ATTACHMENT 2

Response to NRC Nuclear Performance and Code Review Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

NON-PROPRIETARY VERSION

(Cover Page Plus 18 Pages)

Controlled Document

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Attachment 2

ANP-3028(NP)
Revision 0

St. Lucie Plant Unit 1 EPU RAIs – Nuclear Performance & Code (SNPB)

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AREVA NP Inc.

AREVA

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Revision 0

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Nature of Changes

Item	Page	Description and Justification
1.	All	Initial Release

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Nomenclature

Acronym	**Definition**
AOO	Anticipated Operational Occurrence
CE14	Combustion Engineering 14x14
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
EM	Evaluation Model
FCM	Fuel Centerline Melt
GWd/mtU	Giga-Watt day per metric ton of Uranium
HTP	High Thermal Performance
kW/ft	kilo-Watt per foot
LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LHR	Linear Heat Rate
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
MWd/mtU	Mega-Watt day per metric ton of Uranium
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
SBLOCA	Small Break Loss of Coolant Accident
UO_2	Uranium Dioxide
Zr-4	Zircaloy-4

1.0 Introduction

This report documents the responses to the Request for Additional Information (RAI) on the Reactor Systems Section (2.8) of the St. Lucie Plant – Unit 1 Extended Power Uprate (EPU) Project License Amendment Request (LAR). The RAIs consist of questions SNPB 1 through 7.

2.0 Reactor Systems RAIs Questions and Responses

Question SNPB-1: (Sections 2.8.1, 2.8.2, and 2.8.3 MONOBLOC Guide Tube)

(a) It is stated that the "MONOBLOC guide tubes increase the fuel bundle stiffness and therefore increase the mechanical performance of the assembly." Please provide supporting analyses to justify the statement that the MONOBLOC guide tubes improve the mechanical performance of the fuel assembly.

(b) Provide details of the pressure drop, liftoff, and seismic tests and analyses performed on CE14 HTP assemblies with MONOBLOC guide tubes that are proposed for St. Lucie Unit 1 EPU. The details of the results must substantiate your claim that the transition to MONOBLOC design does not constitute a mixed core situation from a mechanical analyses perspective at St. Lucie Unit 1.

Response SNPB-1:

(a) Mechanical testing on full scale prototype fuel assemblies has been performed by AREVA using standard and MONOBLOC guide tubes. This testing shows that the lateral fuel assembly stiffness is slightly greater for the MONOBLOC design (~ [ ]%). As a result, lateral deflections, impacts, and bending stresses will be lower than for the standard design. However, the calculation of margins to stress limits conservatively did not consider this increase in lateral stiffness for the fuel assembly design using MONOBLOC guide tubes.

(b) Pressure drop testing on full scale prototype fuel assemblies has been performed by AREVA using standard and MONOBLOC guide tubes. The measured pressure drop of the inlet, rodded, and outlet regions of the two assembly types have been shown to be equal (within the test measurement uncertainty of [ ]%); therefore the liftoff performance of both designs is considered to be the same, and transition to the MONOBLOC design is not considered to introduce a mixed core situation.

Mechanical testing on full scale prototype fuel assemblies using standard and MONOBLOC guide tubes shows that the fuel assembly dynamic properties (lateral and axial stiffness, natural frequencies, damping, etc.) of the two designs are very similar. To conservatively determine the faulted condition component margins, the largest stiffness increase ([ ]) was used to scale (increase) the component loads determined in previous fuel assembly seismic and LOCA analyses for both fuel types (standard and MONOBLOC). Positive margin to allowable stress was maintained for all fuel assembly components.

Question SNPB-2: (Section 2.8.1.2.3.1, Fuel Rod Analyses)

The licensee states that "the NRC-approved fuel rod performance models (RODEX2 and RAMPEX) were used to model anticipated EPU operating conditions and the fuel characteristics of the CE14 HTP design with MONOBLOC or standard guide tubes," The NRC staff has identified concerns that RODEX2 code lacks a fuel thermal conductivity model that accurately captures the fuel thermal conductivity degradation associated with increasing burnup.

Pursuant to Criterion 10 (Reactor Design) and Criterion 35 (Emergency core cooling) of Appendix A to 10 CFR 50 and the requirements for ECCS Evaluation models, the licensee is expected to incorporate a methodology for calculating the highest cladding and fuel temperatures and thereby the highest calculated stored energy in the fuel during any condition of normal operation including AOOs and for ECCS evaluation models. This methodology shall include the evaluation of thermal conductivity of the fuel as a function of burnup and temperature taking into consideration all of the effects that take place in the fuel during irradiation including but not limited to solid fission product buildup both in solution and as precipitates, porosity, and fission gas-bubble formation. This evaluation shall also include the effects of thermal conductivity on all fuel rod thermal-mechanical analyses (e.g. rod internal pressure) and inputs to downstream safety analyses (e.g. LOCA stored energy).

(a) Describe, in detail, the licensee's efforts to evaluate fuel thermal conductivity as a function of burnup and temperature considering all of the effects that take in the fuel during the irradiation in the reactor core.

(b) Does the thermal conductivity degradation treatment described in Section 6.0 of ANP-2903(P), Revision 1, "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA Summary Report with Zr-4 Fuel Cladding," consider effectively and realistically all the transformations that take place in the fuel during irradiation in the reactor core?

Response SNPB-2:

(a) [

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Fuel Centerline Melt (FCM)

The impact on FCM temperatures and limits resulting from the lack of modeling of the degradation of fuel thermal conductivity with burnup was calculated with a code-to-code comparison between the RODEX2 and the COPERNIC fuel performance codes. COPERNIC (Reference 6) is a contemporary NRC-approved fuel performance code that models exposure dependent degradation of fuel thermal conductivity. By comparing RODEX2 and COPERNIC results using the appropriate approved methodologies for each of the two codes and a consistent set of conditions, penalty factors were developed as a function of burnup for application to RODEX2 FCM temperature.

Temperature penalties to the melt limit were calculated as a function of fuel rod average burnup and fuel rod type (urania-gadolinia concentration) in a manner such that the burnup dependent FCM limits predicted by RODEX2 with the reduced melt limits are bounded by the FCM limits calculated with COPERNIC.

(b) The development of the bias and uncertainty used in the best estimate LOCA evaluation model for the fuel temperature predictions of RODEX3A is described in detail in Section 6.1 of the Summary Report ANP-2903, Revision 1 (Reference 7), under the subtitle "Thermal Conductivity Degradation Related Issues." As described in Section 6.1, the database used is that used to qualify and obtain NRC approval of RODEX4 and includes exposures up to 70 GWd/mtU well beyond the 62 GWd/mtU licensing limit for current fuel. Therefore, the effects of burnup on the fuel performance code predictions and the expected variance of those predictions are effectively incorporated into the best estimate LOCA evaluations.

Question SNPB-3: (Section 2.8.1.2.3.1.2, Cladding Collapse)

Provide a summary of the 'cladding creep analysis' that was performed to verify that the pellet to cladding gap does not close before completion of fuel densification.

Response SNPB-3:

The creepdown and ovalization of the cladding are evaluated using AREVA's RODEX2 (References 1 and 2) and COLAPX codes on a cycle specific basis. The RODEX2 code is used to calculate the cladding creepdown and provides initial in-reactor fuel rod conditions to COLAPX. The COLAPX code calculates the cladding ovality changes as a function of time. The combination of cladding creepdown and ovality are calculated, and at a rod average burnup of [ ]. This will prevent pellet hangups due to cladding creep, allowing the plenum spring to close axial gaps until densification is substantially complete. AREVA's creep collapse methodology is presented in Reference 4 along with the basis for calculating cladding creep collapse at a rod average burnup of [ ]. The methodology uses minimum cladding wall thickness along with maximum fuel densification, minimum rod pre-pressurization, no fission gas release, and "high" first cycle powers in order to calculate conservative creep collapse margins. Creep collapse analysis performed by AREVA shows that all fuel scheduled for operation at EPU conditions satisfy the criterion.

Question SNPB-4: (Section 2.8.1.2.3.1.5, Stress and Strain Limits)

Describe the highlights of the 'ramping analysis' and discuss how the results of the analysis and calculations confirmed that all stresses and strain limits continue to meet for all fuel operating at the EPU conditions.

Response SNPB-4:

AREVA does not have a design criterion for cladding transient stress, but does have a criterion for transient (AOO) strain. During a simulated anticipated operational occurrence (AOO), the total uniform transient strain (elastic + plastic) of the fuel rod cladding is analyzed and must be limited to less than [ ]. AREVA evaluates cladding strain using the RODEX2 code (Reference 1 and 2). For overpower conditions, the ratio of the overpower limit to the maximum allowable steady-state rod power during the exposure provides a factor for "spiking" the rod power. This factor is applied as a momentary excursion in power at various exposures throughout the design history. RODEX2 returns to the condition of the rod before the spike and continues the steady-state analysis until the next power spike. In this way, preceding spikes do not affect subsequent spikes in the analysis. AREVA's cladding AOO strain analysis methodology is presented in Supplement 1 of Reference 5. The methodology uses minimum cladding inner diameter along with the maximum pellet outer diameter (minimum pellet-to-clad gap) for earlier pellet-to-clad contact, maximum pellet density resulting in a harder pellet, and minimum cladding thickness for faster creepdown. AOO strain analysis performed by AREVA shows that all fuel scheduled for operation at EPU conditions satisfy the criterion.

Question SNPB-5: (Section 2.8.1.2.3.1.6, Cladding Rupture)

Describe briefly how compliance to the cladding ballooning and rupture criteria is met in the licensee's evaluation of 'ECCS and Loss of Coolant Accidents'.

Response SNPB-5:

AREVA's current approved best estimate LBLOCA Evaluation Model (EM) (Reference 8) does not model ballooning and rupture as they would potentially occur during a LOCA transient because the occurrence of swelling and rupture, reasonably modeled, acts to reduce the calculated cladding temperatures. This position, as described in Section 6.3 of the of the Summary Report ANP-2903, Revision 1 (Reference 7), is that for ruptures occurring during reflood detrimental effects of swelling, rupture, and fuel relocation are more than compensated by the rupture induced cooling benefits. In applying this methodology, the fuel rod thermal response is reviewed for the occurrence of rupture during the blowdown and refill phases. For St. Lucie Unit 1, rupture does not occur prior to the initiation of reflood.

AREVA's current approved SBLOCA EM (Reference 9) is a conservative Appendix K deterministic evaluation (demonstrated in Reference 10) and models ballooning and ruptures in accordance with the requirements of the Appendix K approach using the NUREG-0630 model for Zr-4 cladding or the equivalently constructed M5 model for M5 cladding.

Question SNPB-6: (Section 2.8.1.2.3.2.1.11, and 2.8.1.2.3.2.4, Oxidation, Hydriding, and Crud buildup)

(a) Describe briefly how the licensee's fuel rod corrosion analysis has demonstrated that all corrosion limits continue to be met for all fuel operating at EPU conditions at Saint Lucie Unit 1, using References 1 and 4 listed Section 2.8.1.4 of Attachment 5 of L-2010-259.

(b) For Section 2.8.1.2.3.2.4, describe the 'corrosion enhancement factor' and how it is applied to corrosion model. What is a 'cage corrosion model'?

Response SNPB-6:

(a) AREVA's Zircaloy water corrosion methodology is presented in Reference 5. A peak local corrosion design limit of [ ] is conservatively established for Zircaloy-4 fuel rods (Reference 3). Cladding corrosion of Zircaloy-4 fuel rods as a function of burnup is calculated using the MATPRO model contained within the RODEX2 code (References 1 and 2) with appropriate enhancement factors. The best fit line to the measured corrosion data is obtained with a multiplier of [ ] to the MATPRO corrosion rate, and the 95/95 upper bound limit is obtained with an additional [ ] multiplier on the calculation results. The 95/95 upper bound results are compared with the design limit to show compliance. Corrosion analysis performed by AREVA shows that all fuel scheduled for operation at EPU conditions satisfy the criterion.

(b) The corrosion and hydrogen content of non-fueled Zircaloy cage components throughout life are determined using the RODEX2 clad corrosion and hydriding models (References 1 and 2). These models are based on MATPRO-11 correlations with the hydrogen uptake linearly related to the oxide thickness. An enhancement factor (or multiplier) of [ ] is applied to the MATPRO corrosion rate. It is assumed that sufficient mixing occurs to assure that the cage components are at the bulk coolant temperature on average throughout life. The bulk temperature is assumed to increase linearly from inlet to outlet through the fueled core region. Each cage component type is evaluated at its highest (hottest) level in the core. This methodology is benchmarked to corrosion measurements in unheated fuel rod plenum regions as well as corrosion measurements from guide tubes and spacers. The result of this calculation is then checked against a corrosion design limit of [ ] to ensure that the structural integrity of these components is maintained throughout the design life of the fuel. Cage component corrosion analysis performed by AREVA shows that all fuel scheduled for operation at EPU conditions satisfy the criterion.

Question SNPB-7: (Section 2.8.3.2.3, Description of Analyses and Evaluations)

(a) Describe the procedure how the gadolinia rods are treated during the transients and accident analyses where the fuel centerline melting is precluded.

Response SNPB-7:

(a) Gadolina fuel is protected along with UO_2 fuel by a fuel centerline melt criterion that protects all rods in the core throughout the cycle. The criterion comes in the form of a cycle specific kW/ft

limit that protects the core from centerline melt during normal operation and accidents. The approved code used for performing melt calculations is RODEX2 (References 1 and 2).

On a cycle specific basis a deterministic melt limit calculation is performed that determines a kW/ft limit for fresh fuel that envelopes all rod types. This limit is used as an input to Chapter 15 event analysis and setpoint analysis for which fuel melt may occur. A conservatively determined peak LHR for each analysis is compared to the limit and as long as it is not exceeded, fuel centerline melt is precluded.

The methodology for protecting the fuel from centerline melt is a part of AREVA's setpoint methodology and is primarily discussed in Reference 11. Appendix A of Reference 11 provides sample calculations that specifically discuss the treatment of gadolinia bearing rods. Some discussion of fuel centerline melt is also contained in Reference 12. Section 3.2 of Reference 12 discusses fuel modeling relative to UO_2 and gadolinia rods (see Hot Spot Model) along with a general discussion of fuel centerline melt relative to protection for the Non-LOCA transients.

For the impact of thermal conductivity degradation with burnup, see the response to question SNPB-2 (a).

3.0 References

1 XF-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984.

2 ANF-81-58(P)(A), Revision 2 and Supplements 3 and 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," April 1990.

3 EMF-92-116(P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," February 1999.

4 XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, & 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986.

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

6 BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

7 ANP-2903(P) Revision 1, "St Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA Summary Report with Zr-4 Fuel Cladding," May 2011.

8 EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

9 EMF-2328(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

10 ANP-3000(P) Revision 0, "St. Lucie Unit 1 EPU – Information to Support License Amendment Request," May 2011.

11 EMF-1961(P)(A) Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," July 2000.

12 EMF-2310(P)(A) Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," May 2004.