period of time that is typically measured in minutes. Analyses of these events require assumptions that no operator interaction occurs that would terminate the event before the maximum consequence is observed. The control rod withdrawal error (CRWE) represents a very challenging localized event, whereas the flow runup represents a challenging core wide event. For LHGR consequences, the CRWE and flow runup events are the most serious and bound other slow transients.

The control rod withdrawal error event involves the selection of a control rod for withdrawal during power operation that is not part of the planned withdrawal sequence. The operator then proceeds to withdraw this error rod and ignores any resulting alarms. The event is not terminated until either: 1) the Rod Block Monitor (or Rod Withdrawal Limiter for BWR/6's) applies a rod block that prohibits any further rod withdrawal, or 2) the rod is fully withdrawn from the core (if a rod block is not applied).

The slow recirculation flow runup is a prescribed event in which a failure of the recirculation flow control system is postulated. The postulated failure results in a slow increase in the recirculation flow (typically in all loops) until the maximum flow condition is reached. Alternatively, the event may be terminated by reaching a reactor protection system (RPS) limit resulting in a scram (e.g., reaching a high flux scram may terminate the event prior to reaching full flow conditions).

In the steady state RODEX4 analysis, the LHGR consequences for both are maximized in the power history files which are input to the RODEX4 analysis. The LHGR consequences for each event consider the combination of the core power excursion triggered by the event in conjunction with localized power increase due to control blade removal.

For the CRWE simulation, events are modeled [

]

For the flow runup, events are simulated [

]

Simulating the CRWE and flow runup events as discussed above produces the most severe changes in LHGR to be analyzed by RODEX4 for slow events.

5. a. Explain in detail, how the gadolinia ($UO_2 + Gd_2O_3$) rods are treated in RODEX2-2a analyses to support the ATRIUM 10XM LOCA analyses.

   b. Give details of the LOCA analysis procedure preventing the gadolinia rods from becoming hot rods.

AREVA Response:

a. The NRC approved methodology used in the Brunswick ATRIUM 10XM LOCA analysis includes RODEX2-2a calculations (Reference 13).

   In LOCA analyses, RODEX2-2a is used to calculate fuel characteristics used in RELAX and HUXY. The material properties used in the RODEX2-2a calculation explicitly include the effects of gadolinia. RELAX is used to calculate the system and hot channel response during a LOCA. [

   ]

   HUXY is used to calculate the exposure-dependent peak clad temperature and the maximum metal water reaction using the boundary conditions from the RELAX calculations. [

   ]

b. As noted above, [ ] As a result, the acceptance criteria are met for all the fuel rods, including the gadolinia rods. [

   ]

3.0 References

1. Email from Farideh Saba (NRC) to William Murray (Brunswick Steam Electric Plant), REQUEST FOR ADDITIONAL INFORMATION BASED ON REGULATORY AUDIT IN NOVEMBER 2 – 5 2010, BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2, LICENSE AMENDMENT REQUEST FOR ADDITION OF ANALYTICAL METHODOLOGY TO TS 5.6.5 COLR, TAC NOS. ME3856, ME3857, ME3858, ME3859," dated November 18, 2010. (38-9149776-000)

2. Progress Energy Letter BSEP 10-0057, *Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324 Renewed Facility License Nos. DPR-71 and DPR-62 Request for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)"*, April 29, 2010.

3. Progress Energy Letter BSEP 10-0052, *Brunswick Steam Electric Plant, Unit Nos. 1 and 2 Renewed Facility License Nos. DPR-71 and DPR-62 Docket Nos. 50-325 and 50-324 Request for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)"*, April 29, 2010.

4. Progress Energy Letter BSEP 10-0133, *Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Response to Additional Information Request Supporting License Amendment Requests for Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5 (NRC TAC Nos. ME3856, ME3857, ME3858, and ME3859)*, November 18, 2010.

5. ANF-88-133(P)(A) and Supplement 1, *Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU*, Advanced Nuclear Fuels Corporation, December 1991.

6. EMF-92-116(P)(A) Revision 0, *Generic Mechanical Design Criteria for PWR Fuel Design, Siemens Power Corporation*, February 1999.

7. 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.

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

9. ANP-2948P Revision 0, *Mechanical Design Report for Brunswick ATRIUM 10XM Fuel Assemblies*, AREVA NP Inc., October 2010.

10. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Siemens Power Corporation, April 1995.

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

12. ANP-2943P Revision 0, *Brunswick Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™ 10XM Fuel*, AREVA NP Inc., September 2010.

13. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model*, Framatome ANP, May 2001.