L-2015-300
Attachment 6

ANP-3352NP
Revision 1

St. Lucie Unit 2 Fuel Transition License Amendment Request
Technical Report

Following 70 pages

AREVA

St. Lucie Unit 2 Fuel Transition License Amendment Request

ANP-3352NP
Revision 1

Technical Report

November 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	Section 1.1	Added last paragraph to describe purpose of this revision.
2	Sections 2.4.1, 2.4.3.1, and 2.4.4	Clarified that References 4 and 5 have subsequently been approved by the USNRC and that the relative statements are still applicable.
3	Table 2-2	Updated results for Items 3.3.1 (Guide Tube) and 3.4 (Structural Deformations) within this table.
4	Table 2-3	Mixed-core results were updated and full-core results were added to the table. Added note regarding the update of the grid strength allowable for hot, OBE conditions. Loads were updated (generally resulting in an increase in loads) to reflect [ ].
5	Section 5.2	Added statement at end of paragraph that additional SBLOCA results are found in a new reference.
6	Section 5.2.1	Changed licensing results in last paragraph and included "charging" in text related to ECCS.
7	Section 7.0	Updated references 4 and 28. Reference 4 has been approved by the USNRC. Reference 28 is the latest SBLOCA Summary Report. Added Reference 32.

Contents

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Nomenclature

AOO	Anticipated Operational Occurrence
ARO	All Rods Out
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
BOC	Beginning of Cycle
BOL	Beginning of Life
B&W	Babcock & Wilcox
CE	Combustion Engineering
CEA	Control Element Assembly
CEAD	Control Element Assembly Drop
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CLT	Centerline Temperature
COLR	Core Operating Limits Report
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EM	Evaluation Methodology
EOC	End of Cycle
EOL	End of Life
FCM	Fuel Centerline Melt
FPL	Florida Power and Light
F_Q	Total Power Peaking Factor
F_r	Assembly Radial Peaking Factor
GDC	General Design Criteria
HFP	Hot Full Power
HMP™	High Mechanical Performance
HPSI	High Pressure Safety Injection
HTP™	High Thermal Performance
HZP	Hot Zero Power
ID	Inner Diameter
IN	Information Notice
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate

Nomenclature *(continued)*

LOCA	Loss-of-Coolant Accident
LOCF	Loss of Coolant Flow
LPD	Local Power Density
LSSS	Limiting Safety System Setting
LTP	Lower Tie Plate
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NAF	Neutron Absorbing Fuel
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OD	Outer Diameter
PCT	Peak Cladding Temperature
PDIL	Power Dependent Insertion Limit
PORV	Power-Operated Relief Valve
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SER	Safety Evaluation Report
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
TCD	Thermal Conductivity Degradation
T-H	Thermal Hydraulic
TMSLL	Thermal Margin Safety Limit Lines
TM/LP	Thermal Margin/Low Pressure
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
UTP	Upper Tie Plate
VHPT	Variable High Power Trip
W	Westinghouse
WPR	Wetted Perimeter Ratio

1.0 Introduction and Design Overview

1.1 *Introduction*

Florida Power and Light (FPL) is planning to transition St. Lucie Unit 2 to AREVA CE-16 High Thermal Performance (HTP™[1]) fuel starting in the spring of 2017. The AREVA fuel design will be the CE-16 HTP™ fuel consisting of a 16x16 assembly configuration with M5®[1] fuel rods, Zircaloy-4 MONOBLOC™[1] corner guide tubes, an Alloy 718 High Mechanical Performance (HMP™[1]) spacer at the lowermost axial elevation, Zircaloy-4 HTP™ spacers in all other axial elevations, a FUELGUARD™[1] lower tie plate (LTP), and the AREVA reconstitutable upper tie plate (UTP).

The AREVA CE-16 HTP™ fuel design for St. Lucie Unit 2 is similar and has the same design features as the AREVA CE-14 HTP™ fuel design operating in St. Lucie Unit 1. It is also similar to the AREVA CE-16 HTP™ lead fuel assemblies operated in San Onofre Unit 2. The fuel rods are also similar to the AREVA CE-16 HTP™ fuel rods operated in the lead fuel assemblies at Palo Verde. The design features of the AREVA CE-16 HTP™ fuel design planned for St. Lucie Unit 2 have demonstrated excellent fuel performance. The HTP™ / HMP™ spacer grids are very resistant to flow induced grid-to-rod fretting failures, the FUELGUARD™ LTP is effective at protecting the fuel from debris in the reactor coolant system, and the M5® cladding has very low oxidation and hydrogen pickup rates.

Section 1.2 of this report provides a more detailed discussion of the design features of the AREVA CE-16 HTP™ fuel assembly. Section 2.0 of the report outlines AREVA's mechanical and structural evaluation methodology for the fuel design including the compatibility assessment and review of operating experience. Section 3.0 discusses the nuclear design bases and the methodologies for transitioning from the Westinghouse fuel design to the AREVA CE-16 HTP™ fuel for St. Lucie Unit 2. Section 4.0 provides the thermal and hydraulic design of the reactor that ensures the core can meet steady state and transient performance requirements without violating the acceptance criteria. Section 5.0 provides information related to the St. Lucie Unit 2 transient and accident analyses for the proposed transition. Also, summary reports of analyses for the non-loss-of-coolant accident (non-LOCA), small break LOCA (SBLOCA), and realistic

[1] HTP, HMP, MONOBLOC, and FUELGUARD are trademarks of AREVA. M5 is a registered trademark of AREVA.

large break LOCA (RLBLOCA) analysis methodologies have been prepared as documented in References 23, 28, and 29, respectively.

Note that demonstration of the evaluation methodologies has been performed with a submittal core design. The submittal core design was developed to provide key safety parameters to support the transition from Westinghouse fuel to AREVA CE-16 HTP™ fuel prior to the development of cycle-specific designs. This provides assurance that the plant licensing bases are met for the operation of St. Lucie Unit 2 with the AREVA CE-16 HTP™ fuel during the transition and full core cycles.

Revision 1 of this document is being issued to update seismic/LOCA results and SBLOCA results, as described in ANP-3440 (Reference 32). Table 2-2 (Items 3.3.1 and 3.4) and Table 2-3 (all items, including footnote) have been revised with the updated seismic/LOCA information (it was determined that the text within Sections 2.4.3.2 and 2.4.4.1 did not need to be revised; Items 3.2.5 and 3.2.7 within Table 2-2 were confirmed to remain bounding and therefore were not revised). Sections 5.2 and 5.2.1 have been revised with the updated SBLOCA information.

1.2 *Fuel Design Overview*

The AREVA fuel assembly for St. Lucie Unit 2 is of a Combustion Engineering (CE) 16x16 lattice design. This lattice contains 236 fuel rods, four (4) corner guide tubes, and one (1) center guide tube. The corner and center guide tubes each occupy four (4) fuel rod positions. The fuel rods are positioned within the fuel assembly by ten (10) spacer grids that are attached to the guide tubes.

The St. Lucie 2 AREVA design is very similar to the St. Lucie 1 AREVA fuel design. They both use HTP™ / HMP™ spacer grids, M5® fuel rod cladding, the FUELGUARD™ LTP, and the AREVA reconstitutable CE UTP. These components have been demonstrated to have excellent fuel performance and reliability. Figure 1-1 is a schematic of the AREVA fuel assembly.

The fuel rod design uses M5® cladding. The M5® material has very low corrosion and hydrogen pickup rates; providing substantial margin for end of life corrosion and hydrogen content. This material was developed in Europe and has been used extensively both in Europe and the United States for fuel rod cladding. The material has been generically reviewed and accepted by the United States Nuclear Regulatory Commission (USNRC) for use on CE fuel designs (Reference 1). Reloads with M5® cladding have been provided in the United States since 2000 and on CE-14 designs since 2006. Performance has been demonstrated to rod exposures in excess of 80 MWd/kgU. The fuel rod design includes uranium dioxide fuel rods and Gadolinia bearing uranium dioxide fuel rods, both with axial blankets of lower enriched uranium dioxide. Also, multiple uranium-235 enrichments are used within an assembly.

The lower tie plate design is the FUELGUARD™ structure. This structure uses curved vanes to provide non-line-of-sight flow paths for the incoming coolant to protect the fuel assembly from debris that may be present. This design is very efficient at preventing debris, including small pieces of wire, from reaching the fuel. The design uses the same vane configuration and spacing that has been used on CE-14, CE-15, CE-16, Westinghouse 14x14, Westinghouse 15x15, Westinghouse 17x17, and Babcock & Wilcox (B&W) 15x15 designs in the United States. This FUELGUARD™ design has been used on reloads in the United States since 1991 and on CE-14 designs since 2001. A schematic of the CE-16 FUELGUARD™ lower tie plate is provided in Figure 1-2.

The upper tie plate (UTP) design is the standard AREVA reconstitutable design for CE configurations. The basic configuration is the same as that used for CE-14 plants supplied by AREVA, with the heights, diameters, and position of the corner and center posts adjusted for the CE-16 lattice and to be compatible with the core plate separation at the St. Lucie Unit 2 plant. Figure 1-3 shows the St. Lucie 2 UTP configuration. The reaction plate has also been modified to match the interface conditions with the fuel handling grapples consistent with the co-resident fuel. This reconstitutable design uses the corner locking nuts to engage with the upper sleeves on the corner guide tubes. The design allows the reaction plate to be depressed to a setting well beyond the end of life deflections. At the fully depressed setting, the corner nuts can be rotated to disengage the upper tie plate from the guide tube locking sleeves; the upper tie plate can then be removed. This design does not create any loose or disposable parts during the reconstitution. The design has been used for AREVA CE-14 reloads in the United States since 1982. The reconstitution capabilities of the AREVA CE designs have been successfully demonstrated in CE-14 and CE-16 fuel examinations.

The cage or skeleton uses four (4) Zircaloy-4 MONOBLOC™ corner guide tubes, one (1) Zircaloy-4 center guide tube, nine (9) Zircaloy-4 HTP™ spacers, and one (1) Alloy 718 HMP™ spacer at the lowest spacer position. The HTP™ spacers are welded directly to the five guide tubes whereas the HMP™ spacer is attached to the guide tubes by mechanically capturing the spacer between rings that are welded to the guide tubes. Since the HMP™ spacer is made from Alloy 718, it cannot be directly welded to the Zirconium alloy guide tubes. The HTP™ spacer design was developed in the late 1980s and has been used on CE-14, CE-15, CE-16, Westinghouse 14x14, Westinghouse 15x15, Westinghouse 17x17, and B&W 15x15 fuel assemblies in the United States. (The CE-16 application was in two lead assembly programs.) The initial reloads were in 1991, and the initial CE-14 reloads were in 2001.

The CE-14 and CE-16 units have very challenging flow conditions on the peripheral assemblies and the peripheral fuel has been susceptible to flow induced grid-to-rod fretting failures. The AREVA HTP™ / HMP™ configuration has been successful in preventing these types of fuel failures on the core periphery. St. Lucie Unit 1 has operated with this design for eight (8) cycles without failures. The HTP™ design provides eight (8) line contacts as the interface between the fuel rod and the spacer grid. This line contact is very resistant to fuel rod failures from flow induced vibration fretting.

The HTP™ design is configured to improve heat transfer. As seen in Figure 1-4, the spring structure provides a flow path at an angle relative to the rod longitudinal direction, causing the water to swirl around the rod without creating a large pressure drop across the spacer. The HMP™ has the same line contact configuration but the channel is not angled. Since this spacer is at the lowermost position, the improved heat transfer is not necessary. As stated previously, the HMP™ material is Alloy 718. This material is very stable in irradiation environments, therefore providing additional assurance that the rod / spacer contact will be maintained throughout the design lifetime.

The assembly uses a MONOBLOC™ guide tube design for the corner guide tubes and a constant outer diameter and wall thickness design for the center guide tube. The MONOBLOC™ design maintains the same inner diameters in the dashpot and non-dashpot regions as the co-resident fuel, but has a constant outer diameter for the full length of the tube. Therefore, the wall thickness in the dashpot region (about the bottom 14 inches of the guide tube) is increased. The MONOBLOC™ guide tube design has been used for fuel reload batches in Europe and in the United States since 1998. The first application for CE plants was for a CE-14 design in 2010. St. Lucie Unit 1 has used the MONOBLOC™ guide tube design since 2013.

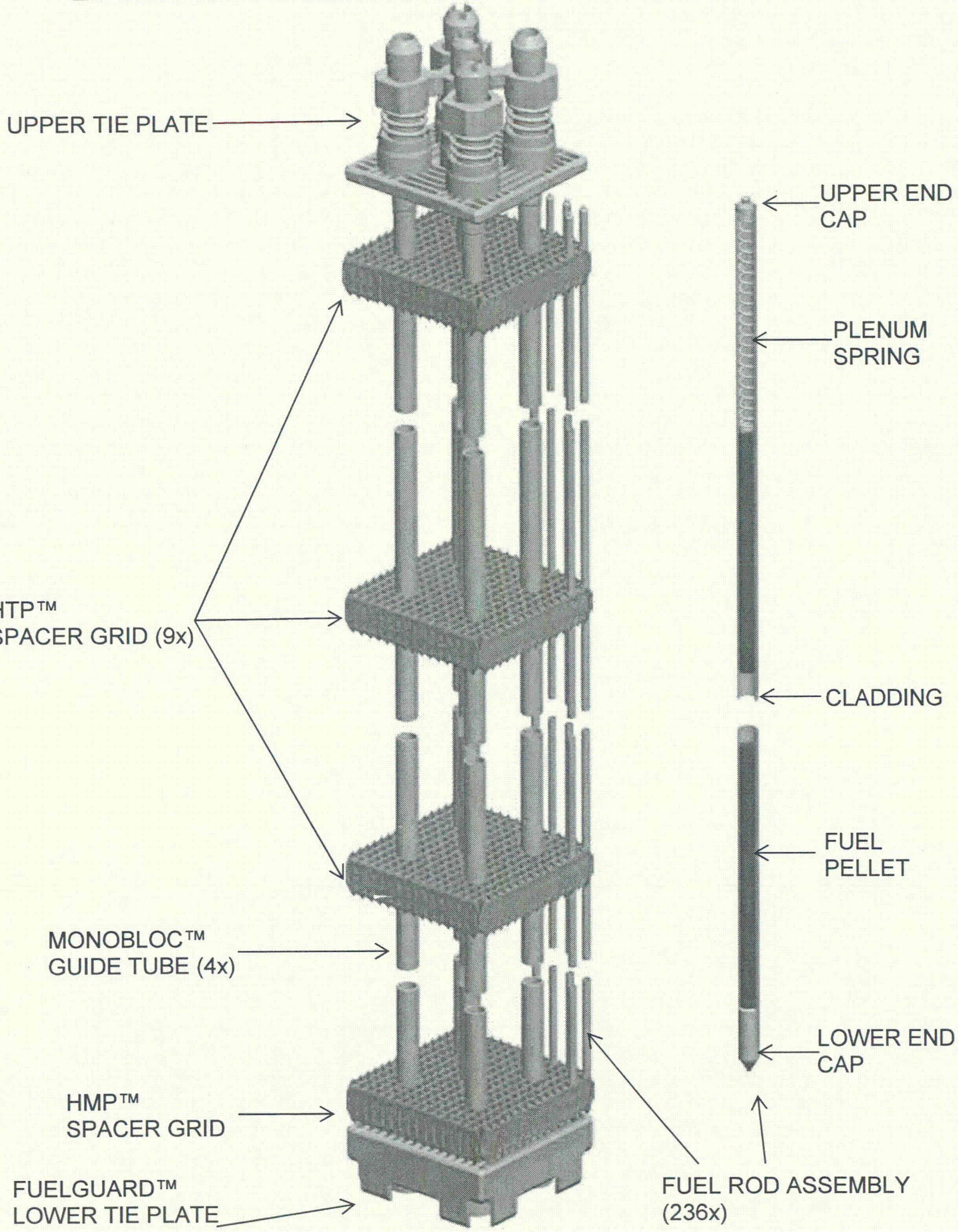

Figure 1-1: AREVA CE-16 Fuel Assembly for St. Lucie Unit 2

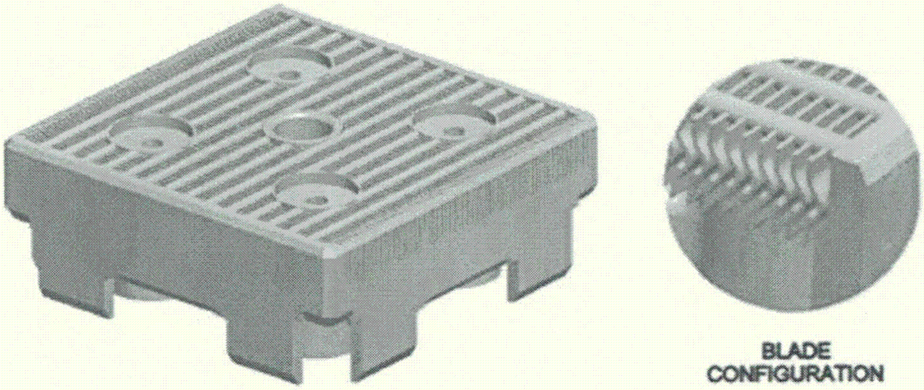

Figure 1-2: St. Lucie Unit 2 FUELGUARD™ Lower Tie Plate

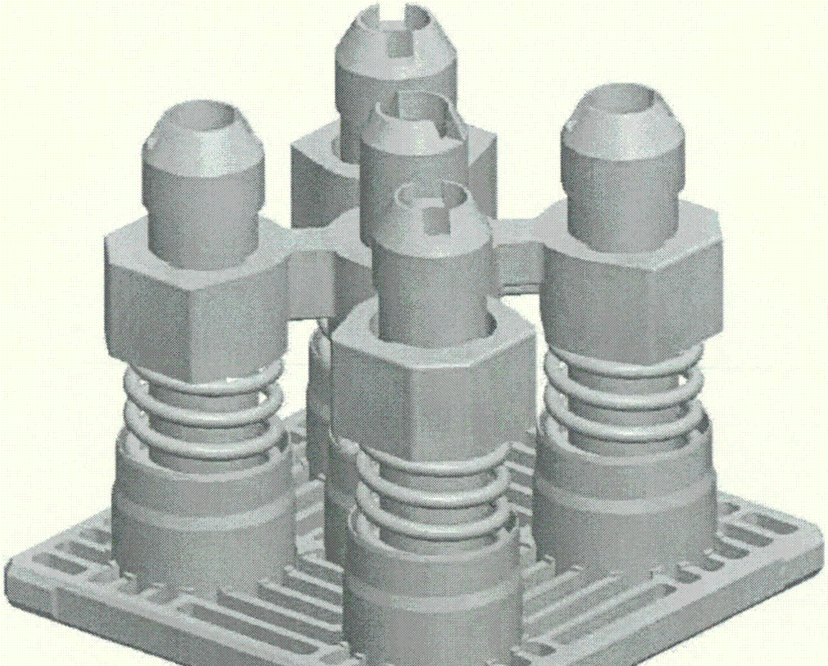

Figure 1-3: St. Lucie Unit 2 Upper Tie Plate

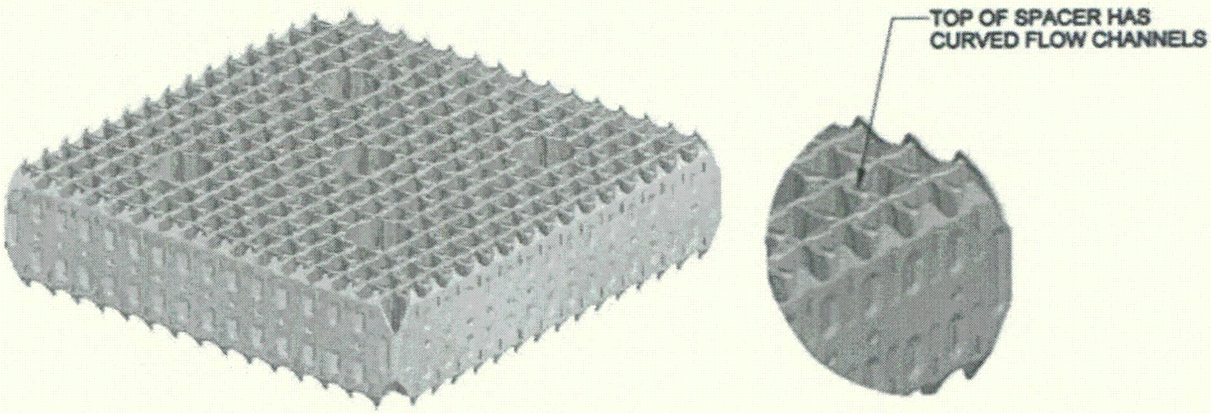

Figure 1-4: St. Lucie Unit 2 HTP™ Spacer Grid

2.0 Mechanical Design

2.1 *Introduction*

This section evaluates the mechanical design of the AREVA CE-16 HTP™ fuel design intended for batch implementation at St. Lucie Unit 2 and its compatibility with the co-resident fuel during the transition from mixed-fuel type core populations to cores with only AREVA CE-16 HTP™ fuel. AREVA has performed mechanical compatibility evaluations to assure acceptable fit-up with St. Lucie Unit 2 reactor core internals, fuel handling equipment, fuel storage racks, and co-resident fuel. A summary of the mechanical compatibility evaluations performed by AREVA is provided in Section 2.3.

The AREVA CE-16 HTP™ fuel assembly design for St. Lucie Unit 2 was analyzed in accordance with the USNRC-approved generic mechanical design criteria in EMF-92-116(P)(A) (Reference 2) in conjunction with USNRC-approved topical report BAW-10240(P)(A) (Reference 1). Reference 1 incorporates the M5® cladding material properties that were previously approved by the USNRC in BAW-10227(P)(A) (Reference 3) into the AREVA mechanical design methodology (Reference 2). All the mechanical design criteria were shown to be met up to the licensed fuel rod burnup limit of 62 MWd/kgU.

Section 2.2 provides an overview of operating experience gained by AREVA with the various CE-16 and CE-14 plants. The operating experience of the various components was also discussed in Section 1.2. Section 2.3 provides a description of the mechanical compatibility assessments. Section 2.4 describes the mechanical evaluations performed to show acceptability with the USNRC approved generic design criteria.

2.2 *Operational Experience of AREVA HTP™ Fuel Assemblies in CE-16 and CE-14 Plants*

The St. Lucie 2 AREVA fuel design is very similar to the AREVA CE-14 HTP™ fuel design and the AREVA CE-16 HTP™ fuel design used by other plants. AREVA provides the fuel for all of the CE-14 units in the United States (St. Lucie Unit 1, Millstone Unit 2, Calvert Cliffs Units 1 and 2, and Ft. Calhoun). All but Ft. Calhoun are sister units with similar fuel features. The current AREVA design for CE-14 fuel for these sister units uses Zircaloy-4 HTP™ spacer grids at every elevation except the bottom grid. The bottom grid is an Alloy 718 HMP™ grid. The guide tubes are either currently a Zircaloy-4 MONOBLOC™ design or in the process of transitioning to a

Zircaloy-4 MONOBLOC™ design. The fuel rods either currently use M5® cladding or are in the process of transitioning to M5® cladding. The LTPs are the FUELGUARD™ design, and the UTPs are the AREVA reconstitutable design. The initial HTP™ / HMP™ / FUELGUARD™ transition began at St. Lucie Unit 1 in 2001 and that fuel design has operated for eight (8) cycles without failures. Fuel failures did occur at Millstone Unit 2 but this design did not have the lower Alloy-718 HMP™ grid. Since replacing the bottom grid at Millstone Unit 2 with an HMP™ grid, there have been no failures. Calvert Cliffs began their transition to the AREVA CE-14 HTP™ design in 2010. The AREVA fuel has not failed at the Calvert Cliffs units through the transition.

AREVA has supplied lead assemblies of CE-16 HTP™ fuel to Palo Verde Unit 1 and San Onofre Unit 2 (SONGS2). The Palo Verde lead assemblies completed their lifetime irradiation and have been discharged and examined. The SONGS2 fuel operated for one cycle (at both in-board and core-periphery locations) before the plant was closed for steam generator issues. Both programs showed excellent fuel performance. The fuel rod at these units has the same radial dimensions and material as the St. Lucie Unit 2 fuel. However, the active fuel length in these lead assemblies was 150.0 inches instead of the 136.7 inches at St. Lucie. The cage structure is different at these two units, but the component features are similar to the standard CE-14 and St. Lucie Unit 2 AREVA designs. The lead assemblies had M5® cladding, MONOBLOC™ guide tubes (Palo Verde has a double expansion ID), HTP™ / HMP™ spacer grids, a FUELGUARD™ LTP (Palo Verde has the incore detectors entering from the bottom), and an AREVA reconstitutable UTP (both lead assembly UTPs are much taller than the St. Lucie 2 UTP). These lead assembly programs confirmed the excellent performance of the AREVA design.

2.3 *Mechanical Compatibility*

AREVA and Florida Power and Light (FPL) have performed an extensive review of the interfaces between the AREVA fuel assembly design and the plant equipment, the core interfaces, the control element assemblies (CEAs), the handling equipment, and the co-resident fuel. Where possible, the AREVA design maintained the same interface dimensions as the co-resident fuel. Also, where possible, the AREVA design maintained the same configurations and functionality as the AREVA designed CE-14 fuel in St. Lucie Unit 1. Table 2-1 shows a comparison of the major dimensions of the St. Lucie Unit 2 AREVA design, the St. Lucie Unit 2 co-resident design, and the St. Lucie Unit 1 AREVA design. Additionally, a prototypic UTP was fabricated and tested successfully for compatibility with plant handling equipment.

Table 2-1: Comparison of Nominal Mechanical Design Features

Feature	**St. Lucie 2 AREVA Design**	**St. Lucie 2 Westinghouse Design**	**St. Lucie 1 AREVA Design**
Fuel Assembly Overall Length, inch	158.529	158.529	157.115
Bundle Pitch, inch	8.18	8.18	8.18
Number of Bundles in Core	217	217	217
Core Power, MWth	3020	3020	3020
Fuel Rod Overall Length, inch	146.60	146.899	145.77
Fuel Rod Pitch, inch	0.506	0.506	0.580
Number of Fuel Rods / Assembly	236	236	176
Number of Corner Guide Tubes / Assembly	4	4	4
Number of Center Guide Tubes (Instrumentation Tubes) / Assembly	1	1	1
Fuel Rod Cladding Material	M5®	ZIRLO™[2]	M5® (starting in Cycle 26)
Fuel Rod Cladding Outer Diameter (OD), inch	0.382	0.382	0.440
Fuel Rod Cladding Thickness, inch	0.025	0.025	0.028
Fuel Pellet Diameter, inch	0.3255	0.3255	0.3770

[2] ZIRLO is a trademark of the Westinghouse Electric Company.

Table 2-1: Comparison of Nominal Mechanical Design Features *(continued)*

Feature	St. Lucie 2 AREVA Design	St. Lucie 2 Westinghouse Design	St. Lucie 1 AREVA Design
Fuel Stack Height (BOL, cold), inch	136.70	136.70	136.70
Axial Blanket Length (top / bottom), inch	6.00(UO_2 Rod) 10.50 (NAF Rod)	6.00(UO_2 Rod) 10.50 (NAF Rod)	6.00(UO_2 Rod) 11.40 (NAF Rod)
Corner Guide Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Center Guide Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Number of Grids	10	10	9
Bottom Grid	Alloy 718 HMP™	Inconel 625 (GUARDIAN™[3])	Alloy 718 HMP™
Upper Grids	Zircaloy-4 HTP™	Zircaloy-4 HID-1L (Mid Grids) Inconel 625 (Top Grid)	Zircaloy-4 HTP™

2.3.1 Fuel Assembly

The fuel assembly overall length was confirmed to be compatible with the dimensions of the core internals (spacing between core support plate and fuel alignment plate) at beginning of life cold and hot conditions. Additionally, positive engagement of the center/locking nuts and fuel alignment plate was demonstrated. An axial growth analysis confirmed adequate assembly to core internals and fuel rod / fuel assembly differential growth margins up to the licensed fuel rod and fuel assembly burnup limits.

The array type, the number of fuel rods and guide tubes, the fuel rod pitch dimensions, and the spacer grid centerline beginning of life elevations are the same as for the co-resident fuel.

These evaluations demonstrated that the AREVA design was compatible with the reactor components and co-resident fuel in the core. Additional evaluations of individual fuel assembly

[3] GUARDIAN is a trademark of the Westinghouse Electric Company.

components were also performed including the Upper Tie Plate, the Lower Tie Plate, and the Center and Corner Guide Tubes.

2.3.2 Upper Tie Plate

The mechanical compatibility of the UTP is explicitly evaluated because it:

- Interfaces with the holes in the fuel alignment plate in the reactor core
- Interfaces with all the fuel assembly grapples when moving the fuel assembly
- Interfaces with the control elements

The UTP evaluations show that the UTP is mechanically compatible. Additionally, FPL has performed compatibility validation testing with the plant equipment using a prototypic UTP.

2.3.3 Lower Tie Plate

The LTP also requires extensive compatibility evaluations because the LTP mates with the features (including alignment pins) of the lower core support plate. The AREVA LTP envelope is slightly smaller [ ] than that of the current St. Lucie Unit 2 fuel design, but is the same as the AREVA LTP used in St. Lucie Unit 1. All of the evaluations show that the LTP is compatible.

2.3.4 Guide Tubes

Besides being the structural components of the fuel assembly, the guide tubes interface with the control rods. The radial positions of the guide tubes within the assembly, the inner diameters of the guide tubes, and the weep hole diameters of the AREVA design are the same as the co-resident fuel. The axial locations of the guide tube dashpot and weep holes are also similar to the co-resident design. These critical dimensions assure that control element assembly drop times and guide tube cooling are not significantly affected by the introduction of the AREVA fuel assemblies. The only significant difference is that the AREVA design uses MONOBLOC™ corner guide tubes which have a constant outer diameter as discussed in Section 1.2.

2.4 *Mechanical Design Evaluations*

2.4.1 Description

The mechanical design evaluations are performed using the USNRC approved design methods and evaluated to the USNRC approved generic design criteria (Reference 2). Additional evaluations are included to address the impact of the thermal conductivity degradation with burnup and to address the impact of burnup on the seismic behavior of the fuel. The methods used for these additional evaluations are consistent with the methods previously reviewed by the USNRC for other applications and the updates to the generic criteria currently under USNRC review (References 4 and 5). (References 4 and 5 have recently been approved by the USNRC subsequent to Revision 0 of this document. The statements relevant to these references are still applicable.) These generic criteria are consistent with the specified acceptable fuel design limits (SAFDLs) identified in Chapter 4.2 of the Standard Review Plan (Reference 6). The USNRC-approved generic design criteria used to assess the performance of the fuel assemblies were developed to satisfy certain objectives (Reference 2). The use of M5® cladding required that the AREVA design methods be modified to incorporate the M5® properties and generic design criteria be evaluated to assure continued applicability. This implementation was documented in Reference 1 and generically reviewed and accepted by the USNRC.

The fuel analyses are broadly separated into fuel rod analyses and structural analyses. The fuel rod analyses include evaluations of the SAFDLs such as internal rod pressure, cladding creep collapse, cladding fatigue, corrosion, etc. These evaluations are very dependent on the rod power. For the transition cycles analyzed for this amendment request, the power histories were created using expected typical cycle core designs projected to the design life of the fuel. These cycle designs were created using the standard AREVA reload analysis codes and methods. The approved AREVA methodology requires these analyses to be redone for each cycle to assure that the actual cycle design will not result in SAFDL non-compliance. The actual reload cycle core designs will be performed by FPL using their standard, USNRC approved codes and methods. The LAR transition cycles are analyzed to demonstrate that the fuel design is acceptable and provide typical results showing SAFDL compliance. The specific reload results will be slightly different, but will continue to show SAFDL compliance.

2.4.2 Input Parameters and Assumptions

The input parameters used to perform the mechanical analyses included fuel design information derived from design documents, fuel assembly and component characteristics established by mechanical / hydraulic testing, plant parameters provided by FPL, fatigue duty cycles created using the fatigue transients provided in the UFSAR, and fuel rod power histories generated for the transition cycles by AREVA.

2.4.3 Recently Identified Analysis Issues

As described in Section 2.4.1, the USNRC generically approved methods and criteria were used to evaluate the St. Lucie 2 AREVA fuel design (References 1 and 2). The USNRC has issued two Information Notices (IN), IN-2009-23 and IN-2012-09 (References 7 and 8), which identify issues that are not addressed in the previous reviews of the generic methods. The first IN (Reference 7) identified the non-conservative impact of the thermal conductivity degradation of the fuel pellets with irradiation. The second IN (Reference 8) identified concerns about the change in the fuel assembly seismic response from irradiation. As discussed below, these issues have been addressed in the St. Lucie 2 mechanical evaluations.

2.4.3.1 Thermal Conductivity Degradation (TCD)

As identified in Reference 7, at high burnup conditions, the thermal conductivity of uranium dioxide fuel is reduced. This reduction results in higher pellet temperatures, and results in a reduction in margins to various SAFDLs. To account for TCD effects, AREVA has developed correction factors to be incorporated into evaluations using the currently approved RODEX2 code. These correction factors conservatively penalize the resulting margins for the affected SAFDLs to account for the thermal conductivity degradation. AREVA has submitted the correction factors generically to the USNRC in References 4 and 5. [

] (References 4 and 5 have recently been approved by the USNRC subsequent to Revision 0 of this document. The statements relevant to these references are still applicable.)

2.4.3.2 Seismic Evaluations

As part of the transition, AREVA performed lateral and vertical seismic evaluations. The fuel assembly lateral seismic and LOCA evaluations included the sensitivity studies to address the impact of AREVA and the co-resident fuel in different core locations for the different row lengths in the core. The USNRC-approved methodology, defined in BAW-10133(P)(A) and Addenda 1 and 2 (Reference 10), was used for the evaluations. As a result of recent USNRC concerns with seismic behavior and feedback from recent AREVA submittals for other units, there were additional evaluations and modifications to the AREVA seismic methods.

The basic methodology for the lateral seismic analysis uses full assembly test data to benchmark the bundle design with the finite element code CASAC. Component tests are performed to determine component characteristics such as stiffness and strength. The time / motion histories provided by the licensee are then imposed on this benchmarked model to determine the deflections of the fuel assemblies at the different core locations and the impact loads between the assemblies and between the assembly and the core shroud. The evaluations addressed the operating basis earthquake (OBE), the safe shutdown earthquake (SSE), and LOCA events. Each event was evaluated independently with lateral and vertical models.

USNRC Information Notice 2012-09, "Irradiation Effects on Spacer Grid Crush Strength," (Reference 8) identified the concern about the impact of the change in behavior of the assembly and assembly components during the operational lifetime. Additional testing and evaluations were 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 CASAC models for both lateral and vertical analyses. These models were 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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2.4.4 Mechanical Analyses Results

The generic criteria (SAFDLs) for the fuel rod and fuel assembly are listed in Table 2-2 along with the corresponding section number from the criteria topical report (Reference 2) and with the LAR transition cycle results. As noted in the specific items, some of the criteria specified below are addressed in analyses other than the mechanical design evaluations.

Table 2-2: Fuel Mechanical Design Evaluation Results

Criteria Section	Description	Criteria	Results
3.2	Fuel Rod Criteria		
3.2.1	Internal Hydriding	Hydrogen content in components controlled to a minimum level during manufacture to limit internal hydriding.	Controlled by manufacturing specifications and verified by Quality Control inspection.
3.2.2	Cladding Collapse	Sufficient plenum spring deflection and cold radial gap to prevent axial gap formation during densification.	Radial gap maintained throughout densification.
3.2.3	Overheating of Cladding	95/95 confidence that fuel rods do not experience DNB during steady state or AOOs.	Table 5-1 demonstrates acceptance criteria are met. Section 4.5.1.5 demonstrates this DNB performance is applicable to transition mixed core configurations. Section 4.5.5 demonstrates the TM/LP trip and DNB LCO barn are effectively set.
3.2.4	Overheating of Fuel Pellets	No centerline melting during normal operation and AOOs.	Table 5-1 demonstrates that acceptance criteria are met. Section 4.5.5 demonstrates the LPD LSSS and LPD LCO barns are effectively set.
3.2.5	Stress and Strain Limits		
	Pellet / Cladding Interaction	For M5® cladding, strain < 1% and no centerline melting.	Transient (AOO) strain: UO_2 rod = 0.498% NAF rod = 0.468% Steady-state strain: UO_2 rod = 0.346% NAF rod = 0.346% See overheating of pellets (above) for temperature.

Table 2-2: Fuel Mechanical Design Evaluation Results *(continued)*

Criteria Section	Description	Criteria	Results
	Cladding Stress	ASME Section III, Division 1, Article III-2000, in combination with the specified 0.2% offset yield strength and ultimate strength of the unirradiated cladding. M5® stress limit based on bi-axial burst strength of cladding and buckling criteria at limiting overpressure at BOL.	Component maintains margin to ASME criteria. Minimum margin is 28%.
3.2.6	Cladding Rupture	Not underestimated during LOCA and used in determination of 10 CFR 50.46 criteria.	Large break LOCA limiting case PCT results are lower than the temperature threshold for clad rupture. Clad rupture did occur for the small break LOCA limiting case. Clad rupture effects are incorporated in the LOCA licensing results.
3.2.7	Fuel Rod Mechanical Fracturing	ASME Section III, Division 1, Article III-2000, in combination with the specified 0.2% offset yield strength and ultimate strength of the unirradiated cladding. M5® stress limit based on bi-axial burst strength of cladding.	Criteria met with a minimum margin of 24%.
3.2.8	Fuel Densification and Swelling	Models included in USNRC approved fuel performance codes and taken into account in analyses contained in Sections 3.2.2, 3.2.4, 3.2.5, and 3.3.7 of this table.	Models included in USNRC-approved fuel performance codes. See Sections 3.2.2, 3.2.4, 3.2.5, and 3.3.7 of this table. Criteria met.
3.3	Fuel System Criteria		
3.3.1	Stress, strain, and loading limits on assembly components. (See 3.3.9 for handling and 3.4 for accident conditions.)		
	Guide Tube	SRP 4.2 Appendix A and ASME Section III, Subsection NG for Normal Operation and SSE and Appendix F for SSE+LOCA	Margins: Normal operation + OBE = 17% Normal + SSE = 8% Normal + SSE + LOCA = 3%

Table 2-2: Fuel Mechanical Design Evaluation Results *(continued)*

Criteria Section	Description	Criteria	Results
	Spacer Grid	Lateral load < load limit.	Normal operation bounded by handling criteria. Handling criteria met with a margin of 62%.
	Upper and Lower Tie Plates	Limiting loads occur during handling and postulated accidents.	Components maintain margin to ASME criteria and approved topical. Shipping and handling margins bounded by guide tubes with a margin of 34%.
3.3.2	Cladding Fatigue	Cumulative usage factor (CUF) [ ].	CUF Results: UO_2 rod = 0.635 NAF rod = 0.643 Criteria met.
3.3.3	Fretting wear	No fuel rod failures due to fretting wear.	Supported by fretting test, evaluation, and operational experience.
3.3.4	Oxidation, Hydriding, and Crud Buildup	Acceptable maximum oxide thickness. For M5® cladding, best estimate oxide < 100 microns. Effects of oxidation and crud included in thermal and mechanical fuel rod analyses. Stress analysis to include metal loss due to oxidation.	Maximum best estimate oxide of 24.8 microns. Approved fuel rod performance code accounts for oxidation and crud buildup. Metal loss accounted for in stress analysis. Criteria met.
3.3.5	Rod Bow	Lateral displacement of the fuel rods shall not be of sufficient magnitude to impact thermal margins.	Section 4.5.3 demonstrates that no rod bow penalty is required.
3.3.6	Axial Irradiation Growth		
	Fuel Rod	Clearance remains between fuel rod and UTP/LTP at EOL.	Criterion is met through design life.
	Fuel Assembly	The fuel assembly length shall not exceed the minimum space between upper and lower core plates in the cold condition at EOL.	Criterion is met through design life.

Table 2-2: Fuel Mechanical Design Evaluation Results *(continued)*

Criteria Section	Description	Criteria	Results
3.3.7	Rod Internal Pressure	Acceptable maximum internal rod pressure. [ ].	Maximum gas pressure: UO_2 rod = 1678.6 psia NAF rod = 1456.7 psia Maximum values remain below criterion limit. Internal pressure does not exceed system pressure.
3.3.8	Assembly Liftoff	No liftoff from core lower support.	Criterion is met for operation and 4th pump startup at 500 °F.
3.3.9	Fuel Assembly Handling	Assembly withstands 2 1/2 times the weight as a static force.	Components maintain margin to ASME criteria. Anti-hangup HTP™ spacer margin = 73%. The plenum spring meets the handling design criteria.
3.4	Fuel Coolability		
	Structural Deformations	Maintain coolable geometry and ability to insert control rods. SRP 4.2 Appendix A and ASME Section III, Appendix F, with lower Level A stress allowable for the guide tubes under SSE.	Verification of spacer and guide tube structural integrity under seismic-LOCA loading calculated based on AREVA + W mixed-core configurations BOL spacer grid design margin[4] = 29% (for OBE = 0.3%) EOL spacer grid design margin = 0.4% (for OBE = 37%) Guide tube margin = 8% (for SSE only) Guide tube margin = 3% (for SSE+LOCA)
3.4.1	Cladding Embrittlement	Include in LOCA analysis.	LOCA analysis peak clad temperature and maximum local cladding oxidation are well within licensing limits, demonstrating protection from cladding embrittlement.
3.4.2	Violent Expulsion of Fuel	< 230 cal/gm energy deposition < 150 cal/gm for HZP conditions.	Table 5-1 demonstrates acceptance criteria are met.

[4] BOL, OBE margin becomes 14% with revised allowable. See note in Table 2-3.

Table 2-2: Fuel Mechanical Design Evaluation Results *(continued)*

Criteria Section	Description	Criteria	Results
3.4.3	Fuel Ballooning	Consider impact of flow blockage in LOCA analysis.	The limiting small break LOCA transient experienced fuel swelling and rupture for the hot rod; results are well within licensing limits; fuel coolability is thus demonstrated.
4.1	Thermal and Hydraulic Criteria		
4.1.1	Hydraulic Compatibility	Hydraulic flow resistance similar to resident fuel assemblies.	Hydraulic compatibility acceptable. See Section 4.5.1.
4.1.2	Thermal Margin Performance	95/95 no DNB.	Section 4.5.5 and Table 5-1 demonstrates acceptance criteria are met
4.1.3	Fuel Centerline Temperature	No centerline melting.	Section 4.5.5 and Table 5-1 demonstrates acceptance criteria are met.
4.1.4	Rod Bow	Protect thermal limits.	Criterion is met. See Section 4.5.3
5.0	Neutronics Criteria		
5.1	Power Distribution	In accordance with Technical Specifications.	Criterion is met. See Section 3.0.
5.2	Kinetic Parameters		
	Doppler Reactivity Coefficient	Negative.	Criterion is met. See Section 3.0.
	Power Coefficient	Negative relative to HZP.	Criterion is met. See Section 3.0.
	Moderator Temperature Coefficient	In accordance with Technical Specification.	Criterion is met. See Section 3.0.
5.3	Control Rod Reactivity	Technical Specification's margin maintained.	Criterion is met. See Section 3.0.

The fuel rod analysis results presented above in Table 2-2 include consideration of the fuel Thermal Conductivity Degradation (TCD) issue. Relevant results have been penalized to include TCD corrections. These corrections are consistent with or more conservative than, the generic penalties developed and submitted for USNRC review in References 4 and 5. [

] (References 4 and 5 have recently been approved by the USNRC subsequent to Revision 0 of this document. The statements relevant to these references are still applicable.)

2.4.4.1 Additional Seismic Analysis Results

The AREVA fuel assembly design for St. Lucie Unit 2 has excellent seismic performance. The large corner guide tubes welded to the nine (9) HTP™ spacer grids creates a cage and bundle structure that has high assembly stiffness. The HTP™ spacer design is very sturdy while remaining flexible resulting in robust seismic performance. Therefore, it can more readily absorb the impacts without plastically deforming.

The St. Lucie 2 Seismic / LOCA evaluations included cases for all the different assembly rows in the core. The mixed core behavior was assessed by performing sensitivity analyses for the different rows with different positions for the AREVA and co-resident designs (including the all-AREVA and all-co-resident design cases). The limiting impact loads and margins for the AREVA assemblies occur in specific mixed core conditions in which the AREVA fuel is on the core periphery and adjacent to the co-resident fuel. These limiting cases are shown in Table 2-3. Based on the evaluations, the AREVA fuel assemblies meet design limits for both mixed core and full core conditions.

Table 2-3: Seismic and LOCA Loadings

OBE					
		Load	**Allowable**	**Margin**	**Row Layout**
Full Core AREVA	BOL	[ ]	[ ]*	86%	9 & 15 assembly row AAA...AAA
	EOL	[ ]	[ ]	77%	4 assembly row AAAA
Mixed Core	BOL	[ ]	[ ]*	0.3%	17 assembly row AWW...WWA
	EOL	[ ]	[ ]	37%	17 assembly row AWW...WWA
SSE + LOCA					
		Load	**Allowable**	**Margin**	**Row Layout**
Full Core AREVA	BOL	[ ]	[ ]	76%	4 assembly row AAAA
	EOL	[ ]	[ ]	43%	11 assembly row AAA...AAA
Mixed Core	BOL	[ ]	[ ]	29%	15 assembly row AWW...WWA
	EOL	[ ]	[ ]	0.4%	17 assembly row AWAW...WAWA
* This grid allowable has been updated to [ ] based on the inclusion of additional crush test data for St. Lucie 2 specific grid type. The margin for the BOL, OBE, Mixed Core limiting case increases to 14% and the margin for the BOL, OBE, Full Core case increases to 88%.					

2.5 *Mechanical Design Conclusions*

The AREVA CE-16 HTP™ fuel design is mechanically compatible with the co-resident fuel design, the plant structures, and fuel handling / interfacing equipment and structures at St. Lucie Unit 2. The AREVA CE-16 HTP™ fuel design has been analyzed in accordance with USNRC-approved mechanical design criteria using transition cycle inputs. Adaptations to the methodologies have been identified and explained to address USNRC Information Notices and to align with recently approved submittals. All of the design criteria were shown to be met up to the licensing fuel rod burnup of 62 MWd/kgU under normal and faulted operating conditions.

3.0 Nuclear Design

3.1 *Introduction*

The licensing basis for the reload core nuclear design is defined in UFSAR Section 4.3. The purpose of the core analysis is to verify that the cycle-specific reload design and the key safety parameters are properly addressed in the reload analysis. The effects of transitioning from Westinghouse CE 16x16 fuel to AREVA CE-16 HTP™ fuel on the nuclear design bases and methodologies for St. Lucie Unit 2 are evaluated in this section.

3.2 *Input Parameters*

The AREVA St. Lucie CE-16 HTP™ fuel differs from that of existing Westinghouse CE 16x16 fuel design, with the unique features as described in Sections 1.2 and 2.3. Refer to Section 4.5.1.5, for discussion of the application of a mixed core penalty to the departure from nucleate boiling (DNBR) safety limits. The power distribution effects are discussed in the specific analyses presented in Section 5.1.

3.3 *Methodology*

The nuclear design methodology and codes are updated to include the standard AREVA methodology and code package for the transition cycles and future operation of AREVA St. Lucie CE-16 HTP™ fuel. References 12, 14, and 15 are the USNRC-approved topical reports outlining the approved AREVA neutronics methodology and codes.

The safety evaluation report (SER) for Reference 12 requires that application of the methodology to a CE-16 fuel assembly design be supported by additional validation and that this validation be maintained by AREVA and available for USNRC audit. This SER requirement has been met for St. Lucie Unit 2.

Benchmarking of the AREVA neutronics methodology and codes was performed and demonstrated acceptable modeling of previous and current St. Lucie Unit 2 cores. [

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Key safety parameters are calculated as part of the core design neutronics analysis (see Table 3-1). These parameters are then biased in the safety analysis. Key safety parameters are then calculated for the cycle-specific reload and compared with the values used in the safety analysis. These cycle-specific parameters will be generated based on AREVA methodology using both AREVA codes and the current set of codes used by FPL. If the key parameters are not within the reference safety analysis, then the transient will be re-analyzed or re-evaluated on a cycle-to-cycle basis using the stated methods.

Table 3-1: Range of Key Safety Parameters

Technical Specification	Safety Parameter	Transition Analysis Value
TS 1.25	Nominal Reactor Core Power (MWt)	3020
TS 3.2.5 COLR Table 3.2-2	Vessel Average Coolant Inlet Temp HFP (°F)	551
Not a TS	Nominal Coolant System Pressure (psia)	2250
TS 3.1.1.4	Most Positive Moderator Temperature Coefficient (MTC) (pcm/°F)	≤ +5 (Power ≤ 70%) ≤ 0 (Power = 100%) Linear ramp from +5 at 70% to 0 at 100%
COLR Section 2.1	Most Negative MTC (pcm/°F)	-33
Not a TS	Doppler Temperature Coefficient (DTC) (pcm/°F) (See footnote[5])	-1.60 to -1.30
Not a TS	Beta-Effective (See footnote[5])	0.0052 to 0.0065
TS 3.2.3 COLR Section 2.5	Normal Operation HFP Unrodded F_r^T (without uncertainties)	≤ 1.65
COLR Section 2.8 and 2.9	Shutdown Margin (pcm)	≥ 3600 (≥ 200°F) ≥ 3000 (≤ 200°F)

[5] Beta-effective and DTC do not have analyses or TS limits directly associated with them. These parameters are major contributors to transient analysis behavior and are good early indicators of significant physics characteristics changes in the core. Current design values for these parameters are expected ranges only.

Table 3-1: Range of Key Safety Parameters *(continued)*

Technical Specification	Safety Parameter	Transition Analysis Value
TS 3.2.1 COLR Section 2.4	Linear Heat Rate (kW/ft)	≤ 13.0
TS 3.2.5 COLR Section 2.6	DNB LCO Axial Shape Index (100% Power)	> -0.08 < 0.15
Not a TS	Maximum Ejected Rod, F_Q (See footnote[6])	BOC HZP: 4.867 BOC HFP: 2.681 EOC HZP: 8.781 EOC HFP: 2.320
Not a TS	Total Deposited Enthalpy, (cal/gm) (See footnote[6])	BOC HZP: 24.9 BOC HFP: 144.1 EOC HZP: 26.9 EOC HFP: 136.9

3.4 *Description of Design Evaluations*

Standard nuclear design analytical models and methods (Reference 12) accurately describe the neutronic behavior of the AREVA St. Lucie CE-16 HTP™ fuel. The specific design bases and their relation to the GDCs in 10 CFR 50, Appendix A for the AREVA St. Lucie CE-16 HTP™ design are discussed in Reference 2.

The effect of extended burnup on nuclear design parameters has been previously approved in detail in Reference 13. That discussion is valid for the AREVA St. Lucie CE-16 HTP™ discharge burnup level.

A transition core design and two additional follow-on core designs have been developed for St. Lucie Unit 2 to model the transition to AREVA St. Lucie CE-16 HTP™ fuel. The loading patterns were developed based on design requirements (e.g. energy, peaking, and assembly placement) for St. Lucie Unit 2. The loading patterns were depleted at a core power of 3020 MWt. These cycles were not developed to be bounding of future cycle designs, but were developed to be representative of future cycle designs to demonstrate acceptable margins. The

[6] The control rod ejection analysis values do not have TS limits directly associated with them. The design values listed are expected based on the transition.

first transition cycle contains fresh AREVA St. Lucie CE-16 HTP™ fuel with once-burnt and twice-burnt Westinghouse CE 16x16 fuel. The second transition cycle contains fresh and once-burnt AREVA St. Lucie CE-16 HTP™ fuel with twice-burnt Westinghouse CE 16x16 fuel. The third transition cycle contains only AREVA St. Lucie CE-16 HTP™ fuel. These models show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. Table 3-2 contains key information based on the nominal transition cycle designs. Key safety parameters were verified for the core design in Table 3-1.

The standard methods of fresh fuel enrichment loading and integrated burnable poisons will be applied to control the peaking factors and maintain compliance with the Technical Specifications and COLR. Changes in boron concentration and axial offset are typical of normal cycle-to-cycle variations in the core design.

Table 3-2: Projected Transition Cycle Core Characteristics

Transition Cycle	Cycle Energy (EFPD)	Number of Feed AREVA Assemblies	Maximum HFP ARO F_r^T		Maximum HFP ARO F_Q	
			AREVA Fuel	Westinghouse Fuel	AREVA Fuel	Westinghouse Fuel
N	518.7	88	1.538	1.220	1.859	1.428
N+1	515.9	84	1.571	1.312	1.894	1.567
N+2	504.9	84	1.556	N/A	1.858	N/A

3.5 *Results*

Margin to key safety parameter limits (Table 3-1) is maintained during the transition from Westinghouse CE 16x16 fuel to AREVA St. Lucie CE-16 HTP™ fuel.

The changes in fuel design and discharge burnup result in only a small impact on the results of the reload transition core analysis relative to the current design. The variations in these parameters are typical of the normal cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle.

Changes to the core power distributions and peaking factors are the result of the normal cycle-to-cycle variations in core loading patterns. These will vary cycle-to-cycle based on actual energy requirements. The normal methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control peaking factors. Compliance with the peaking factor TS will be assured using these methods.

3.6 *Conclusion*

The nuclear core design analysis of the core design for the transition from Westinghouse CE 16x16 fuel to AREVA St. Lucie CE-16 HTP™ fuel has confirmed peaking factor and key safety parameters can be maintained within their specified limits using only AREVA methodologies and codes. The key safety parameters generated with the core design are used in the applicable analyses and evaluated to meet the acceptance criteria.

The key safety parameters and the peaking factor limits will be verified on a cycle specific basis. However, the values are planned to be created using FPL and AREVA methods.

4.0 Thermal and Hydraulic Design

4.1 *Description*

This section describes the Thermal Hydraulic (T-H) analysis supporting the transition to AREVA CE-16 HTP™ Fuel at St. Lucie Unit 2.

4.2 *Input Parameters and Assumptions*

XCOBRA-IIIC is the core T-H sub-channel analysis code that was used for the AREVA HTP™ fuel analysis. USNRC approval of the XCOBRA-IIIC code was issued in the SER attached to Reference 20.

For the Thermal Hydraulic analysis, fuel-related safety and design parameters of the AREVA CE-16 HTP™ fuel design have been used. These parameters have been used in safety and design analyses discussed in this section and in other relevant sections of this LAR.

Table 4-1 lists T-H parameters used for the fuel transition thermal-hydraulic analysis.

Table 4-1: Thermal-Hydraulic Design Parameters

Parameter	Value
Reactor core heat output, MWt	3020
Heat generated in fuel, %	97.5
Pressurizer/core pressure, psia	2250
Nominal vessel/core inlet temperature, °F	551
RCS minimum flow rate (including bypass), gpm	370,000
Core bypass flow, %	4.2
Core area, ft^2	54.39
Core inlet mass velocity (excluding bypass, based on TS minimum flow rate, 10^6 $lb_m/hr\text{-}ft^2$	2.45
Pressure drop across core, psi (full-core AREVA CE-16 HTP™)	[ ]
Core average heat flux, kW/ft	5.2

The limiting directions for biased parameters are shown in Table 4-2. Biases were applied to input parameters according to the approved methodology (Reference 21). For the transient analyses, uncertainties were deterministically applied. Thus, steady-state measurement and instrumentation errors were taken into account in an additive fashion to ensure a conservative analysis. For statistical departure from nucleate boiling (DNB) calculations, uncertainties were statistically treated according to the approved methodology (Reference 19). The system related uncertainties bounded by the non-loss of coolant accident (non-LOCA) safety analyses are listed in Table 4-3.

Table 4-2: Limiting Parameter Directions

Parameter	Limiting Direction for DNB
Reactor core heat output (MWt)	maximum
Heat generated in fuel (%)	maximum
Nominal vessel / core inlet temperature	maximum
Fr, enthalpy rise hot channel factor	maximum
Pressurizer/core pressure (psia)	minimum
RCS flow (See note 1 below) (gpm)	minimum
Note 1: The limiting (minimum) value of the RCS flow is the TS minimum flow.	

Table 4-3: System Related Uncertainties

Parameter	Uncertainty
Reactor Thermal Power	±0.3% (at 100% RTP)
RCS Flow	±12,500 gpm
RCS Pressure	±45.0 psi
Core Inlet Temperature	±3.0 °F

Control grade equipment was modeled in such a way that it does not mitigate the effects of an event. The reactor trip setpoints and time delays modeled in the transient analyses were conservatively applied to provide bounding simulations of the plant response. To the extent that the reactor protection system and engineered safety features system are credited in the accident analyses, the setpoints have been verified to adequately protect the plant for the fuel transition.

4.3 *Acceptance Criteria*

The reactor core is designed to meet the following limiting T-H criteria:

- There is at least a 95% probability at a 95% confidence level that DNB will not occur on the limiting fuel rods during Modes 1 and 2, operational transients, or any condition of moderate frequency.

- No fuel melting during any anticipated normal operating condition, operational transients, or any conditions of moderate frequency.

The ratio of the heat flux causing DNB at a particular core location, as predicted by a DNB correlation, to the actual heat flux at the same core location is the DNBR. Analytical assurance that DNB will not occur is provided by showing the calculated DNBR to be higher than the 95/95 limit DNBR for conditions of normal operation, operational transients and transient conditions of moderate frequency.

4.4 *Method of Analysis*

The T-H analysis of the AREVA CE-16 HTP™ fuel is based on the approved methodologies for performing DNB calculations (References 25 and 21). The S-RELAP5 code was used for the transient analysis. The XCOBRA-IIIC code was used to calculate minimum DNBR (MDNBR) using the HTP and Biasi critical heat flux (CHF) correlations. RODEX2-2A (References 9 and 22) was developed to perform calculations for a fuel rod under normal operating conditions.

For non-LOCA applications, RODEX2-2A was used to establish the fuel centerline melt linear heat rate (LHR) as a function of exposure. The HTP DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in DNB performance due to the flow mixing nozzles effects. USNRC acceptance of a 95/95 HTP correlation safety limit DNBR of 1.141 for HTP CHF Correlation is documented in Reference 18. The Biasi CHF correlation (Reference 26) is used to calculate the DNBR for post-scram reactor conditions. The 95/95 Biasi correlation safety limit DNBR used in analysis is [ ]. The ranges of parameters used in the AREVA CE-16 HTP™ design have been verified to fall within the range of applicability for these correlations.

[

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The approved methodology for performing DNB calculations using the XCOBRA-IIIC code in a mixed core is described in Reference 25. The SER for the Reference 20 topical report states that the use of XCOBRA-IIIC is limited to the "snapshot" mode. Thus, MDNBR calculations were performed using a steady-state XCOBRA-IIIC model with core boundary conditions at the time of MDNBR from the S-RELAP5 transient analyses.

The Reference 19 topical report describes the method for performing statistical DNB analyses. Two conditions were noted in the SER for the Reference 19 methodology:

- The methodology is approved only for Combustion Engineering (CE) type reactors which use protection systems as described in the Reference 19 topical report.
- The methodology includes a statistical treatment of specific variables in the analysis; therefore, if additional variables are treated statistically, Siemens Power Corporation, now AREVA, should re-evaluate the methodology and document the changes in the treatment of the variables. The documentation will be maintained by AREVA and will be available for USNRC audit.

Protection against the fuel centerline melting (FCM) SAFDL is expressed as a limit on LHR allowed in the core. The FCM limit was explicitly calculated for the AREVA fuel transition. Due to the reduced thermal conductivity of gadolinia fuel rods, the FCM limit may be set by gadolinia fuel. A FCM limit is established for UO2 fuel rods such that, FCM is precluded for all fuel rod types. A penalty to address thermal conductivity degradation (TCD) was applied where applicable.

The impact of rod bowing on the MDNBR and peak LHR was evaluated using the rod bow methodology described in Reference 27.

4.5 *Results*

4.5.1 Thermal-Hydraulic Compatibility

4.5.1.1 Core Pressure Drop

The Westinghouse fuel assemblies have a lower overall resistance to flow than the AREVA HTP™ fuel assemblies; therefore, as the core transitions from a full core of Westinghouse fuel to a full core of AREVA fuel, the core pressure drop increases. An analysis was performed to assess the change in core pressure drop associated with the fuel transition.

The core pressure drop for a full core of AREVA HTP™ fuel assemblies is [ ].

The total pressure drop associated with the full core of AREVA HTP™ fuel is [ ] than the total pressure drop of the Westinghouse core. The pressure drop profile between the two assembly types is shown in Figure 4-1.

Figure 4-1: Pressure Drop Profiles

4.5.1.2 Total Bypass Flow

The change in total bypass flow was examined to determine if the active heat transfer coolant flow will be adversely impacted by the fuel transition. The bypass flow includes the following flow paths: guide tubes, vessel upper head, inlet-to-exit nozzle, and core barrel/baffle. The change in total bypass flow was determined by examining the change due to non-guide tube paths and guide tube paths. Bypass flow for the non-guide tube paths is affected by changes in core pressure drop, while guide tube bypass flow is dependent on both core pressure drop and assembly geometry.

The core pressure drop for a full core of AREVA fuel is higher than the core pressure drop for a Westinghouse core. As a result, the driving force for bypass flow increases and the total bypass flow increases. [

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4.5.1.3 Crossflow Velocity

The Inter-Assembly Crossflow velocities affecting the AREVA HTP™ fuel assemblies were analyzed to assure satisfactory performance during the transition. Different core configurations were considered in the analysis, ranging between bounding configurations with a single AREVA assembly and a single Westinghouse assembly.

Although other geometries and operating conditions may result in different crossflow velocity profiles, the analyzed scenario provides representative crossflow velocities to cover core configurations associated with the fuel transition. The results are representative of anticipated operating conditions and are used to develop bounding inputs for mechanical analyses.

4.5.1.4 RCS Flow Rate

An analysis was performed to assess the change in primary system loop flow attributed to the fuel transition. The change in the Reactor Coolant System (RCS) loop flow will not impact the Technical Specification minimum loop flow rate.

[

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4.5.1.5 Transition Core DNB Performance

XCOBRA-IIIC was used to analyze the effect of the fuel transition on the DNB performance of the AREVA CE-16 HTP™ fuel assemblies. The power level was selected to achieve MDNBR close to the HTP CHF correlation limit. A mixed core penalty was applied to all core configurations, including the full core of AREVA HTP™ fuel.

The AREVA HTP™ fuel assembly is associated with more overall flow resistance than the co-resident Westinghouse fuel. This results in flow transferring from the AREVA HTP™ fuel into the Westinghouse fuel, which is detrimental to DNB performance of the AREVA fuel. [

] The impact will decrease for subsequent transition cycles.

4.5.1.6 Control Rod Drop Times

An assessment was performed to validate that the Technical Specification requirement for the control rod drop time is not challenged as a result of the fuel transition. The control rod drop time is primarily dependent on the number, size, and location of the guide tube weep holes, as well as the inner diameter and height of the guide tube dashpot region.

Due to the similarities between the Westinghouse and AREVA guide tube designs, the control rod drop times will not be significantly impacted by the fuel transition and will remain below the required drop time of 3.25 seconds.

4.5.2 Thermo-Hydrodynamic Instability

AREVA has evaluated the St. Lucie reactor for its susceptibility to a wide range of potential thermo-hydrodynamic instabilities. It concludes that St. Lucie Unit 2 will not experience thermo-hydrodynamic instabilities during normal operation and AOOs.

4.5.3 Rod Bow

The impact of rod bowing on the MDNBR and peak LHR was evaluated using the rod bow methodology described in Reference 27. The objective was to determine the threshold burnup level at which a rod bow penalty must be applied to either the MDNBR or peak LHR results. The results show that no rod bow penalty is required for DNB or LHR calculations.

4.5.4 Guide Tube Heating

Boiling of coolant within the guide tubes has the potential to increase corrosion rates and be detrimental for neutron moderation. An analysis was performed to demonstrate that boiling will not occur within the guide tubes of the AREVA fuel assemblies. For conservatism, severe operating conditions were used in the analysis.

Guide tube heating is most severe when a neutron absorbing material is inserted into the guide tube. The analysis considered a high powered assembly with the control rods at PDIL conditions. The analysis demonstrates that control rod linear heat generation rates less than or equal to 9.2 kW/ft will preclude boiling within the guide tube.

4.5.5 Setpoint Analyses

The setpoint analyses ensure there is sufficient margin for the Limiting Safety System Settings (LSSS) and Limiting Condition for Operation (LCO) systems that monitor various reactor system variables designed to protect the SAFDLs and other design limits. The results of the setpoint analyses are presented in Table 4-4.

Table 4-4: Minimum Margin Summary for Setpoint Calculations

Setpoint Analysis	Margin
LPD LCO (see note 1 below)	1.2%
LPD LSSS	29%
TM/LP LSSS	4 psid
DNB LCO LOCF	5%
DNB LCO CEAD	5%
Note: The setpoints are verified every cycle based on cycle specific core design	
Note 1: Applicable only when Incore Monitoring System is unavailable.	

The TS LSSS are designed to scram the reactor if the monitored parameters reach values that are conservatively set to protect the fuel SAFDLs. The LSSS include reactor trips such as thermal margin/low pressure (TM/LP), local power density (LPD) LSSS, variable high power trip (VHPT), low flow trip, and component pressure and water level trips. The analyses discussed in this section verified the TM/LP and LPD LSSS trip settings.

The TS LCOs provide requirements for parameters associated with the DNB LCO and LPD LCO. The DNB LCO is designed to protect the DNB SAFDL. The LPD LCO is more restrictive and is designed to protect against the LOCA linear heat generation rate (LHGR) limit when the incore detectors are not in service.

The methodology used in the setpoint verification analyses has been approved by the USNRC and is described in Reference 19.

The LPD LSSS barn and results are presented in Figure 4-2 and Figure 4-3, respectively. The TM/LP trip functions analyzed are presented in Figure 4-4 and Figure 4-5. The DNB LCO barn and results of the transient simulations are presented in Figure 4-6, Figure 4-7, and Figure 4-8, respectively. The LPD LCO barn and results are presented in Figure 4-9 and Figure 4-10, respectively. The verification of DNB LCO, LPD LCO, TM/LP LSSS and LPD LSSS is redone for each reload to ensure margin to SAFDLs.

The LSSS and LCO functions are unchanged from the current TS/COLR settings.

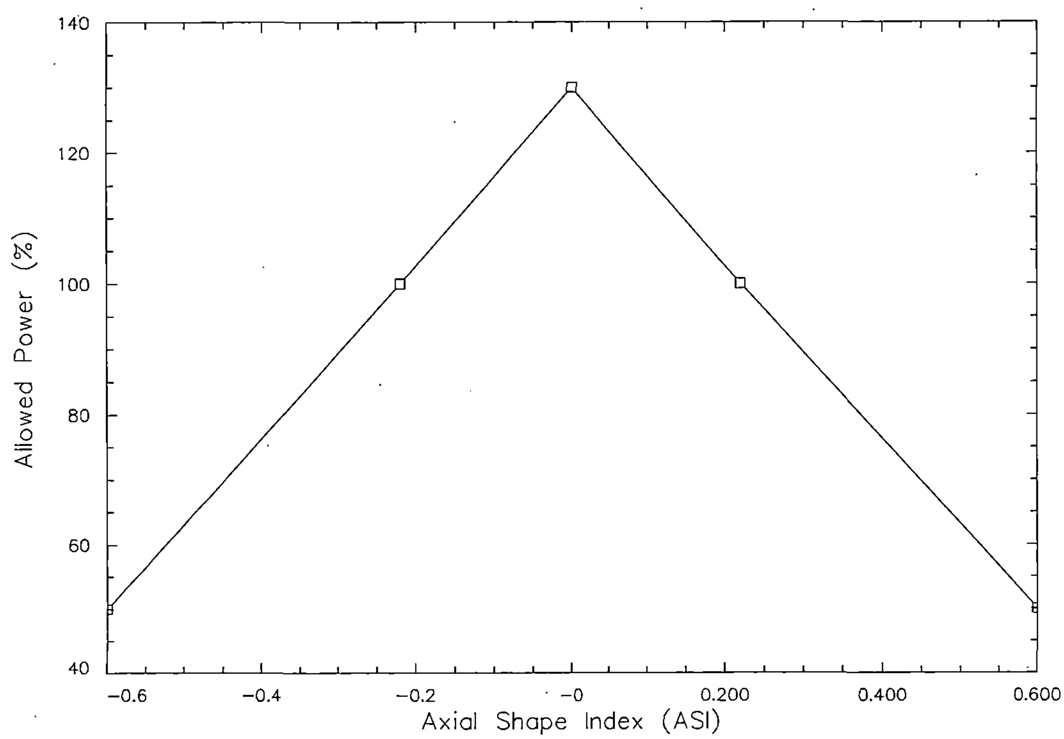

Figure 4-2: LPD - High Trip Setpoint

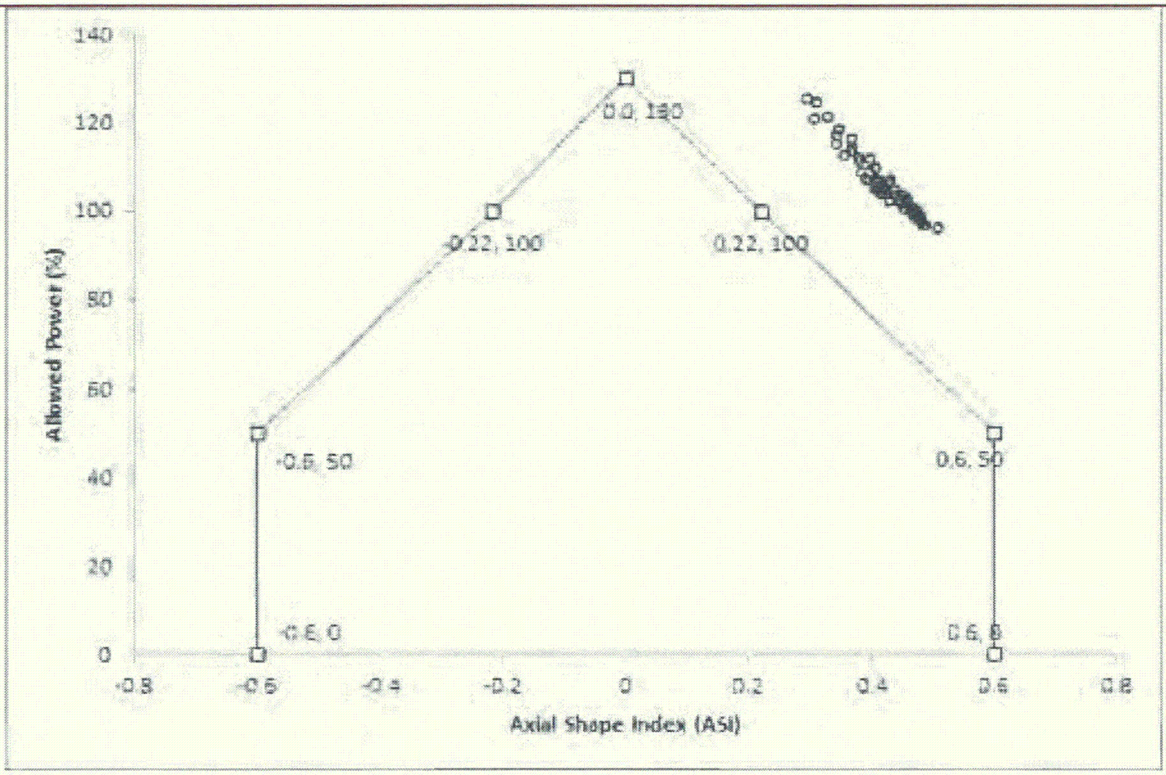

Figure 4-3: LPD LSSS Verification Results

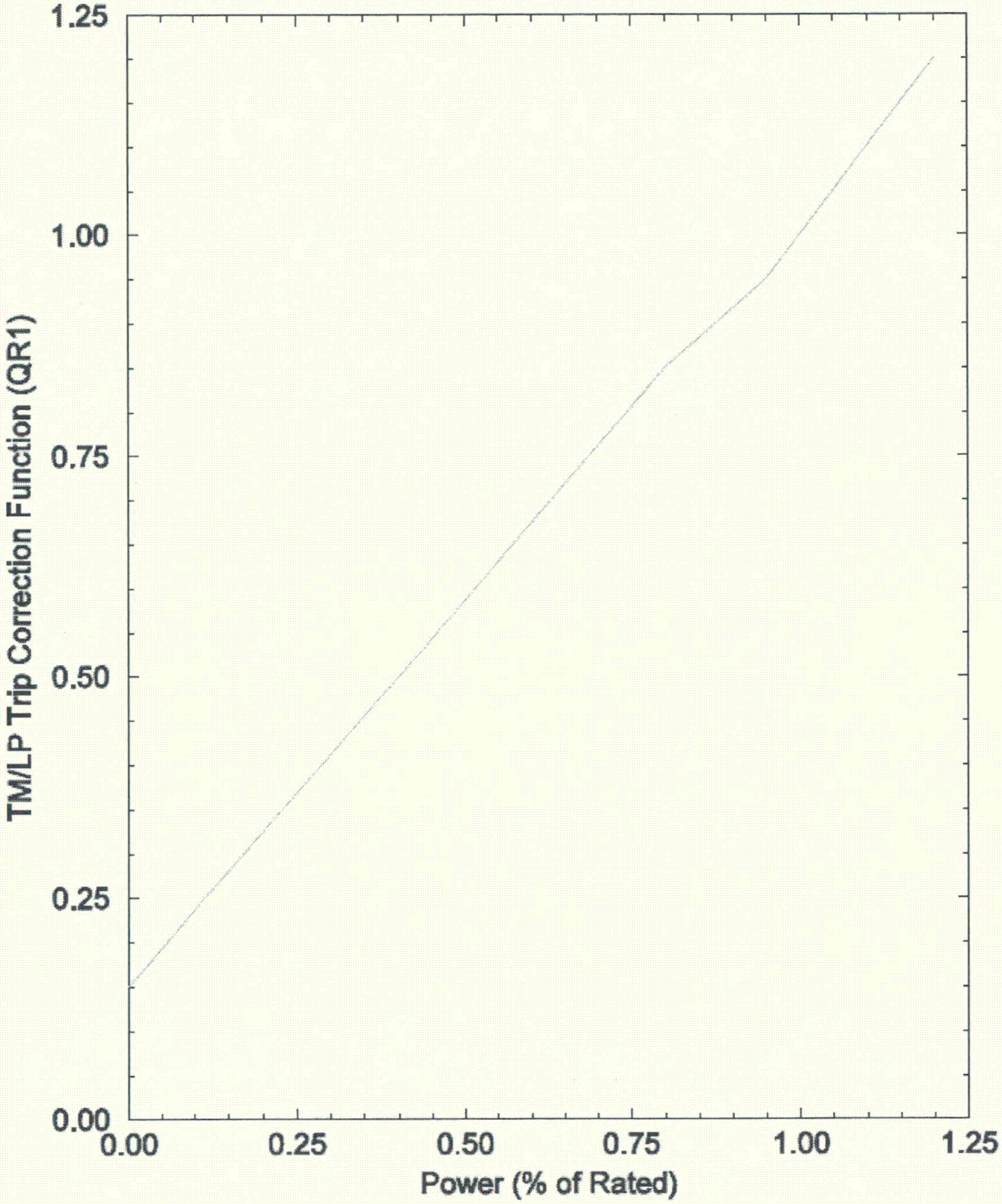

Figure 4-4: TM/LP Trip Setpoint - QR1 Function

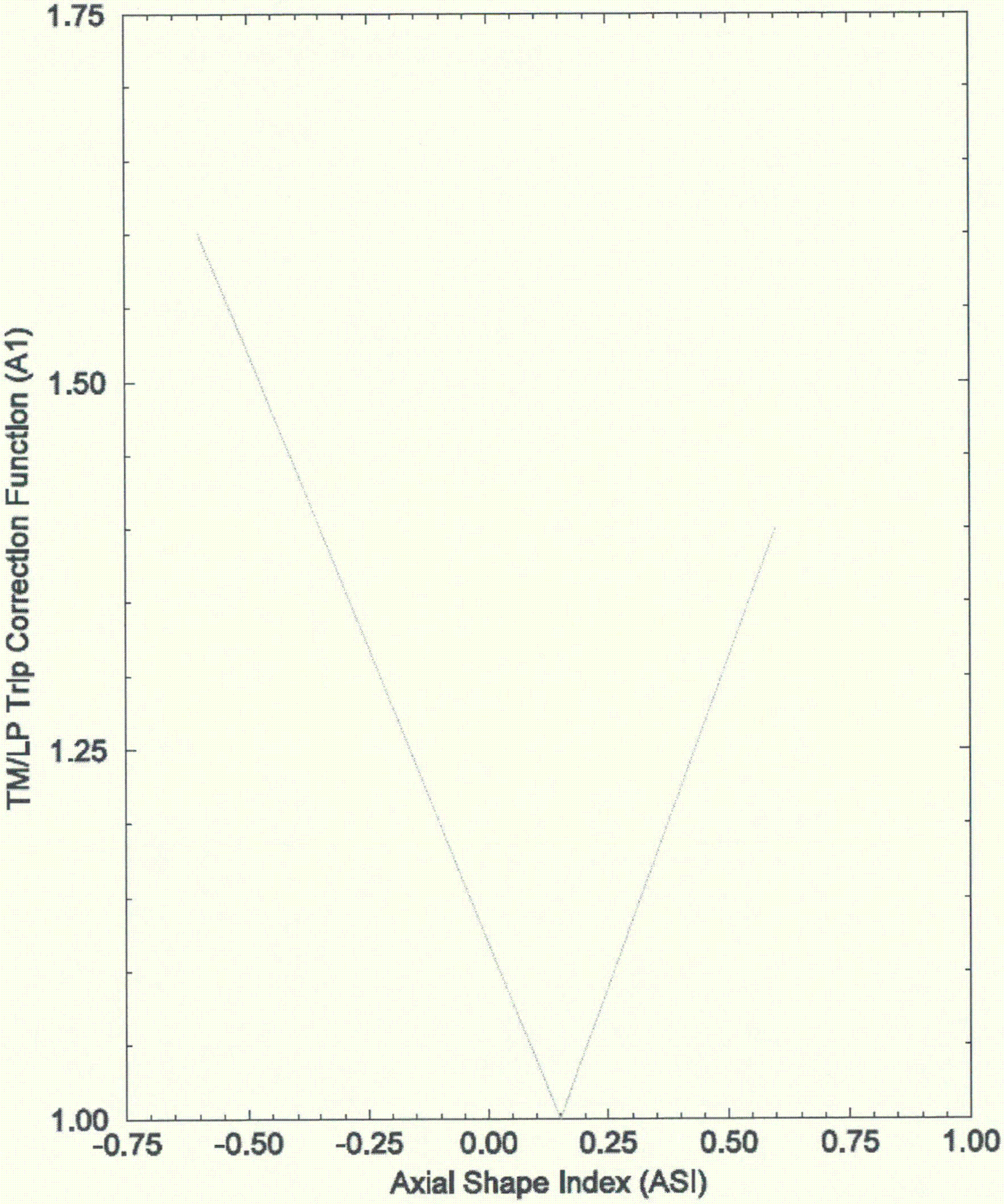

Figure 4-5: TM/LP Trip Setpoint - QA Function

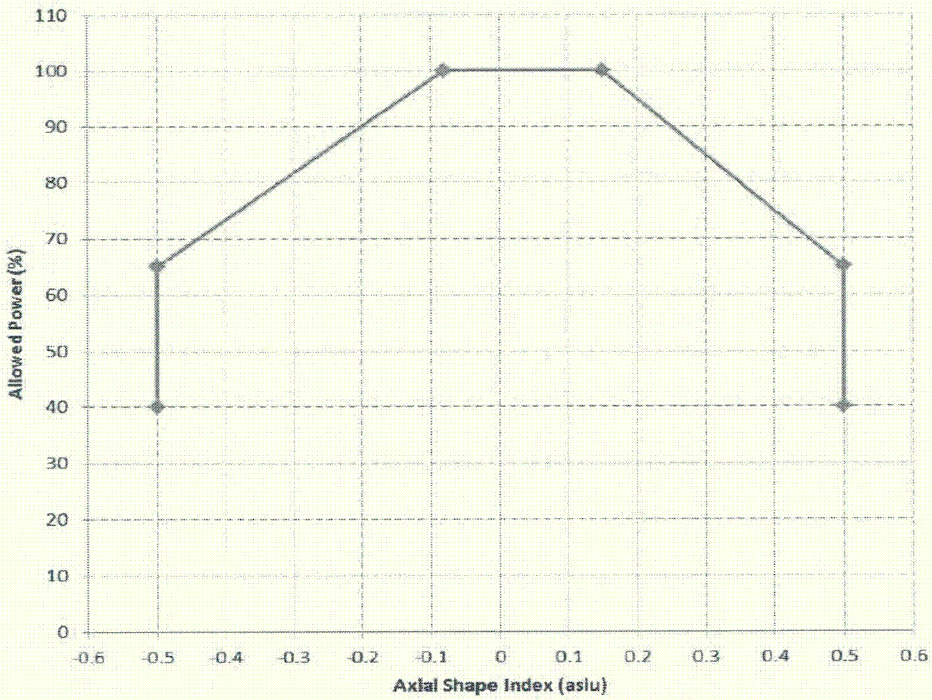

Figure 4-6: ASI Limits for DNB vs. Thermal Power

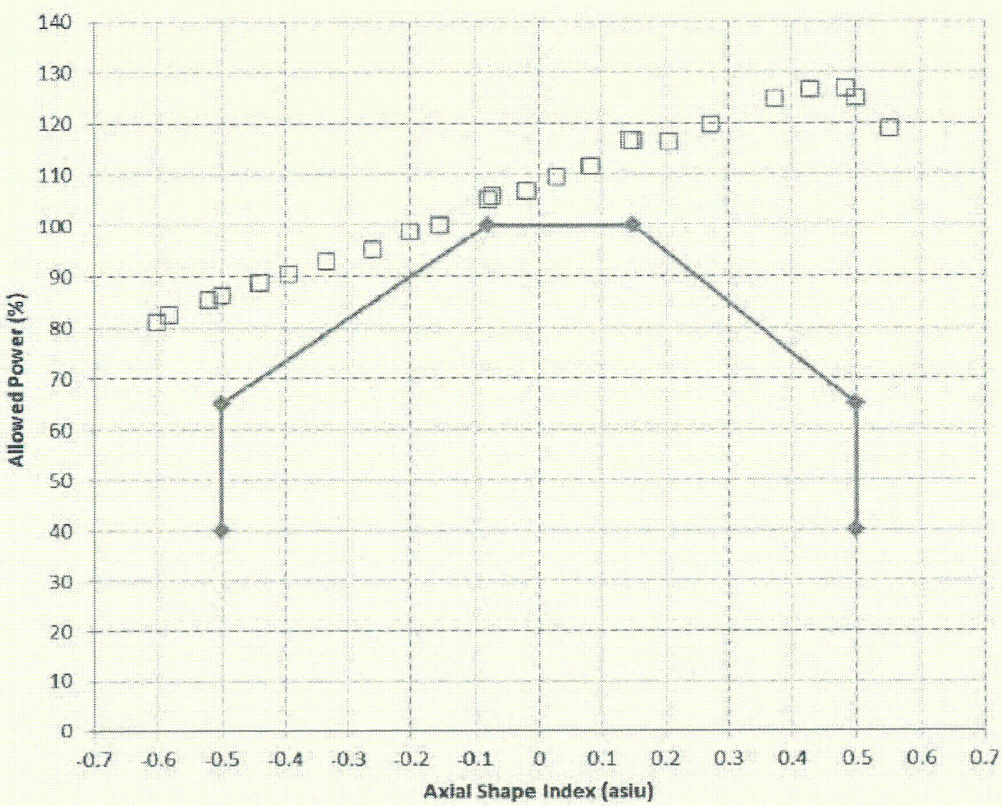

Figure 4-7: DNB LCO CEAD Results

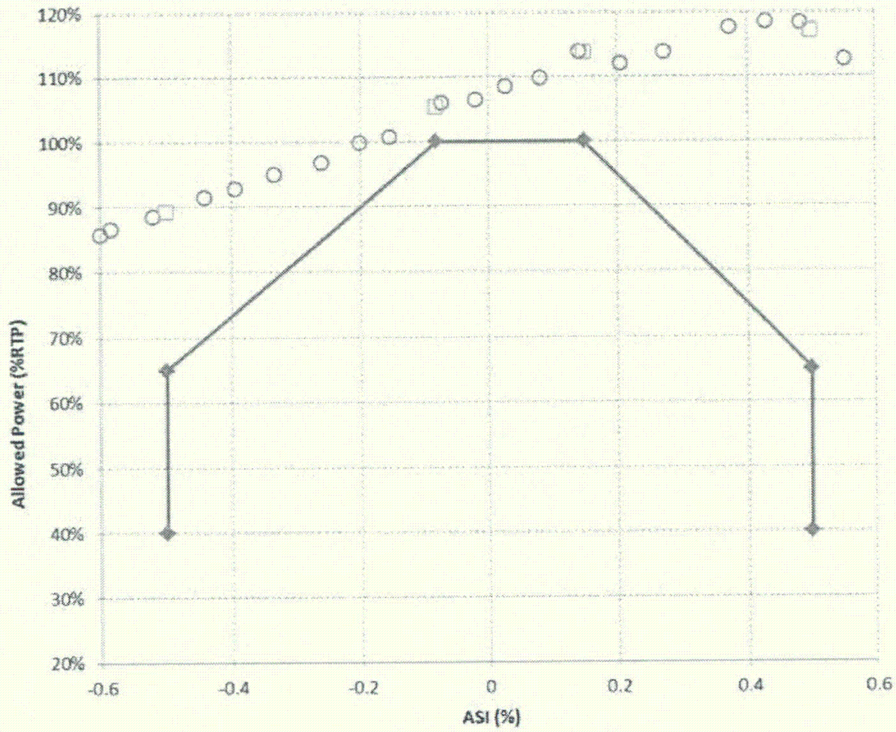

Figure 4-8: DNB LCO LOCF Results

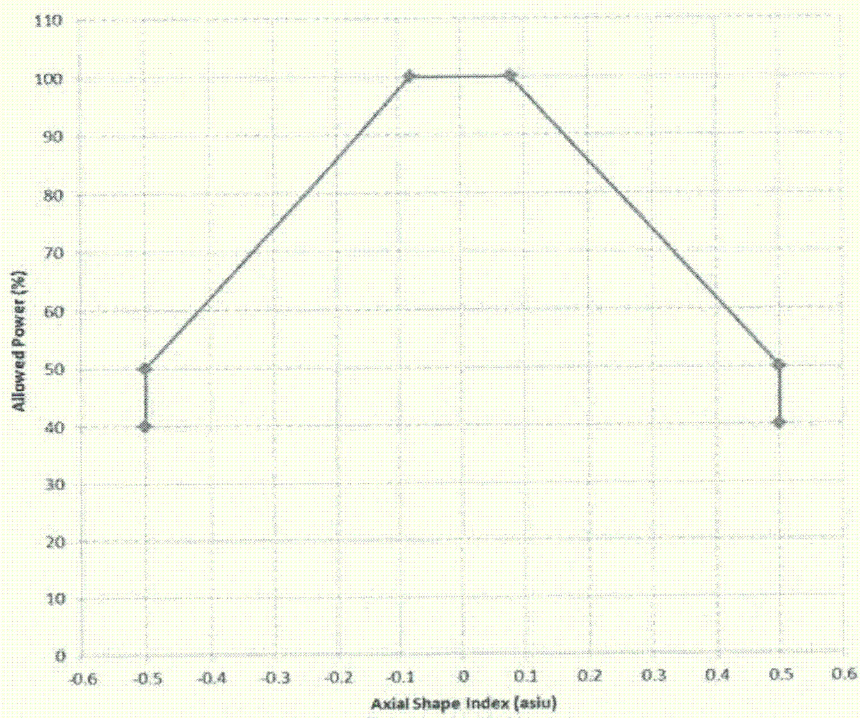

Figure 4-9: ASI Limits for LHR vs. Maximum Allowable Power Level when Using the Excore Detectors

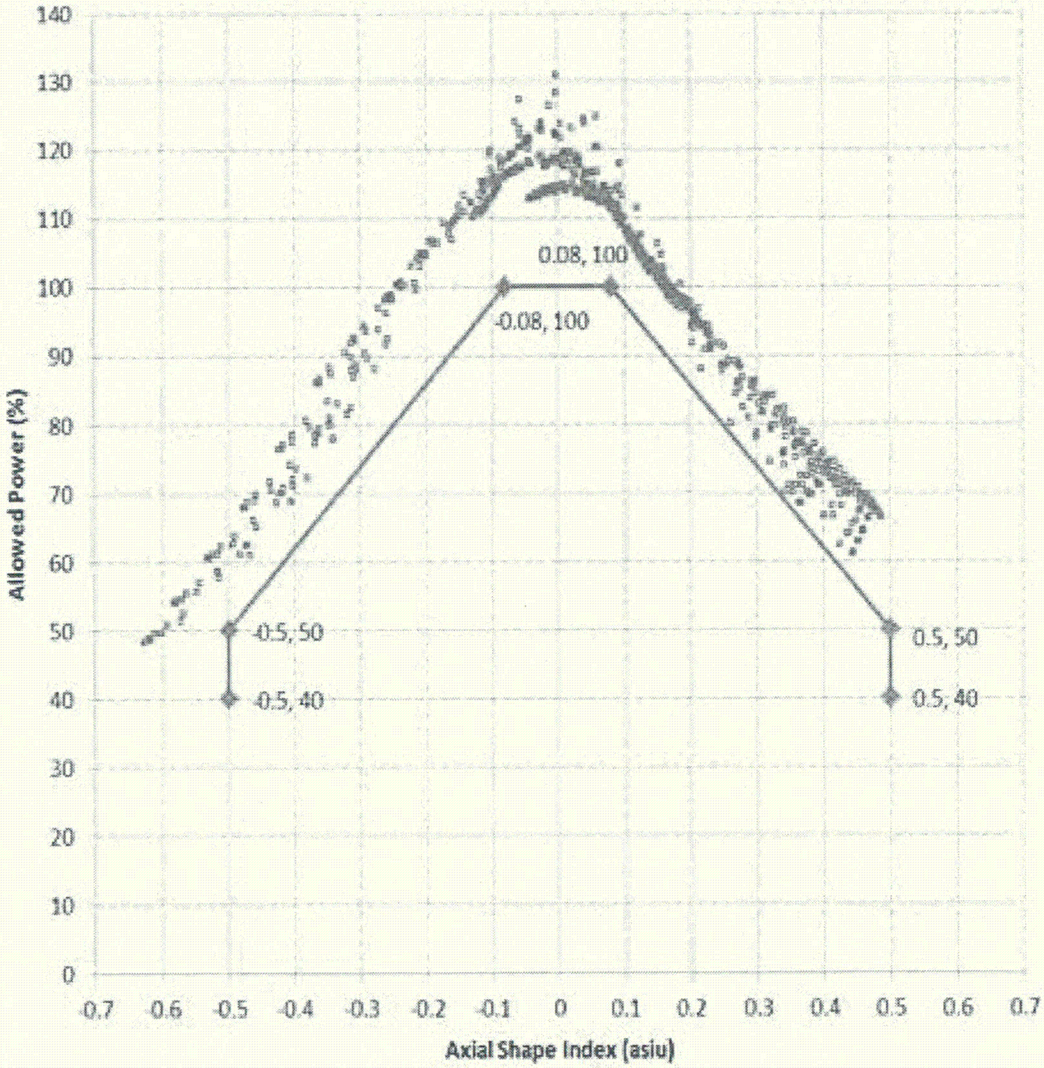

Figure 4-10: LPD LCO Verification Results

4.5.5.1 Thermal Margin Safety Limit Line Verification

The Thermal Margin Safety Limit Lines (TMSLLs) at St. Lucie Unit 2 are a series of isobars in power and inlet temperature that establish the operating frontiers in power and temperature at each pressure such that DNB in the core and hot leg saturation are both nominally avoided.

The St. Lucie Unit 2 TMSLLs are nominally based lines and therefore are analyzed using nominal values for all parameters without accounting for uncertainties.

The St. Lucie Unit 2 TMSLLs are verified using the following approach:

Each isobar is made up of two regions. The first, flatter region is established by hot leg saturation and the second, steeper portion is established by DNB. The axial shape used is the TMSLL design basis shape for St. Lucie Unit 2 and is shown in Figure 4-12.

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The TMSLLs presented in Figure 4-11 are the same as in the current TS and were verified to be conservative for use with the HTP correlation for the St. Lucie Unit 2 transition.

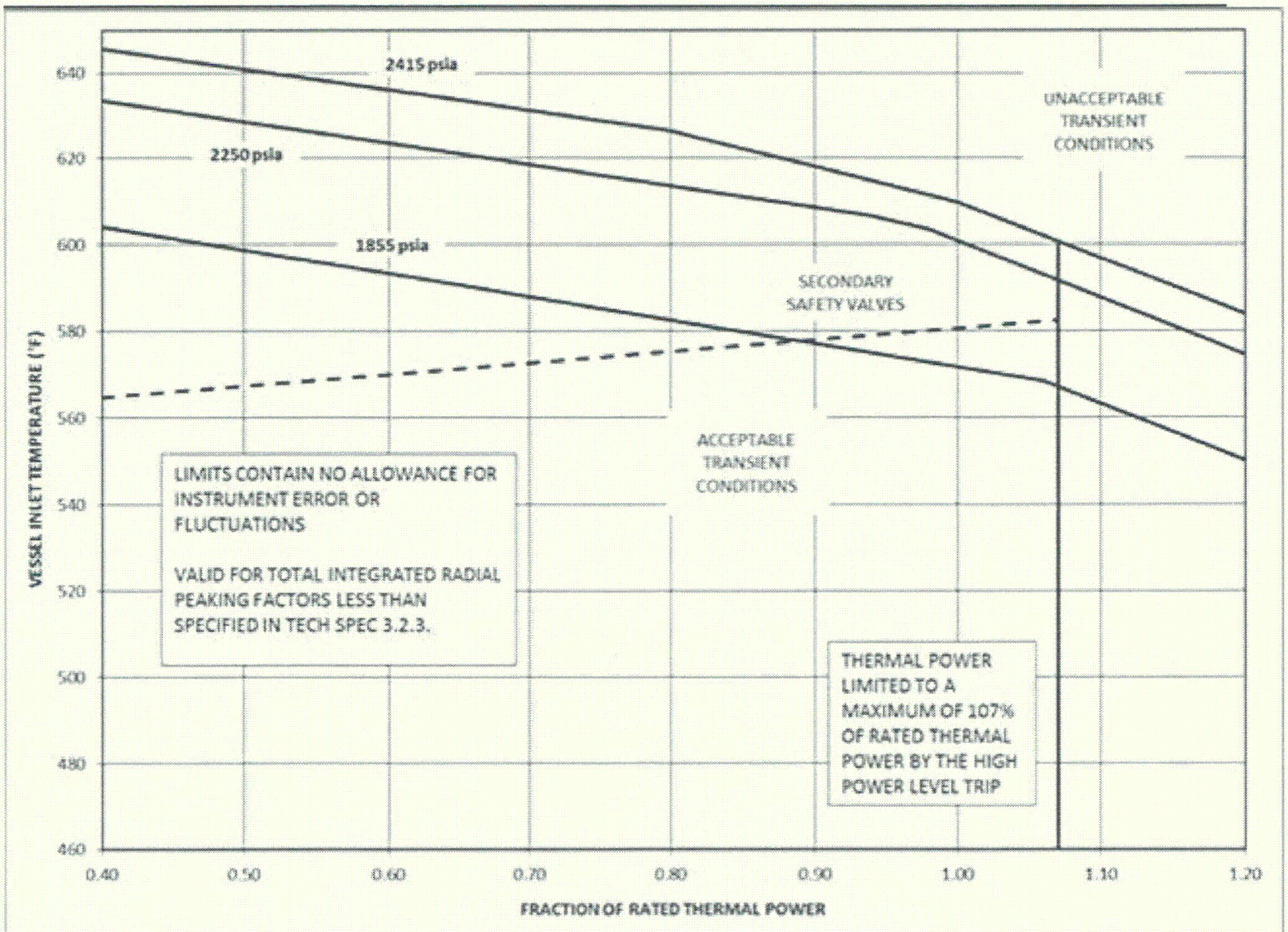

Figure 4-11: Thermal Margin Safety Limit Lines

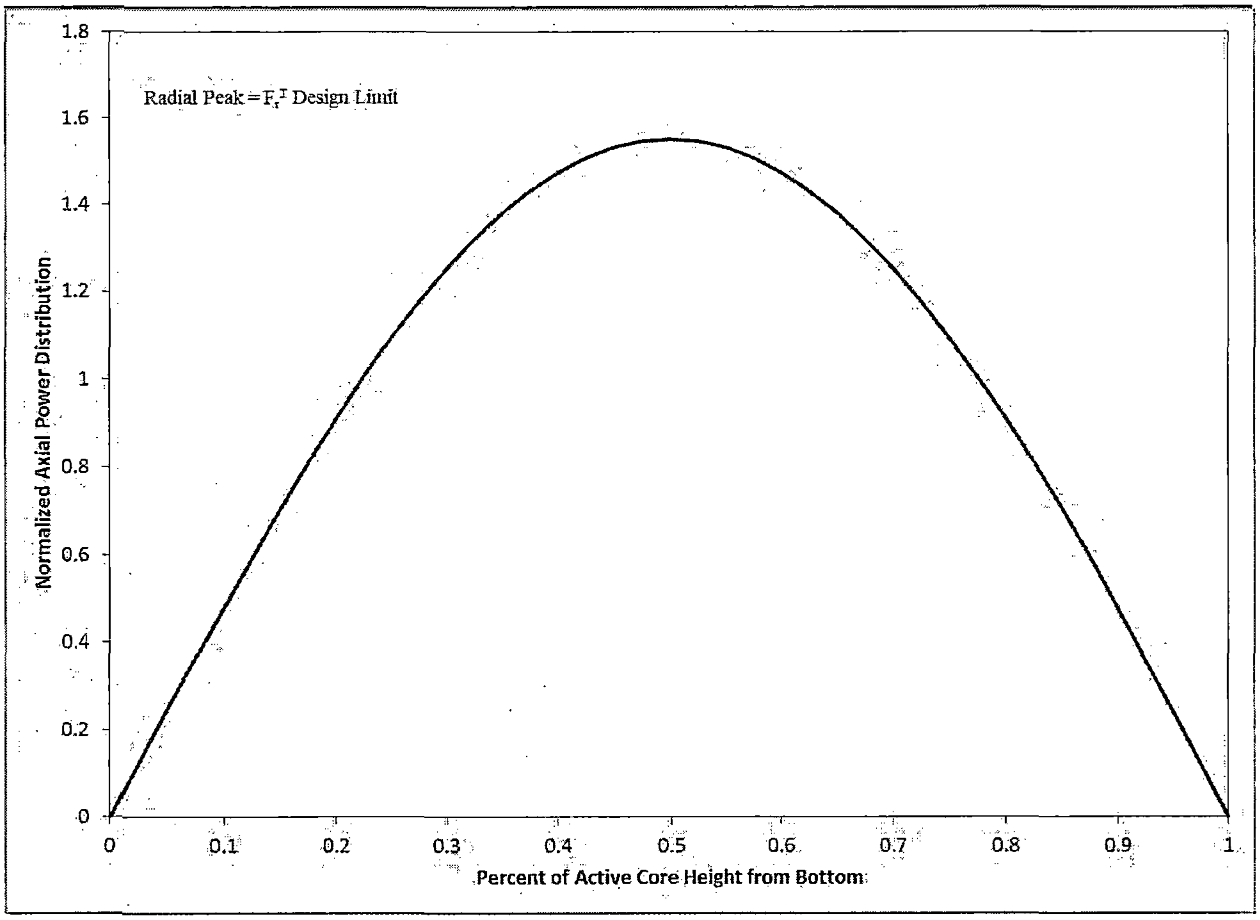

Figure 4-12: Axial Power Distribution for Thermal Margin Limit Lines

5.0 Accident and Transient Analyses

5.1 *Non-LOCA Analyses*

5.1.1 Introduction

This section provides information related to the St. Lucie Unit 2 nuclear power plant transient and accident analyses for the proposed transition to AREVA fuel. It includes a brief description of methodology used to evaluate the St. Lucie Unit 2 UFSAR Chapter 15 events affected by the transition to AREVA fuel. Also, a discussion is included on the basis by which the St. Lucie Unit 2 UFSAR Chapter 15 events not affected by the transition to AREVA fuel have been dispositioned. A summary report that provides a detailed description of analyses for the non-LOCA events using the AREVA methodology is found in Reference 23.

5.1.2 Computer Codes

Descriptions of the principal computer codes used in the non-LOCA transient analyses are provided below.

S-RELAP5

The S-RELAP5 (Reference 21) code is an AREVA modification of the RELAP5/MOD2 code. S-RELAP5 is used for simulation of the transient system response to loss-of-coolant accident (LOCA) as well as non-LOCA events. Control volumes and junctions are defined which describe all major components in the primary and secondary systems that are important for the event being analyzed. The S-RELAP5 hydrodynamic model is a two-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. S-RELAP5 uses a six-equation model for the hydraulic solutions. These equations include two-phase continuity equations, two-phase momentum equations, and two-phase energy equations. The six-equation model also allows both non-homogeneous and non-equilibrium situations encountered in reactor problems to be modeled.

RODEX2-2A

For non-LOCA applications, RODEX2-2A (References 9 and 22) is used to establish the fuel centerline melt linear heat rate (LHR) as a function of exposure as part of the Thermal Hydraulics portion of the AREVA fuel transition, which is discussed in Section 4.0.

COPERNIC

COPERNIC (Reference 24) performs thermal-mechanical calculations for a fuel rod under normal operating conditions. The code incorporates models to describe the thermal-hydraulic condition of the fuel rod in a flow channel; the gas release, swelling, densification and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion. The code has been extensively benchmarked and its predictive capabilities were correlated over a wide range of conditions applicable to light water reactor fuel conditions.

COPERNIC accounts for thermal conductivity degradation (TCD) with increasing rod exposure. To account for the effects of TCD in the non-LOCA S-RELAP5 simulations, COPERNIC was 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. The properties from COPERNIC were developed for beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions in accordance with Reference 21 and replaces RODEX2 for this purpose in the approved topical report. The COPERNIC fuel properties and gap coefficients were conservatively implemented relative to the RODEX2 inputs as approved in Reference 21.

XCOBRA-IIIC

The XCOBRA-IIIC analyses are performed as part of the Thermal Hydraulics portion of the AREVA fuel transition, which is discussed in Section 4.0.

5.1.3 Analysis Methodologies

The approved AREVA methodology for evaluating non-LOCA transients is described in Reference 21. For each non-LOCA transient event analysis, the nodalization, chosen parameters, conservative input and sensitivity studies are reviewed for applicability to the fuel transition in compliance with the SER for Revision 0 of the non-LOCA topical report (Reference 21).

- The nodalization used for the calculations supporting the fuel transition is specific to St. Lucie Unit 2 and is in accordance to the (Reference 21) methodology.

- The parameters and equipment states are chosen to provide a conservative estimate of the challenge to the acceptance criteria. The biasing and assumptions for key input parameters are consistent with or more conservative relative to the approved Reference 21 methodology.

- The S-RELAP5 code assessments in Reference 21 validated the ability of the code to predict the response of the primary and secondary systems to Chapter 15 non-LOCA transients and accidents. No additional model sensitivity studies are needed for this application.

The method used for the non-LOCA system transient analyses differs from that in the approved Reference 21 topical report as described below:

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Another change allowed by the Reference 21 methodology was to replace RODEX2 with COPERNIC for the purpose of generating the fuel thermal-conductivity, heat capacity and fuel pellet-to-clad gap coefficient inputs for the average core and hot spot models. This change was made to explicitly account for the effects of TCD. The properties from COPERNIC were developed for BOC and EOC conditions in accordance with Reference 21 and COPERNIC replaces RODEX2 for this purpose in the approved topical report. The COPERNIC fuel properties and gap coefficients were conservatively implemented relative to the RODEX2 inputs as approved in Reference 21.

Reference 1 incorporates M5® properties into the S-RELAP5 based non-LOCA methodology. No restrictions or requirements were identified in the SER for the Reference 1 methodology relative to its application to S-RELAP5 non-LOCA analyses.

The approved methodology for calculating the enthalpy deposition for a CEA ejection accident is given in Reference 14. No restrictions or requirements were identified in the SER for this methodology.

5.1.4 Event Disposition and Analysis

Reference 23 summarizes the Chapter 15 non-LOCA safety analyses supporting the transition to AREVA fuel. The analyses provide the required elements to demonstrate applicability of the method to St. Lucie Unit 2 and addresses the SER requirements as discussed in Section 5.1.3.

A review of each UFSAR Chapter 15 event was conducted relative to the transition to AREVA fuel.

- Several events (or subevents) are affected by the transition to AREVA fuel, specifically because of changes in thermal hydraulic performance and neutronics inputs to the safety analyses. The events (or subevents) that challenge the non-LOCA fuel related criteria, i.e., DNB and fuel centerline melt, were analyzed using the AREVA safety analysis methodology (Reference 21), as supplemented in Section 5.1.3. In addition, event specific criteria, i.e., time-to-criticality for Boron Dilution and deposited enthalpy for CEA Ejection, were analyzed with the Reference 21 and Reference 14 methodologies, respectively. The following events were analyzed for the fuel transition with respect to the fuel related criteria:

- Feedwater System Malfunctions That Result in a Decrease in Feedwater Temperature (UFSAR 15.1.1)
- Feedwater System Malfunctions That Result in an Increase in Feedwater Flow (UFSAR 15.1.2)
- Excessive Increase in Secondary Steam Flow (UFSAR 15.1.3)
- Pre-Trip Steam System Piping Failure (UFSAR 15.1.5)
- Post-Trip Steam System Piping Failure (UFSAR 15.1.6)
- Loss of Condenser Vacuum (UFSAR 15.2.3)
- Loss of Load to One Steam Generator (UFSAR 15.2.9)
- Complete Loss of Forced Reactor Coolant Flow (UFSAR 15.3.2)
- Reactor Coolant Pump Shaft Seizure (UFSAR 15.3.3)
- Uncontrolled CEA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR 15.4.1)
- Uncontrolled CEA Bank Withdrawal at Power (UFSAR 15.4.2)
- CEA Misoperation (Dropped CEA) (UFSAR 15.4.3)
- Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (UFSAR 15.4.6)
- Spectrum of CEA Ejection Accidents (UFSAR 15.4.8)
- Inadvertent Opening of a Pressurizer Safety or Relief Valve (UFSAR 15.6.1)

- Other UFSAR Chapter 15 events (or subevents) are not affected by the AREVA fuel transition because the key parameters for these events are plant related system responses (e.g., core power, decay heat, auxiliary feedwater capability, offsite power availability, safety valve setpoints and capacities, safety injection and/or charging capability, etc.) rather than the fuel design parameters. These events (or subevents) challenge criteria other than the SAFDLs, e.g., system overpressure. As such, these events will not be analyzed as part of the transition to AREVA fuel. These events (or subevents) remain bounded by the current analyses of record.

Reference 23 (Section 2.0) provides the key input parameters assumed for the non-LOCA analyses. A summary of the initial conditions assumed for each Chapter 15 non-LOCA event that was analyzed using S-RELAP5 to support the fuel transition is provided in Reference 23 (Table 2.3). Reference 23 (Table 3.1) provides a summary of the non-LOCA disposition of events. Reference 23 (Section 4.0) discusses each UFSAR Chapter 15 event in detail. The results in Reference 23 demonstrate that acceptance criteria are met for each non-LOCA event that was analyzed for the transition to AREVA fuel. The results are summarized in Table 5-1.

Table 5-1: Non-LOCA Limiting Results

UFSAR Section	Event Description	Criterion	Analytical Limit	Limiting Result
15.1.1	Decrease in Feedwater Temperature	MDNBR	1.164	1.257
		Peak LHR, kW/ft	[ ]	18.24
15.1.2	Increase in Feedwater Flow	MDNBR	1.164	1.220
		Peak LHR, kW/ft	[ ]	18.50
		Peak CLT, °F	[ ]	3385 (HZP)
15.1.3	Increase in Steam Flow	MDNBR	1.164	1.271
		Peak LHR, kW/ft	[ ]	19.12
		Peak CLT, °F	[ ]	3491 (HZP)
15.1.5	Pre-scram Main Steam Line Break	MDNBR (%fuel failure)	1.164	1.203 (0%)
		Peak LHR, kW/ft (% fuel failure)	[ ]	17.67 (0%)
15.1.6	Post-scram Main Steam Line Break	MDNBR (% fuel failure)	[ ]	1.740 (0%)
		Peak LHR, kW/ft (% fuel failure)	[ ]	17.02 (0%)
15.2.3	Loss of Condenser Vacuum	MDNBR	1.164	1.553
		Peak LHR, kW/ft	[ ]	16.04
15.2.9	Transients Resulting from the Malfunction of One Steam Generator	MDNBR	1.164	1.713
		Peak LHR, kW/ft	[ ]	15.74
15.3.2	Loss of Forced Reactor Coolant Flow	MDNBR	1.164	1.227
15.3.3	Reactor Coolant Pump Rotor Seizure	MDNBR (% fuel failure)	1.164	1.205 (0%)

Table 5-1: Non-LOCA Limiting Results *(Continued)*

UFSAR Section	Event Description	Criterion	Analytical Limit	Limiting Result
15.4.1	Uncontrolled CEA Withdrawal from a Subcritical or Low Power Startup Condition	MDNBR	1.164	1.994
		Peak CLT, °F	[ ]	3194
15.4.2	Uncontrolled CEA Withdrawal at Power	MDNBR	1.164	1.177
		Peak LHR, kW/ft	[ ]	16.43
15.4.3	CEA Misoperation/CEA Drop	MDNBR	1.164	1.554
		Peak LHR, kW/ft	[ ]	15.71
15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant/Boron Dilution	Min. time to loss of shutdown margin, min.	15	15.08
			30	30.59
15.4.8	CEA Ejection	MDNBR (% fuel failure)	1.164	1.179 (0%)
		Peak CLT, °F (% fuel failure)	[ ]	4876 (0%)
		Total deposited enthalpy limit, cal/gm	230 (HFP) 150 (HZP)	144.1 (HFP) 26.9 (HZP)
15.6.1	Inadvertent Opening of Pressurizer Safety or Relief Valve	MDNBR	1.164	1.237

5.1.5 Conclusions

The non-LOCA transient analyses were performed in accordance with the Reference 21 non-LOCA methodology, as supplemented in Section 5.1.3. Reference 23 demonstrates the application of the AREVA non-LOCA safety analysis methodology to St. Lucie Unit 2 for the fuel transition and shows that acceptance criteria are met for each non-LOCA event that was analyzed for the transition to AREVA fuel.

5.2 *Loss-of-Coolant Accident Analyses*

The loss-of-coolant accident (LOCA) is analyzed to assure that the design bases for the Emergency Core Cooling System (ECCS) satisfy the requirements of 10 CFR 50.46 acceptance criteria for the St. Lucie Unit 2 transition to AREVA fuel. Summary reports that provide a detailed description of supporting small break LOCA and realistic large break LOCA (SBLOCA and RLBLOCA) analyses are found in References 28 and 29, respectively. Additional results supporting the SBLOCA analysis are found in Reference 32 (responses to SNPB RAI-11 through SNPB RAI-20).

5.2.1 Small Break Loss-of-Coolant Accident

A SBLOCA is defined as a break in the RCS pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the reactor coolant pump, which results in the largest amount of inventory loss and the largest fraction of ECCS fluid being lost to the break. This behavior produces the greatest degree of core uncovery and the longest fuel rod heatup time.

The SBLOCA event is characterized by a slow depressurization of the RCS with a reactor trip occurring on a low pressurizer pressure signal. The safety injection actuation signal (SIAS) occurs when the system pressure continues to drop. For some of the break sizes, the rate of inventory loss from the primary system is such that the charging system and High Pressure Safety Injection (HPSI) pumps cannot preclude significant core uncovery. The slow RCS depressurization rate extends the time required to reach the safety injection tank (SIT) pressure or to recover core liquid level on charging and HPSI flow. Core recovery for the limiting break begins when the charging and HPSI flow to the RCS exceeds the mass flow rate out of the break, followed by injection of SIT flow.

The AREVA SBLOCA evaluation methodology (EM) simulates thermal-hydraulic response of the primary and secondary systems and hot fuel rod and requires the use of two computer codes, S-RELAP5 and RODEX2/2A (Reference 16). The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated. The EM has been reviewed and approved by the USNRC to perform SBLOCA analyses.

Results from the St. Lucie Unit 2 SBLOCA analysis show that the 10 CFR 50.46(b) acceptance criteria for PCT, maximum oxide thickness, and hydrogen generation are met with significant margin. Analysis results show that the limiting PCT occurred for a 2.70-inch diameter cold leg pump discharge break. This case yielded a limiting PCT of 2057 °F as provided in Reference 32 (response to SNPB RAI-15). The transient maximum local oxidation is less than 9%. The total maximum local oxidation is less than 12%, including a pre-transient oxidation of 2.3925%. The maximum core-wide oxidation is less than 0.3%.

5.2.2 Large Break Loss-of-Coolant Accident

A large break loss-of-coolant accident (LBLOCA) is initiated by a postulated large rupture in the RCS cold leg. The RCS depressurizes rapidly and the reactor is shut down by coolant voiding in the core. An SIAS occurs on either high containment pressure or low RCS pressure. Pumped ECCS and passive SIT fluid injection actuates to mitigate the transient.

The St. Lucie Unit 2 RLBLOCA analysis is performed by applying the S-RELAP5, RODEX3A, and ICECON computer codes. The EM is documented in Reference 30; specific alternative methods to the EM are outlined in the RLBLOCA summary report. These alternative methods are a response to USNRC inquiries related to the methodology updates to the EM. This altered methodology is referred to as the "transition program or transition package." This methodology follows the Code Scaling, Applicability, and Uncertainty evaluation approach (Reference 31), which outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties for the RLBLOCA analysis. The approach described in the summary report has been used successfully in multiple applications for support of licensing AREVA fuel transitions.

Results from the St. Lucie Unit 2 RLBLOCA analysis show that the 10 CFR 50.46(b) acceptance criteria for PCT, maximum oxide thickness, and hydrogen generation are met with significant margin. Analysis results show that the limiting PCT occurred for a fresh UO2 rod in a case with no offsite power availability. This case yielded a limiting PCT of 1732 °F.

6.0 Summary and Conclusion

This report shows acceptability for the application of the AREVA CE-16 HTP™ fuel design at St. Lucie Unit 2. The results displayed within the report show compliance of the AREVA CE-16 HTP™ fuel design with USNRC-approved topical reports regarding mechanical and structural analyses, nuclear design analyses, thermal-hydraulics analyses for steady state and transient core performance, and non-LOCA / LOCA safety analyses addressing transient and accident conditions. Alternative methods to the approved topical reports are conservatively applied and clearly described within the document, where appropriate.

Note that demonstration of the evaluation methodologies has been performed with a submittal core design. The submittal core design was developed to provide key safety parameters to support the transition from Westinghouse fuel to AREVA CE-16 HTP™ fuel prior to the development of cycle-specific designs. This provides assurance that the plant licensing bases are met for the anticipated operation of the AREVA CE-16 HTP™ fuel during the transition and full core cycles.

The AREVA fuel design will be the CE-16 HTP™ fuel consisting of a 16x16 assembly configuration with M5® fuel rods, Zircaloy-4 MONOBLOC™ corner guide tubes, an Alloy 718 HMP™ spacer at the lowermost axial elevation, Zircaloy-4 HTP™ spacers in all other axial elevations, a FUELGUARD™ lower tie plate (LTP), and the AREVA reconstitutable upper tie plate (UTP).

The AREVA CE-16 HTP™ fuel design for St. Lucie Unit 2 is similar and has the same design features as the AREVA CE-14 fuel design operating in St. Lucie Unit 1. It is also similar to the AREVA CE-16 HTP™ lead fuel assemblies operated in San Onofre Unit 2 as well as the fuel rods operated in the AREVA CE-16 HTP™ Palo Verde Lead Fuel Assemblies. The design features of the AREVA CE-16 HTP™ fuel design planned for St. Lucie Unit 2 have demonstrated excellent fuel performance. The HTP™ / HMP™ spacer grids are very resistant to flow induced grid-to-rod fretting failures, the FUELGUARD™ LTP is effective at protecting the fuel from debris in the reactor coolant system, and the M5® cladding has very low oxidation and hydrogen pickup rates.

In conclusion, this report supports the use of AREVA CE-16 HTP™ fuel at St. Lucie Unit 2.

7.0 References

1. BAW-10240(P)(A), Revision 0, “Incorporation of M5 Properties in Framatome ANP Approved Methods."

2. EMF-92-116(P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs.”

3. BAW-10227(P)(A), Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel.”

4. EMF-92-116(P)(A), Revision 0, Supplement 1, Revision 0(P)(A), “Generic Mechanical Design Criteria for PWR Fuel Designs." (Supplement 1 is pending approval).

5. Letter NRC:14:049, P.Salas (AREVA Inc.) to USNRC, “Response to a Request for Additional Information Regarding EMF-92-116(P)(A), Revision 0, Supplement 1, Revision 0.”

6. NUREG-0800, Revision 2, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition."

7. USNRC Information Notice, IN-2009-23, "Nuclear Fuel Thermal Conductivity Degradation."

8. USNRC Information Notice, IN-2012-09, “Irradiation Effects on Spacer Grid Crush Strength."

9. XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, “RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.”

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

11. BAW-10172(P)(A), Revision 0, "Mark-BW Mechanical Design Report."

12. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results."

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

14. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors."

15. XN-75-27(A) and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors", Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

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

17. EMF-2087(P)(A), Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications."

18. EMF-92-153(P)(A), Revision 1, "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel."

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

20. XN-NF-75-21(P)(A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation."

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

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

23. ANP-3347(P), Revision 0, "St. Lucie Unit 2 Fuel Transition Chapter 15 Non-LOCA Summary Report."

24. BAW-10231(P)(A) Revision 1, "COPERNIC Fuel Rod Design Computer Code."

25. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations."

26. Energia Nucleare, Volume 14, No. 9, September 1967, "Studies on Burnout, Part 3 – A New Correlation for Round Ducts and Uniform Heating and Its Comparison with World Data" L. Biasi et. al.

27. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing."

28. ANP-3345(P), Revision 1, "St. Lucie Unit 2 Fuel Transition Small Break LOCA Summary Report."

29. ANP-3346(P), Revision 0, "St. Lucie Unit 2 Fuel Transition Realistic Large Break LOCA Summary Report."

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

31. NUREG/CR-5249, EGG-2552, Technical Program Group, "Quantifying Reactor Safety Margins," October 1989.

32. ANP-3440(P), Revision 1, "St. Lucie Unit 2 Fuel Transition: Responses to NRC Questions SRXB-RAI-1 and SNPB RAI-2 thru SNPB RAI-20."