Non-proprietary AREVA report ANP-3396NP, Revision 0,
St. Lucie Unit 2 Fuel Transition Supplemental Information to Support the LAR

AREVA

St. Lucie Unit 2 Fuel Transition Supplemental Information to Support the LAR

ANP-3396NP
Revision 0

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AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	This is a new document.

Contents

Tables

Nomenclature

BOL .... Beginning of Life
BOC .... Beginning of Cycle
B&W .... Babcock & Wilcox

CE .... Combustion Engineering
CEA .... Control Element Assembly
CHF .... Critical Heat Flux

DNB .... Departure from Nucleate Boiling
DTC .... Doppler Temperature Coefficient

EOC .... End of Cycle
EOL .... End of Life

FCM .... Fuel Centerline Melt
F_Q .... Total Power Peaking Factor

GT .... Guide Tube

HFP .... Hot Full Power
HMP™ .... High Mechanical Performance
HTP™ .... High Thermal Performance
HZP .... Hot Zero Power

ID .... Inner Diameter
ISG .... Intermediate Spacer Grid

LAR .... License Amendment Request
LOCA .... Loss-of-Coolant Accident
LTP .... Lower Tie Plate

MDNBR .... Minimum Departure from Nucleate Boiling Ratio

NRC .... Nuclear Regulatory Commission

OD .... Outer Diameter
OBE .... Operating Basis Earthquake

PCMI .... Pellet Cladding Mechanical Interaction
PWR .... Pressurized Water Reactor

SER .... Safety Evaluation Report
SRP .... Standard Review Plan
SSE .... Safe Shutdown Earthquake

USNRC .... United States Nuclear Regulatory Commission
UTP .... Upper Tie Plate

1.0 Introduction

Through review of the St. Lucie Unit 2 license amendment request (LAR) and recent submittals (References 8 and 3), the Nuclear Regulatory Commission (NRC) staff has requested justification related to AREVA methodologies used for the control element assembly (CEA) ejection accident (Reference 2) and seismic analyses (Reference 9). The use of these methodologies was discussed with the NRC in a meeting on March 19, 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 justifications

Justification for use of AREVA's methodologies for the cited analyses is identified in Section 2.0 and Section 3.0.

2.0 USNRC Request for Supplemental Information Item #1

2.1 *Supplemental Request*

From Reference 1:

> "Justification for the applicability of the legacy control rod ejection methodology, XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient Pressurized Water Reactors," October 1983, to the Combustion Engineering (CE) 16-by-16 fuel."

2.2 *Acceptance Criteria*

The St. Lucie Unit 2 fuel-related acceptance criteria for control element assembly (CEA) ejection are as follows (Reference 3, Section 4.25.3):

1. Fuel failures due to departure from nucleate boiling (DNB) and fuel centerline melt (FCM) should be limited, so as not to impair the capability to cool the core. Additionally, the fuel failures should be within the limits of fuel failures used in the radiological analysis.

2. Reactivity excursions should not result in a peak radial average fuel enthalpy greater than the following limits:

   a. 230 cal/gm for fuel coolability.

      XN-NF-78-44(NP)(A) lists a deposited enthalpy limit of 280 cal/g. Based on more recent information, this limit is conservatively set to 230 cal/g in the St. Lucie Unit 2 evaluation, consistent with the Standard Review Plan (SRP) requirement.

   b. 150 cal/gm for fuel failure for hot zero power (HZP) conditions.

3. The pellet / cladding mechanical interaction (PCMI) failure criterion is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Reference 4 (Appendix B, Figure B-1).

2.3 *Method of Analysis*

The XN-NF-78-44(NP)(A) methodology (Reference 2) applies to Acceptance Criteria #2 and #3, related to the deposited enthalpy.

Acceptance Criterion #1 was evaluated using the EMF-2310(P)(A) Revision 1 methodology (Reference 5). The CEA ejection analyses were performed using the S-RELAP5 code to calculate neutron power, fuel thermal response, surface heat transfer, coolant flow rates, temperatures, and pressures, and to estimate the time of minimum departure from nucleate boiling ratio (MDNBR). Then the XCOBRA-IIIC code was used to calculate the MDNBR, using the HTP™ critical heat flux (CHF) correlation.

2.4 *Discussion of Applicability to CE 16x16 HTP™ Fuel and to St. Lucie Unit 2*

The applicability of this method is stated in Section 1.0 in the XN-NF-78-44(NP)(A) Topical Report as follows:

> "The ejected control rod accident can be parameterized by the following variables: 1) reactivity worth of ejected control rod, 2) power peaking factor, 3) reactivity coefficients and 4) delayed neutron fraction, βeff. With these variables defined, the core size, bank worth, etc., are not significant. Therefore, the ejected rod analysis presented here will be applicable to all future ENC (AREVA) reloads for PWR type reactors."

The calculation of the parameters that the control rod ejection methodology uses (defined above) is performed with an USNRC-approved neutronics methodology. These calculations explicitly model the reactor and fuel assembly geometry consistent with the St. Lucie Unit 2 configuration.

The reason that the methodology for energy deposition is independent of fuel pin dimensions is that the Doppler (fuel heatup) terminates the prompt critical excursion. This methodology conservatively assumes adiabatic heatup of the pin, so that the conduction losses from different sized pins are ignored. This behavior reflects the simple analytical expression (referred to as the Nordheim-Fuchs model) from Reference 6 (Page 166) that relates the core rod ejection accident parameters to the energy released for an adiabatic heatup:

$$ER = \frac{2 * (\rho - \beta) * C_p}{\alpha}$$

Where:

ER=Energy Released

β = the effective delayed neutron fraction for the core

ρ = ejected rod worth

Cp = heat capacity

α = the Doppler Temperature Coefficient

The total core energy release for a particular plant is only a function of ejected rod worth (ρ), effective delayed neutronics fraction (β), and Doppler Temperature Coefficient(α). XN-NF-78-44(NP)(A) methodology uses detailed calculations for the deposited energy using a similar approach that depends on these same parameters in conjunction with local power peaking to obtain the local energy deposited.

The parameters listed above are calculated specific to the fuel and plant type being analyzed. The range of values determined for the St. Lucie Unit 2 deposited enthalpy calculation are shown below compared to the range of values used in recent applications (values reported in the Safety Analyses {Reference 3, Section 4.25} include additional biasing). Recent applications include 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 2.1 Comparison of Control Rod Ejection Parameters

Figures 4.3 and 4.4 of Reference 2 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 2 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 USNRC-approved neutronics methodology.

3.0 USNRC Request for Supplemental Information Item #2

3.1 *Supplemental Request*

From Reference 1:

> "Justification for the applicability of BAW-10133P-A, Revision 1, and Addenda 1 and 2, "Mark-C Fuel Assembly LOCA (Loss-of-Coolant Accident) - Seismic Analysis," October 2000, to the CE 16-by-16 fuel."

3.2 *Applicability of the Method to CE 16x16 HTP™ Fuel and to St. Lucie Unit 2*

In support of the St. Lucie Unit 2 transition from Westinghouse fuel to AREVA fuel, BAW-10133(P)(A) Revision 1 and Addenda 1 and 2 (Reference 9) has been used as the methodology to evaluate the structural response of the fuel assembly to externally applied forces. This evaluation starts with bundle testing to establish the bundle response characteristics such as the fundamental frequencies, bundle stiffness, mechanical damping, etc., as described below.

Finite element models of the specific fuel types (St. Lucie Unit 2 BOL and EOL bundles) are benchmarked to match the dynamic characteristics measured from the tests. Core and baffle plate motions (from the St. Lucie Unit 2 plant-specific seismic analyses) are then superimposed on these benchmarked finite element models to determine the deflections and impact loads experienced by the fuel during seismic or LOCA events. The BAW-10133(P)(A) topical report describes this generic process. For St. Lucie Unit 2, the events were analyzed for a full core of the current fuel design, a full core of the AREVA CE 16x16 HTP™ fuel, and for a wide range of mixed core configurations, in order to verify that the limiting loads and deflections remain within acceptable fuel design limits.

The BAW-10133(P)(A) topical report, although referencing the Mark C fuel design in the title, describes the general methodology, which is structured to be generically applicable to PWR fuel designs. This generic applicability is captured in the Safety Evaluation Report (SER) for BAW-10133(P)(A), Rev. 1 in which it is noted that the methodology is acceptable for the "Mark C fuel design and similar designs". This methodology was modified in Addendum 1 where it is stated that "the application of this method is for generic use." Addendum 2 introduces the damping values to apply in this analysis and is noted to be "justified for all FCF [AREVA] PWR fuel designs based on the supporting test data." Damping is known to be a significant modifier of the fuel assembly response under seismic and LOCA loadings and it should be noted that the USNRC issued the SER for this addendum with recognition of the significant conservative margins that are retained in the approved values. While the sample problem for BAW-10133(P)(A), Rev. 1 was executed using the Mark C fuel design for Babcock & Wilcox reactors, Addendum 1 & 2 included a new sample problem to demonstrate application to the 17x17 fuel design for Westinghouse reactors. This new sample problem demonstrates the generic applicability of the method. It should be emphasized that the CE 16x16 HTP™ design for St. Lucie Unit 2 uses the same basic grid design, guide tube design, connections, and materials that are used in other AREVA CE 14x14 and 16x16 fuel designs. The rod array and specific dimensions are accounted for in the bundle and component testing that were performed with the St. Lucie Unit 2 design. Key characterization data for the CE 16x16 HTP™ design, along with other designs where BAW-10133(P)(A) has been applied, is presented in Table 3.1.

Table 3.1 Nominal, BOL Mechanical Design Data Comparison

	CE-16 HTP™	CE-14 HTP™	Mk-C	Advanced Mk-BW	Mk-B HTP™
Nominal Dimensional Characteristics					
Array	16x16	14x14	17x17	17x17	15x15
Square Envelope (in)	8.105	8.105	8.536	8.425	8.536
Assy-to-Assy gap, hot (in)	0.075	0.075	0.051	0.049	0.051
Assy-to-Baffle gap, hot (in)	0.1175	0.1175	0.095	0.1245	0.108
Longest Row length	17	17	17	15	15
Fuel Assy Length (in)	[ ]	[ ]	[ ]	[ ]	[ ]
Dry Fuel Assy Wgt (lbf)	[ ]	[ ]	[ ]	[ ]	[ ]
Fuel Rod Count	236	176	264	264	208
Fuel Cladding OD (in)	0.382	0.440	0.379	0.374	0.43
Fuel Cladding ID (in)	0.332	0.382	0.324	0.329	0.38
Grid Span Effective (in)	15.81	18.859	21.09	21.04	21.60
Intermediate Grids	8, HTP™	7, HTP™	6, 6-point contact	6, 6-point contact	6, HTP™
End Grid Type	Top, HTP™; Bottom, HMP™	Top, HTP™; Bottom, HMP™	2, 6-point contact	2, 6-point contact	Top, HTP™; Bottom, HMP™
Intermediate Flow Mixer	N/A	N/A	N/A	M5	N/A
ISG / GT Connection	Spot weld	Spot weld	Floating	Floating	Spot weld
Guide Tube Count	4 GT	4 GT	24	24	16
Guide Tube OD (in)	[ ]	[ ]	[ ]	[ ]	[ ]
Guide Tube ID (in)	[ ]	[ ]	[ ]	[ ]	[ ]
Dynamic Characteristics					
1st Mode Frequency in Air, room temperature at [ ] Deflection (Hz)	[ ]	[ ]	[ ]	[ ]	[ ]
BOL Bundle stiffness (lbf/in) (range)	[ ]	[ ]	[ ]	[ ]	[ ]
BOL, Hot Grid Stiffness (lbf/in)	[ ]	[ ]	[ ]	[ ]	[ ]

Consistent with the stated intent for this method to be generic, the described methods are transparent to the design being analyzed. BAW-10133(P)(A) defines a generic set of finite element model architecture and testing protocols that can represent the structure of any PWR fuel assembly. Design specificity is introduced in the characterization of the fuel through testing and the definition of model parameters that define assembly behavior, such as bundle stiffness, frequency, etc. Plant specificity is introduced in the geometry of the model boundary conditions (core dimensions, etc.) and inputs (e.g. core plate motion time histories). The application of BAW-10133(P)(A) topical report is thus justified for CE 16x16 fuel when fuel design specific bundle/component testing and model benchmarking is performed.

AREVA recognizes recent USNRC concerns related to the industry issue of plant seismic analyses. In the application of BAW-10133(P)(A) to the fuel at St. Lucie Unit 2, AREVA has gone beyond the scope of this topical to address emerging concerns. These additions/modifications to the methodology are clearly delineated in the License Amendment Request and are discussed below.

USNRC Information Notice 2012-09, "Irradiation Effects on Spacer Grid Crush Strength," identifies a concern about the impact of the change in behavior of the assembly and assembly components during the operational lifetime. Additional testing and evaluations are included in the analyses to address this information notice. A simulated EOL fuel assembly and simulated EOL spacer grids were tested and used to benchmark EOL-specific models for both lateral and vertical analyses. These models are then applied in the same manner as the standard BOL models to evaluate impact loads and fuel assembly deflections during seismic and LOCA events.

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3.3 *Bundle and Component Testing*

The following tests (Table 3.2) were performed in support of the LOCA Seismic Analysis using grids, end fittings, and bundles (BOL and EOL) that are prototypic of the St. Lucie Unit 2 CE 16x16 HTP™ design. The characterization of the St. Lucie Unit 2 fuel is in agreement with AREVA's expectations established from experience with the BAW-10133(P)(A) testing protocol.

Table 3.2 CE16x16 Assembly and Assembly Component Characterization Tests Performed

Component	Test Performed
BOL Assembly	Lateral stiffness test Free vibration test Baffle impact test Forced vibration test Axial stiffness test Vertical drop test Wet and dry assembly weights
EOL Assembly	Lateral stiffness test Free vibration test Baffle impact test Forced vibration test Axial stiffness test Vertical drop test Wet and dry assembly weights
HTP™ Grids	BOL/EOL Dynamic Crush Test BOL Slip Load BOL Spring Rate
HMP™ Bottom Grid	BOL Slip Load BOL Spring Rate
Upper Tie Plate	Load Test
Lower Tie Plate	Load Test (part of assembly vertical drop test)

4.0 References

1. U.S. Nuclear Regulatory Commission, ADAMS Accession Number ML15055A473, "ST. LUCIE PLANT, UNIT NO. 2- SUPPLEMENTAL INFORMATION NEEDED FOR ACCEPTANCE OF REQUESTED LICENSING ACTION REGARDING LICENSE AMENDMENT REQUEST AND EXEMPTION REQUEST REGARDING THE TRANSITIONING TO AREVA FUEL (TAC NOS. MF5494 AND MF5495)," March 10, 2015.

2. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors", Exxon Nuclear Company, September 1983.

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

4. U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Section 4.2, "Fuel System Design," Revision 3, March 2007.

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

6. David L. Hetrick, "Dynamics of Nuclear Reactors." American Nuclear Society, 1993.

7. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), "Request for Concurrence on SER Clarifications." NRC:99:030, July 28, 1999.

8. ANP-3352P, Revision 0, "St. Lucie Unit 2 Fuel Transition License Amendment Request – Technical Report", December 2014.

9. BAW-10133(P)(A), Revision 1 and Addenda 1 and 2, "Mark C Fuel Assembly LOCA-Seismic Analyses."

AREVA affidavit for withholding proprietary information from the public

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in ANP-3396P, Revision 0, entitled, "St. Lucie Unit 2 Fuel Transition Supplemental Information to Support the LAR," dated March 2015 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing ~~for nondisclosure and limited use of the information.~~

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 20th
day of March, 2015.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
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My Commission Expires Oct 31, 2018