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Attachment 4

St. Lucie Unit 2 Fuel Transition
St. Lucie Unit 2 Fuel Transition: Response to SNPB-RAI-1
ANP-3428NP, Revision 0

Next 16 Pages

AREVA

ANP-3428NP
Revision 0

St. Lucie Unit 2 Fuel Transition: Response to SNPB-RAI-1

Licensing Report

July 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

Nomenclature

Acronym	Definition
BOC	Beginning of Cycle
CE	Combustion Engineering
CEA	Control Element Assembly
CRE	Control Rod Ejection
DNB	Departure from Nucleate Boiling
DTC	Doppler Temperature Coefficient
EOC	End of Cycle
EPU	Extended Power Uprate
FPL	Florida Power and Light
HFP	Hot Full Power
HZP	Hot Zero Power
LAR	License Amendment Request
LOCA	Loss of Coolant Accident
MDNBR	Minimum Departure from Nucleate Boiling Ratio
NRC	Nuclear Regulatory Commission
PIRT	Phenomenon Identification and Ranking Tables
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
SRP	Standard Review Plan
TCD	Thermal Conductivity Degradation
UFSAR	Updated Final Safety Analysis Report

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has requested additional information (Reference {1}) related to the St. Lucie Unit 2 fuel transition LAR and supporting reports (References {2}, {3}, and {4}). The AREVA methodologies used for the control element assembly (CEA) ejection accident are described in References {5} and {6}. The use of these methodologies was discussed with the NRC in a meeting on June 11, 2015. The purpose of this document is to support the NRC review of the St. Lucie Unit 2 fuel transition LAR by providing the requested additional information.

2.0 USNRC REQUEST FOR ADDITIONAL INFORMATION

2.1 *Request for Additional Information SNPB-RAI-1*

From Reference {1}:

Background

Section 4.25 of ANP-3347P, Revision 0, "St. Lucie, Unit 2 Fuel Transition Chapter 15 Non-LOCA [Loss of Coolant Accident] Summary Report," provides a description regarding "Spectrum of Control Element Assembly [CEA] Ejection Accidents (UFSAR [Updated Final Safety Analysis Report] 15.4.8)." The nonproprietary version of this document is publicly available under ADAMS Accession No. ML 15002A092.

The CEA ejection accident is analyzed using the methodology described in Topical Report XN-NF-78-44 (NP)(A) for total deposited enthalpy in the fuel. Topical Report XN-NF-78-44 (NP)(A) indicates that "the ejected rod analysis presented here will be applicable to all future ENC reloads for PWR type reactors." However, as indicated in the acceptance letter of April 8, 2015, to the licensee, the application of the above methodology to AREVA's CE 16x16 fuel is considered an expansion of the applicability of this topical report.

The Phenomenon Identification and Ranking Table in Section 3 of NUREG/CR-6742, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," identifies certain fuel physical, mechanical, and thermal properties of the fuel design under high burnup conditions during the PWR Rod Ejection Accident. These parameters are fuel thermal conductivity, gap conductance, clad conductivity (including oxide layer), transient cladding-to-coolant heat transfer coefficient, heat capacities of fuel and cladding, and other applicable parameters. It is not evident from your submittal on the control rod ejection analysis whether or not a more recent fuel performance code is used to compute the fuel and clad properties.

Upon review of the control rod ejection accident analysis submitted by the licensee in ANP-3347P, Section 4.25, the staff has determined that the margin to the centerline temperature for 20 percent and 65 percent rated thermal power has been substantially reduced for beginning of cycle.

RAI

Based on the above, perform a plant-specific analysis of control rod ejection accident for St. Lucie, Unit 2 using an alternate methodology as was proposed to the staff during the pre-submittal meeting on November 3, 2014. By performing the alternate analysis using the combination of latest fuel performance methodology and the proposed supplement to the ARCADIA reactor analysis system methodology, provide additional information that is necessary for determining the acceptability of the applicability of XN-NF-78-44(NP)(A) to AREVA CE 16x16 fuel design.

2.2 *Discussion of XN-NF-78-44(NP)(A) Applicability to AREVA CE16x16 Fuel*

2.2.1 Comparison of XN-NF-78-44(NP)(A) Results

In response to this RAI (Reference {1}), additional information regarding the applicability of the XN-NF-78-44(NP)(A) method to CE 16x16 fuel is provided in this document, following our discussion during the June 11, 2015 meeting between the NRC, Florida Power and Light (FPL), and AREVA on the draft of this RAI. In the June 11 meeting, it was discussed that the ARCADIA® method is still under development, and has not been submitted to NRC for review. It was discussed in that meeting that an alternative method of demonstrating the applicability of XN-NF-78-44(NP)(A) to CE 16x16 fuel (consistent with the basis for the original approval of the Topical Report) would be a comparison to the results of a different NRC-approved methodology.

The basis for the applicability of the XN-NF-78-44(NP)(A) method to CE 16x16 fuel was described in Reference {4}. Additional information is provided in this RAI response to verify the applicability of the original XN-NF-78-44(NP)(A) methodology (Reference {5}) to the AREVA CE16x16 fuel design. The additional information is based on a comparison to the results from a separate NRC approved methodology. FPL provided the necessary input to support a comparison between the AREVA control rod ejection (CRE) deposited enthalpy results and the CRE deposited enthalpy results determined by Westinghouse using a different NRC-approved CRE methodology. The comparison case chosen was the St. Lucie Unit 2 EPU LAR analysis performed by Westinghouse. The key analysis input parameters required for the XN-NF-78-44(NP)(A) method are the same as those described in the current FSAR analysis for the Westinghouse EPU case (Reference {7}, Section 15.4).

The table below provides the results of this comparison.

Table 2.1 Comparison of AREVA and Westinghouse Deposited Enthalpy Results

Case	AREVA Total Fuel Enthalpy	Westinghouse Maximum Fuel Pellet Enthalpy (Reference {7}, Table 15.4.8-1)
BOC HZP	40.5 cal/g	78.2 cal/g
BOC HFP	**181.1 cal/g**	**151.4 cal/g**
EOC HZP	52.8 cal/g	88.7 cal/g
EOC HFP	173.3 cal/g	141.3 cal/g

This comparison shows that both NRC approved methods identify the same case (BOC HFP) to be the limiting condition, and that the AREVA method produces a conservative result relative to the Westinghouse method for that case. The result of this comparison to a separate NRC approved method demonstrates that the AREVA control rod ejection methodology (Reference {5}) can be conservatively applied to the AREVA CE 16x16 fuel assembly design.

2.2.2 Comparison of the St. Lucie Unit 2 Results to XN-NF-78-44(NP)(A) Experience

As discussed in Reference {4}, Section 2.4, the XN-NF-78-44(NP)(A) deposited enthalpy results are based on four main parameters. These parameters are the ejected rod worth, the post-ejection peaking factor, the Doppler temperature coefficient, and the delayed neutron fraction.

The input values determined for the St. Lucie Unit 2 deposited enthalpy calculation are shown below and are compared to the range of values used in recent AREVA applications. The St. Lucie Unit 2 values reported in Reference {3}, Section 4.25 include additional biasing for application to the safety analysis, thus they may not compare directly with those used in the deposited enthalpy calculation. The range of recent AREVA applications includes CE-14, CE-15, Westinghouse 15x15 and Westinghouse 17x17 designs. It can be seen that the parameters for St. Lucie Unit 2 fall within the range of previous applications, and that the resulting deposited enthalpy calculated for St. Lucie Unit 2 is at the low end of the range of AREVA's recent operating experience.

Table 2.2 Comparison of Control Rod Ejection Parameters

[ ]

Figures 4.3 and 4.4 of Reference {5} provide the rod worth ranges based on the F_Q value being analyzed. Due to this change in ranges, the values are not explicitly noted here. [ ]

Figures 4.5 and 4.6 of Reference {5} provide the effective delayed neutron fraction range based on the case type that will be analyzed. [ ]

The application of XN-NF-78-44(NP)(A) methodology is thus justified for CE 16x16 fuel when the parameters used in the methodology are generated by explicit modeling of the CE 16x16 fuel design and plant configuration using the NRC-approved neutronics methodology.

[ ]

2.3 *Discussion of EMF-2310(P)(A) Revision 1 in Support of RAI Background Information*

The XN-NF-78-44(NP)(A) method applies only to the deposited enthalpy portion of the control rod ejection analysis. The fuel parameter and centerline temperature items identified in the "Background" portion of the RAI are addressed by the EMF-2310(P)(A) topical report (Reference {6}), which is the basis for the CEA Ejection transient response to evaluate the challenge to the DNB and fuel centerline melt fuel failure acceptance criteria using the S-RELAP5 and XCOBRA-IIIC codes. S-RELAP5 determines the core average transient response, the hot spot fuel centerline temperature and the thermal-hydraulic boundary conditions for subsequent XCOBRA-IIIC MDNBR calculations. The S-RELAP5 model explicitly accounts for the St. Lucie Unit 2 plant and fuel dimensions.

To account for the effects of TCD, COPERNIC (Reference {8}) is used to generate the fuel thermal conductivity, heat capacity and fuel pellet-to-clad gap coefficient inputs for the average core and hot spot models.

In the S-RELAP5 calculation, key neutronic parameters are conservatively biased to ensure that bounding results are calculated; specifically, Doppler reactivity feedback is biased less negative, moderator reactivity is biased more positive, ejected CEA worth is maximized to mitigate the event on Doppler feedback prior to reactor scram, power peaking factors are maximized and fuel rod exposures are maximized. These parameters are reviewed each reload cycle to verify that the acceptance criteria continue to be met for this event.

In addition to biasing key neutronics parameters, the S-RELAP5 calculation contains conservative design basis assumptions that bound the challenge to the fuel failure criteria: conservative reactor protection system setpoints and delays are modeled, minimum Technical Specification RCS flow and maximum core inlet temperature are used to increase the challenge to MDNBR, [

]. Also, fuel thermal properties for the average core and hot spot are conservatively biased to maximize core power and hot spot fuel centerline temperature.

The CEA Ejection is a postulated accident that allows limited DNB and fuel centerline melt fuel failures. The radiological dose analyses for St. Lucie Unit 2 support up to 9.5% fuel failures due to DNB and 0.5% fuel failures from fuel centerline melt. Since the AREVA analyses for the fuel transition do not result in any fuel failure for the calculations submitted as part of the license amendment, significant margin is maintained to the fuel failure limits based on DNB and fuel centerline melt.

3.0 REFERENCES

{1} U.S. Nuclear Regulatory Commission, ADAMS Accession No. ML15166A368, St. Lucie Plant, Unit 2 – Request for Additional Information Regarding License Amendment Request and Exemption Request Regarding the Transitioning to AREVA Fuel (TAC NOS. MF5494 and MF5495), June 2015.

{2} ANP-3352P, Revision 0, St. Lucie Unit 2 Fuel Transition License Amendment Request, December 2014.

{3} ANP-3347P, Revision 0, St. Lucie Unit 2 Fuel Transition Chapter 15 Non-LOCA Summary Report, December 2014.

{4} ANP-3396P, Revision 0, St. Lucie Unit 2 Fuel Transition Supplemental Information to Support the LAR, March 2015.

{5} XN-NF-78-44(NP)(A), A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors, October 1983.

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

{7} St. Lucie Unit 2 Final Safety Analysis Report, Amendment No. 21.

{8} BAW-10231P-A Revision 1, COPERNIC Fuel Rod Design Computer Code, Framatome ANP, Inc., January 2004