UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2016

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING TRANSITIONING TO AREVA FUEL (CAC NO. MF5495)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 182 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2 (PSL-2). This amendment consists of changes to the Technical Specifications in response to your application dated December 30, 2014, as supplemented by letters dated March 23, 2015; June 2, 2015; June 18, 2015; July 30, 2015; October 2, 2015; November 3, 2015; and December 8, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15002A091, ML15084A011, ML15161A316, ML15181A290, ML15219A184, ML15279A226, ML15322A134, and ML15356A184, respectively).

This amendment modifies the Technical Specifications to allow the use of AREVA fuel at PSL-2. Specifically, PSL-2 will transition to the AREVA Combustion Engineering 16x16 High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material. PSL-2 is planning to transition to the AREVA fuel beginning with Cycle 23. PSL-2 is currently using Westinghouse Combustion Engineering 16x16 fuel.

The licensee also requested an exemption from the requirements of Title 10 of *the Code of Federal Regulations* (10 CFR), Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," to 10 CFR Part 50, to allow the use of fuel rods clad with M5® alloy for future reload applications. The NRC granted that exemption in separate correspondence dated April 19, 2016 (ADAMS Accession No. ML16015A286).

NOTICE: Enclosure 2 to this letter contains proprietary information. Upon separation from Enclosure 1, this letter is DECONTROLLED.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert L. Hadney for

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 182 to Renewed NPF-16
2. Safety Evaluation (non-public)
3. Safety Evaluation (public)

cc w/ Enclosures 1 and 2: Addressee only
cc w/ Enclosures 1 and 3: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER AND LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 182
Renewed License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated December 30, 2014, as supplemented by letters dated March 23, 2015; June 2, 2015; June 18, 2015; July 30, 2015; October 2, 2015; November 3, 2015; and December 8, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, Renewed Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 182 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented upon the start of the St. Lucie Plant, Unit No. 2, Cycle 23 refueling outage to support the AREVA fuel transition project plan.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: April 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 182

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of Renewed Operating License No. NPF-16 with the attached revised pages. The revised pages are identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3
7	7

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 2-2	3/4 2-2
5-3	5-3
6-20e	6-20e
	6-20ea

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 FR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 182 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

NRC dated December 9, 2003, and October 29, 2004, in response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.15.d, shall be within 36 months in a staggered test basis, plus the 138 days allowed by SR 4.0.2, as measured from November 13, 2006, which is the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

N. FATES3B Safety Analyses (Westinghouse Fuel Only)

FATES3B has been specifically approved for use for St. Lucie Unit 2 licensing basis analyses based on FPL maintaining the more restrictive operational/design radial power fall-off curve limits as specified in Attachment 4 to FPL Letter L-2012-121, dated March 31, 2012 as compared to the FATES3B analysis radial power fall-off curve limits. The radial power fall-off curve limits shall be verified each cycle as part of the Reload Safety Analysis Checklist (RSAC) process.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight April 6, 2043.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

J. E. Dyer, Director

Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan
3. Appendix C, Antitrust Conditions
4. Appendix D, Antitrust Conditions

Date of Issuance: October 2, 2003

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100% of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_r^T curve of COLR Figure 3.2-3.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO™ or M5® clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 91 full-length control element assemblies and no part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

61. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), "St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit," L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

65. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April 1999.

66. WCAP-7908-A, Rev. 0, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod", December 1989.

67. WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code", January 1975.

68. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods", January 1975.

69. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results," Siemens Power Corporation, January 1997.

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

71. 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).

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

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

73. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986.

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

75. EMF-92-116(P)(A), Rev. 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February, 1999.

76. BAW-10240(P)(A), Rev.0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., May 2004.

77. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

78 EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," January 2005.

79. EMF-1961(P)(A), Revision 0, "Statistical/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

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

81. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," October 1983.

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

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

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle on the NRC.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated December 30, 2014, as supplemented by letters March 23, 2015; June 2, 2015; June 18, 2015; July 30, 2015; October 2, 2015; November 3, 2015; and December 8, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15002A091, ML15084A011, ML15161A316, ML15181A290, ML15219A184, ML15279A226, ML15322A134, and ML15356A184, respectively), Florida Power & Light Company, et al. (FPL, or the licensee) requested to amend Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2 (PSL-2).

The licensee's supplements dated June 2, 2015; June 18, 2015; July 30, 2015; October 2, 2015; November 3, 2015; and December 8, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 9, 2015 (80 FR 32620).

The proposed amendment would modify the Technical Specifications (TSs) to allow the use of AREVA fuel at PSL-2. Specifically, as indicated in its license amendment request (LAR), PSL-2 will transition to the AREVA Combustion Engineering (CE) 16x16 High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material. PSL-2 is planning to transition to the AREVA fuel beginning with Cycle 23. PSL-2 is currently using Westinghouse CE 16x16 fuel.

The licensee also requested an exemption from the requirements of Title 10 of *the Code of Federal Regulations* (10 CFR), Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," to 10 CFR Part 50, to allow the use of M5® fuel rod cladding for future reload applications. The U.S. Nuclear Regulatory Commission (NRC or the Commission) granted that exemption in separate correspondence dated April 19, 2016 (ADAMS Accession No. ML16015A286).

Enclosure 3

Transitioning to AREVA CE 16x16 HTP™ fuel requires changes to the license and TSs. In its application, the licensee proposed to revise License Condition 3.N, "FATES3B Safety Analyses," to specifically address Westinghouse Fuel. The licensee also proposed to revise TS Surveillance Requirement (SR) 4.2.1.3, "Excore Detector Monitoring System," to delete the surveillance requirement for the linear heat rate using W(z). In addition, the licensee proposed to revise TS 5.3.1, "Fuel Assemblies," to add M5® as a cladding material and TS 6.9.1.11, "Core Operating Limits Report (COLR)," to add NRC-approved AREVA methods for neutronics, fuel mechanical, thermal-hydraulics, and safety analyses. The licensee also indicated that, as a result of these changes, it will need to make changes to the TS Bases and COLR to make them consistent with the proposed TS changes and provided marked up pages for the TS Bases and COLR.

2.0 REGULATORY EVALUATION

The licensee submitted its application during PSL-2 Cycle 21. PSL-2 is currently operating in Cycle 22 and uses Westinghouse CE 16x16 fuel. The proposed amendment would allow PSL-2 to transition to AREVA CE 16x16 HTP™ fuel design, beginning with Cycle 23. The AREVA fuel design, with its HTP™ grids, is the first full reload application at PSL-2, which is a CE 16x16 plant. The HTP™ grid design has been used in CE plants of 14x14 design. The HTP™ grid design has been successfully used in the industry for other fuel types such as Westinghouse 15x15 and 17x17 plants. Transitioning to AREVA 16x16 HTP™ fuel requires several license and TSs changes. The proposed changes include the change in fuel assembly design specifications to include the use of M5® material for cladding, removal of a linear heat generation rate (LHGR) surveillance requirement due to the operation on ex-core detector monitoring system, and the inclusion of AREVA's approved methodology and analysis topical reports. Since the AREVA fuel uses M5® as cladding material, a 10 CFR 50.46 and 10 CFR 50, Appendix K, exemption request was included in the licensee's application. This exemption was granted in separate correspondence, dated April 19, 2016.

The regulatory requirements and guidance that the NRC staff applied in the review of the licensee's application include the following:

The regulations in 10 CFR, Part 50, "Domestic Licensing of Production and Utilization Facilities," establish the fundamental regulatory requirements for nuclear power plants.

Per 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," whenever a holder of an operating license desires to amend the license, an application for an amendment must be filed with the Commission fully describing the changes desired, and following, as far as applicable, the form prescribed for original applications.

The regulations in 10 CFR 50.92 "Issuance of amendment," provide requirements for issuance of an amendment to a license, construction permit, or early site permit. Specifically, per 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. As indicated in 10 CFR 50.36(b) the license

includes TSs derived from the analyses and evaluation included in the safety analysis report submitted as part of the application for an initial license.

Pursuant to 10 CFR 50.36(c), the TSs are required to include the following five categories related to station operation:

(1) Safety limits, limiting safety system settings, and limiting control settings;
(2) Limiting conditions for operation;
(3) Surveillance requirements;
(4) Design features; and
(5) Administrative controls.

The proposed surveillance requirement for the LHGR is consistent with AREVA methodology when operating an excore detector system for monitoring the LHGR. The surveillance requirement using W(z) is no longer required and is deleted. The new AREVA methodologies that are applicable to the new fuel design will be inserted in TS 6.9.1.11.b.

Section 50.46 of 10 CFR establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance. Evaluation models for loss-of-coolant accident (LOCA) events are defined in 10 CFR 50.46. This definition, which the NRC staff considers applicable to non-LOCA analyses as well, states that:

> An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated [LOCA]. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Appendix K to 10 CFR Part 50 establishes required and acceptable features of evaluation models for heat removal by the ECCS during and after the blowdown phase of a LOCA.

Section II of Appendix K to 10 CFR Part 50, also written for LOCA analyses, and considered by the NRC staff to be applicable to non-LOCA analyses as well, contains the documentation requirements for ECCS evaluation models. The regulations state:

> 1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

1. b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability – Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this Appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

Section III of Appendix B to 10 CFR Part 50 governs references to design control measures in the COLR. It states, in part, "Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests."

In 10 CFR 50.67, "Accident source term," paragraph (b)(2) states that the NRC may issue a requested license amendment under that section only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem [roentgen equivalent man])[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

General Design Criteria

The NRC's evaluation criteria are also based on the following General Design Criteria (GDC) of Appendix A to 10 CFR Part 50:

1. GDC 4, which requires that structures, systems, and components important to safety be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

2. GDC 10, which requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

3. GDC 11, which requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

4. GDC 12, which requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.

5. GDC-15, which requires the reactor coolant system (RCS) and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs.

6. GDC 19, "Control room," which states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem [0.05 Sv] whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and

(2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

7. GDC 20, which requires that the reactor core protection system be designed to initiate the reactivity control systems automatically to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and to sense accident conditions and to automatically initiate operation of systems and components important to safety.

8. GDC 25, which requires that the protection system be designed to assure that the SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

9. GDC 26, which requires, in part, that two independent reactivity control systems of different design principles be provided, with both systems capable of reliably controlling the rate of reactivity changes from planned, normal power changes.

10. GDC 27, which requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.

11. GDC 28, which requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and

12. GDC 35, which requires, in part, that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel and fuel clad damage that could interfere with continued effective core cooling will be prevented.

Standard Review Plan

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), provides guidance to the NRC staff in performing safety reviews of operating license applications (including requests for amendments) under 10 CFR Part 50.

- SRP Section 4.2, "Fuel System Design"; Section 4.3, "Nuclear Design"; and Section 4.4, "Thermal and Hydraulic Design," provide guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable General Design Criteria (GDC) specified in Appendix A to 10 CFR Part 50. Specifically, as discussed in SRP Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and AOOs,
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

- SRP Chapter 6, "Engineered Safety Features," provides guidance to the NRC staff for the review of engineered safety features (ESF), which are provided in nuclear plants to mitigate the consequences of design-basis events/accidents or LOCAs, even though the occurrence of these accidents is very unlikely. In particular, SRP Section 6.3, "Emergency Core Cooling Systems," Rev. 3, March 2007 provides guidance to the NRC staff for the review of amendment requests involving the ECCS.

- SRP Chapter 15, "Transient and Accident Analysis" provides guidance to the NRC staff for the review of analyses of the plant's responses to postulated equipment failures or malfunctions for LARs. The review criteria for several specific events are included in this chapter.

- In addition, SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000, provides guidance to the staff for the review of alternative source term amendment requests. SRP 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000. Table 6 provides the dose acceptance criteria for the design-basis accidents (DBAs) at the exclusion area boundary for the maximum TEDE in any 2-hour period, and at the outer boundary of the low population zone during the entire period of the postulated radioactive cloud passage. The NRC staff also considered relevant information in the PSL-2 Updated Final Safety Analysis Report (UFSAR).

Regulatory Guide 1.183 provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. This regulatory guide provides guidance to licensees on acceptable application of alternate source term (AST, also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

Application of M5® Fuel Cladding

The NRC staff previously documented its approval of AREVA topical report BAW-10227P-A, Revision 0, "Evaluation of Advanced Cladding and Structure Material (M5®) in PWR [Pressurized-Water Reactor] Fuel," in a safety evaluation (SE) dated February 4, 2000 (ADAMS Accession No. ML003681490), and concluded that 10 CFR 50.46 and 10 CFR Part 50, Appendix K, criteria are applicable to M5® fuel cladding, subject to compliance with specified burnup conditions. The AREVA topical report BAW-10227P-A, which was submitted to the NRC by letter dated February 11, 2000 (ADAMS Accession No. ML003685828), provided the

justification for the application for M5® cladding material in reactors. BAW-10227P-A, Revision 1, dated June 2003, as noted by letter dated April 19, 2004 (ADAMS Accession No. ML15162B047), is a revision to BAW-10227P-A, Revision 0 and incorporated the portion of the NRC's approval in the NRC SE for BAW-10186(P)(A), Revision 1, Supplement 1, "Extended Burnup Evaluation," dated June 18, 2003 (ADAMS Accession No. ML031700090), in which the applicable restrictions on burnup were removed. Subsequently, in an SE dated May 5, 2004 (ADAMS Accession No. ML041260560), the NRC staff approved topical report BAW-10240P-A, "Incorporation of M5 Properties in Framatome ANP [AREVA] Approved Methods," which further addressed M5® material properties with respect to LOCA applications and included specified conditions.

The specific conditions that address the use of M5® under approved methods that were provided in the SE for BAW-10240P-A are: (1) the corrosion limit, as predicted by the best-estimate model, will remain below 100 microns for all locations of the fuel; (2) all of the conditions listed in the NRC SEs for all AREVA methodologies used for M5® fuel analysis will continue to be met; (3) all AREVA methodologies will be used only within the range for which M5® data was acceptable and for which the verifications discussed in the applicable topical reports were performed; and (4) the burnup limit for implementation of M5® is 62 gigawatt-days per metric ton uranium metal (GWd/MTU). The staff determined that the licensee has satisfied these conditions. As indicated by the licensee, the corrosion limit stated in condition (1) is verified by the licensee for each reload as required by TS 6.9.1.11, "Core Operating Limits Report (COLR)." The licensee indicated that the conditions from NRC-approved SEs stated in condition (2) are incorporated as restrictions in AREVA design procedures and guidelines that will control the core reload designs for PSL-2, which are also verified for each reload as required by the COLR. The licensee also indicated that restrictions on the use of AREVA methodologies stated as condition (3) are also incorporated as restrictions in AREVA design procedures and guidelines that will control the core reload designs for PSL-2, which are also verified for each reload as required by the COLR. Finally, the burnup limit stated in condition (4) is currently part of the AREVA design processes (as stated by the licensee), and is also verified as part of the cycle-specific reload analysis as required by the COLR.

As previously indicated, the licensee submitted an exemption request to allow the use of fuel rods clad with M5® alloy for future reload applications. The NRC granted that exemption in separate correspondence, dated April 19, 2016.

Classification of Plant Conditions in the UFSAR

Section 15.0.6 of the PSL-2 UFSAR, "Classification of Plant Conditions," states:

> The American Nuclear Society (ANS) classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public.

The four categories are defined as follows:

A. Condition I: Normal operation and operational transients
B. Condition II: Faults of moderate frequency

C. Condition III: Infrequent faults
D. Condition IV: Limiting faults

The following is also stated in PSL-2 Final Safety Analysis Report (FSAR) Section 15.0.6:

> The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Functioning of the reactor trip system and engineered safeguards is assumed to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

3.0 TECHNICAL EVALUATION

3.1 Nuclear Performance and Code Review

The NRC staff's review of nuclear performance and codes included evaluations of the mechanical, nuclear, seismic, and thermal hydraulic designs, as well as evaluations in regard to the accident and transient analyses, for the proposed transition to AREVA fuel.

3.1.1 Introduction

FPL is proposing a fuel transition from the Westinghouse CE 16x16 fuel design to an AREVA CE 16x16 fuel design at PSL-2, starting in Cycle 23. The CE 16x16 fuel design is a HTP™ design that has M5® fuel rod cladding; Zircaloy-4 MONOBLOC corner guide tubes; alloy 718 high mechanical performance (HMP™) spaces at the lowermost axial elevation, with Zircaloy-4 HTP™ spacers in all other axial elevations; a FUELGUARD lower tie plate (LTP); and the AREVA reconstitutable upper tie plate (UTP). The HTP™/HMP™ spacer grids are resistant to flow-induced grid-to-rod fretting failures. The FUELGUARD LTP is effective at protecting the fuel from debris in the RCS. The M5® cladding material has proven to have very low oxidation and hydrogen pickup rates (References 8 and 9).

3.1.2 Fuel Mechanical Design

The proposed AREVA fuel assembly design for PSL-2 is a CE 16x16 lattice design that contains 236 fuel rods, four corner guide tubes, and one center instrument guide tube. The fuel rods are positioned within the fuel assembly by ten spacer grids that are attached to the guide tubes.

The M5 cladding material provides substantial margin for end-of-life (EOL) corrosion and hydrogen content since this material has very low corrosion and hydrogen pickup rates. The M5® material was accepted for fuel rod cladding by the NRC staff (References 11 and 12). The fuel rod design includes uranium dioxide fuel rods with axial blankets of lower enriched uranium dioxide. Multiple uranium-235 enrichments are used within an assembly. The CE 16x16 fuel design for PSL-2 is similar and has the same design features as the AREVA CE 14x14 HTP™ fuel design operating in the St. Lucie Plant, Unit No. 1 (PSL-1).

The LTP design is the FUELGUARD structure that uses curved vanes to provide non-line of sight flow paths for incoming coolant to protect the fuel assembly from debris. This design is efficient in preventing debris, including small pieces of wire from reaching the fuel. The UTP design configuration is the same as that in CE 14x14 plants. The reaction plate has been modified to match the interface conditions with the fuel handling grapples consistent with the co-resident fuel. This design allows the reaction plate to be depressed to a setting well beyond the EOL deflections. This design has been used in the United States since 1982, and the reconstitution capabilities of the AREVA CE designs have been successfully demonstrated in CE 14x14 and CE 16x16 fuel examinations. The reconstitutable design uses the corner locking nuts to engage with the upper sleeves of the corner guide tube.

The fuel assembly cage (skeleton) uses four Zircaloy-4 MONOBLOC corner guide tubes, one Zircaloy-4 center guide tube, nine Zircaloy-4 HTP™ spacers, and one 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. The HMP™ spacer cannot be welded directly to the guide tubes since the spacer is made of alloy 718. The AREVA HTP™/HMP™ configuration has been successful in preventing flow-induced grid-to-rod fretting fuel failures on the core periphery.

The HTP™ design, as shown in Figure 1-4 of Reference 9, and its spring structure, provides a flow path at an angle relative to the rod longitudinal direction and causes the water to swirl around the rod without creating a large pressure drop across the spacer, thereby improving heat transfer from fuel rod to the coolant.

3.1.3 Mechanical Compatibility of AREVA CE 16x16 with Co-resident Fuel

The NRC staff requested additional information on details of the licensee's compatibility analysis during the transition from a mixed core type (Westinghouse/CE 16x16 and AREVA CE 16x16 fuel designs) to a core with only AREVA CE 16x16 HTP™ fuel (References 6 and 13). AREVA performed mechanical compatibility evaluations to assure acceptable alignment with the PSL-2 reactor core internals, fuel handling equipment, fuel storage racks, and co-resident fuel. This section will provide a summary of the compatibility evaluations for the core internals.

The compatibility evaluation performed by AREVA and FPL was to ensure compatibility with the new fuel grapple and a prototypic fuel assembly (containing tungsten carbide pellets to provide the necessary weight) has been received by FPL and is available for testing to assure compatibility with the fuel handling equipment and consistency with the plant equipment, processes, and procedures. Table 2-1 of Reference 10 compares major dimensions of the fuel rods, fuel assembly, and few core internals for PSL-2 AREVA fuel design, PSL-2 Westinghouse fuel design, and the PSL-1 AREVA design. The staff reviewed the table and found that the mechanical design features listed in the table are compatible.

Compatibility evaluations were mainly for the UTP, since it interfaces with the fuel alignment plate (FAP), control element assemblies (CEAs), and fuel handling grapples. The interface

compatibility evaluations have to ensure that (1) the posts will insert into the FAP, (2) the posts remain engaged at beginning-of-life (BOL) hot conditions when the core barrel thermal expansion will increase the core plate separation relative to the fuel assembly, (3) the reaction plate will remain engaged with the FAP at BOL hot conditions when the core barrel thermal expansion will increase the core plate separation relative to the fuel assembly, and (4) the posts will not bottom out in the FAP at EOL cold conditions when the fuel assembly has irradiation growth and the differential thermal expansion is not beneficial. Figure 2-1 of Reference 13 illustrates the interface of the UTP with the FAP. The licensee and AREVA have verified that the positions of the posts are the same as the position of the CEA rods, the size of the UTP holes in the posts will accommodate the CEA rods, and the UTP contact area and strength will accommodate the CEA impact and support the CEA at the proper full insertion elevation. The NRC staff has determined that all the interfaces between the FAP and CEAs, and the fuel handling grapples have sufficient clearance margins.

The four alignment pins for lower core support plate are inserted into the holes in the LTP to properly position the fuel assembly in the core. The coolant inlet flow is through holes in the lower core support plate in positions between the alignment pins. NRC staff accepts the LTP compatibility assessment, which verified that the LTP could accommodate the alignment pins, including the diameter, pin height, and position with sufficient clearance.

AREVA guide tubes have the same inner diameter (ID) and same number of weep holes at approximately the same elevations as those of the co-resident fuel. The insertion length of the CEA rods with respect to the available guide tube length demonstrated there is more margin than with the current design for the CEA rod insertion. Since the center tube does not have a dashpot, it has a uniform ID for the entire length of the tube. The nominal ID of the AREVA design is the same as the co-resident fuel and thereby capable of accommodating the CEA center rod, and in different locations, the in-core detectors. The number of weep holes in the AREVA design and co-resident fuel design are the same. NRC staff has determined that there is mechanical compatibility between the AREVA fuel design and the co-resident fuel in the transition cycles for PSL-2 core.

The licensee has compared the LTP elevation, fuel rod elevations, spacer grid elevations, UTP bottom surface elevation, and component envelopes for the AREVA fuel assembly and co-resident Westinghouse fuel assembly (References 6 and 13). Listed below are the conclusions from this evaluation:

- The AREVA-designed LTP has the same elevations as the co-resident fuel. This enables the platform for the attachment of the guide tubes to have a resting surface for the fuel rods after the spacer grid springs relax due to thermal and irradiation effects.

- The nominal fuel column lengths for the AREVA and co-resident designs are the same. Due to the differences in fuel assembly fabrication processes, the AREVA design has a gap between the bottom of the end cap and the top surface of the LTP of about 0.12 inch. This aspect makes the fuel column initially elevated in the AREVA design relative to the co-resident design at BOL by this gap. The impact of this gap is

insignificant since the minimum node length in the neutronic calculation is, at a minimum, 10 times greater than this gap.

- The centerline elevations of the spacer grids, with the exception of the bottom-most and top-most spacers) relative to lower core support plate, align within 0.005 inch. The AREVA grids have different grid heights than the co-resident design. At BOL, these differences in grid height assure that there is 100 percent overlap between the grids in adjacent fuel assemblies, including the top and bottom most spacers. The overlap between the spacer grids for adjacent fresh and EOL discharged assemblies is evaluated since the fresh fuel with no irradiation growth may be placed next to irradiated fuel during subsequent reloads.

3.1.4 Mechanical Design Evaluations

Fuel mechanical design evaluations are performed by using NRC-approved methods (Reference 14). Additional evaluations were performed to address the impact of thermal conductivity degradation (TCD) with fuel burnup using the NRC-approved methods (References 15 and 16). The fuel rod analyses include evaluations of the SAFDLs such as internal rod pressure, cladding creep collapse, cladding fatigue, corrosion, and other pertinent fuel mechanical design parameters listed in Table 2-2 of Reference 9. These evaluations are very dependent on the rod power. For the transition cycles analyzed for this LAR, the power histories were created using expected typical cycle core designs projected to the design life of the fuel. The transition cycles are analyzed to demonstrate that the fuel design is acceptable and provide typical results showing SAFDL compliance. NRC staff has determined that the mechanical design evaluations performed by the licensee are adequate to show SAFDL compliance.

The NRC has identified the lack of TCD with fuel burnup in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," and its Supplement 1. Information Notice 2009-23 identifies the TCD issues that were not addressed in generic legacy methodology described in Reference 14. This information notice identified the non-conservative impact of the thermal conductivity degradation of the fuel pellets with irradiation. The reduction in thermal conductivity results in higher pellet temperatures and results in a reduction in margins to various SAFDLs. In order to account for the TCD effect, AREVA developed correction factors in its legacy codes (RODEX2). These correction factors conservatively penalize the resulting margins for the affected SAFDLs to account for the thermal conductivity degradation (Reference 15).

In response to an NRC staff request for additional information (RAI) to provide details of the RODEX2 regarding its application to non-LOCA and LOCA analyses (Reference 13), the licensee has stated that the legacy code (RODEX2) was replaced with COPERNIC (Reference 35) for generating thermal conductivity, heat capacity, and fuel-to-clad gap coefficient input for the average core and hot spot models. This change enabled explicit accounting of the TCD effect in non-LOCA, LOCA, and thermal-hydraulics analyses.

Table 2-2 of Reference 9 lists the description, criteria, and results from fuel mechanical design evaluations performed by the licensee and AREVA.

The NRC staff has reviewed the licensee's fuel mechanical design evaluations for fuel rod criteria, stress and strain limits, fuel system criteria, axial irradiation growth, and fuel coolability, and determined that the AREVA CE 16x16 fuel design is mechanically compatible with the co-resident fuel design. All of the fuel design criteria have been found to be met up to the fuel rod burnup limit of 62 GWd/MTU under normal and AOO conditions.

3.1.5 Seismic Evaluations

The NRC staff had several communications with the licensee and AREVA regarding the seismic evaluations of AREVA CE 16x16 fuel design and the seismic evaluations for the mixed core at PSL-2 during the transition cycles (References 2, 6, and 8).

In an RAI, the NRC staff requested that the licensee justify the application of the legacy methodology (Reference 17) for the seismic analysis of AREVA CE 16x16 HTP™ fuel design, since this methodology was originally developed for Mark-C fuel assembly LOCA-seismic analysis. The licensee responded to the RAI and stated that although Reference 17 references Mark C fuel design in the title, it 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 for BAW-10133(P)(A), Rev. 1, in which it is noted that the methodology is acceptable for "Mark C fuel design and similar designs." The methodology was modified in Addendum 1 approved for generic use and further modified in Addendum 2, which introduced damping values and is justified for all AREVA PWR fuel designs based on the supporting test data. It should be emphasized that the CE 16x16 HTP™ design for PSL-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. Table 3.1 of Reference 2 lists key characterization data for CE 16x16 HTP™ design, comparing the data with other similar designs (CE 14x14 HTP™, Mark-C, Advanced Mark-BW, and Mark-B HTP™ designs). The staff has determined that the CE 16x16 design is comparable to the design characteristics of the other designs listed in Table 3.1 of Reference 2.

The finite element code CASAC is used for the lateral seismic analysis of full assembly test data to benchmark the bundle design. 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, the safe shutdown earthquake (SSE), and LOCA events. Each event was evaluated independently with lateral and vertical models. Additional testing and evaluations were included in the analyses to address the NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," which identified the concern about the impact of the change in behavior of the assembly and assembly components during their operational lifetime (see Section 3.1.5.3 of this SE). 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.

A conservative modification to the [[

]] Based on its review of the information provided in the LAR and additional information included in the RAI response from the licensee and AREVA on the additional calculations and modifications discussed in the above paragraphs, the NRC staff has accepted this conclusion.

The guide tube stresses and margins were also [[ ]] [[ ]].

[[

]] Based on its review of the additional information provided to the NRC staff on the [[ ]] the NRC staff has determined that this approach is acceptable.

Table 3.2 of Reference 2 lists all the tests performed in support of the LOCA seismic analysis using grids, end fittings, and bundles (BOL and EOL) that are prototypical of the PSL-2 CE 16x16 HTP™ design.

3.1.5.1 Descriptions of Models

In response to an NRC staff RAI, the licensee and AREVA provided details of the [[ ]]. This [[ ]] approach is a

modification to the method described in BAW-10133(P)(A), in which a [[ ]] is described. [[

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A [[ ]] consists of two columns representing the guide tubes and the fuel rods and models them as linear translational springs. All nodal degrees of freedom in the lateral and rotational directions are constrained, and hence, no extraneous reaction forces or moments are produced. A third column is added to the model to represent an instrument tube. The hold-down spring force is accounted for by using a linear translational spring element. In addition, the nonlinear capabilities of the model consist of a number of gap-springs, gap dampers, and slider elements, as shown in Figure 2-7 of Reference 18.

This model has the following elements:

- The stiffness of the non-linear gap-spring-damper element representing the LTP is used to account for the stiffness of the load-path between the guide tube connections and the lower core plate for the BOL case and from the LTP upper face and the lower core plate for the EOL case. This element has damping capability, which is necessary to accurately capture the fuel assembly rebound height in the case of an impact.

- The non-linear gap-spring-damper element between the fuel rod lower end and the top face of the LTP is a function of the fuel design and BOL or EOL condition. In the BOL condition, the gap is open for the CE 16x16 HTP™ design. In the EOL condition, the gap is closed due to subsequent seating of the rods on the upper face of the LTP, and the impact load is carried through the stiffness of the LTP grillage.

- The non-linear slider elements between the fuel rod nodes and the corresponding spacer grid nodes on the guide tube column are characterized by stiffness (or slope) and a saturation force at which the fuel rods begin to slip within the spacer grids (i.e., the grid slip load).

The input parameters for all of the above elements are benchmarked using results from axial drop tests for the first two and using the results from the slip loads measured from test loads obtained from tests and using the results of axial stiffness tests for the third element, with all tests performed on the PSL-2 CE 16x16 fuel assembly.

The vertical load analysis for a single fuel assembly model that applies the seismic or LOCA loads in the vertical direction calculates the axial loads primarily arising from fuel assembly

impacts with the upper and lower core plate during seismic and LOCA excitations. Table 2-3 of Reference 19 lists the maximum vertical impact loads for seismic and LOCA loads. These loads are combined with the loads from the normal operating and horizontal load analysis listed in Table 2-3 of Reference 9 and are evaluated using the ASME-derived limits (See Section 3.1.5.2 of the SE). The NRC staff has determined that there is sufficient margin for the impact loads to the ASME-derived limits.

3.1.5.2 Seismic Evaluation for Mixed Core at PSL-2 Transition

The NRC staff requested additional information regarding the licensee's methodology and results of the limiting loads and deflections analysis for a wide range mixed core configurations during the fuel transition to AREVA CE 16x16 fuel at PSL-2. The analyses performed by AREVA evaluate its fuel against safety criteria required for licensing. The analyses consider scenarios in which the core is fully loaded with AREVA fuel and also transition (or mixed) cores as AREVA fuel is introduced. Six row models representing different row lengths in the PSL-2 core were evaluated with mixed configurations of both AREVA and co-resident fuel. For mixed core configurations, limiting load conditions for either the AREVA or co-resident fuel **[[**

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For PSL-2 analyses, a total of **[[** **]]** fuel assembly patterns were analyzed as shown in Figures 2-1 through 2-6 of Reference 19. Each row model was subjected to the full set of seismic and LOCA loadings required for PSL-2 and the analyses were based on the methodology discussed in Reference 17. The maximum grid impact forces on the AREVA fuel for both LOCA and seismic accidents occur **[[**
]] Table 2-1 of Reference 19 shows the loads for all row configurations are analyzed. Results of the analyses show that there are significant margins for the limiting seismic and LOCA loadings in the full core AREVA fuel and a reduced margin for the mixed core configurations. The reduced margin for the mixed core configuration is due to additional conservatisms included in the mixed core cases to assure uniform treatment of different assembly types. Specifically, all fuel assembly damping values for the mixed core cases were conservatively set to match the co-resident fuel. These reduced damping values for the AREVA fuel are very restrictive and are less than half of the damping that is approved for use in BAW-10133P-01, Addendum 2 (Reference 17).

In order to support the evaluation of co-resident fuel, AREVA first analyzed row models consisting entirely of co-resident fuel that provided a baseline against which to provide comparison. Similar to AREVA fuel, **[[**

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Westinghouse performed an evaluation for its fuel (co-resident) assemblies under the seismic/LOCA loadings associated with mixed core configurations of AREVA and Westinghouse fuel, using the above load increase, and demonstrated that the Westinghouse fuel assemblies would continue to satisfy the applicable seismic/LOCA design criteria consistent with the licensed methodology of CENPD-178 and its associated safety evaluation report (Reference 20).

The NRC staff has reviewed the methodology and results of seismic and LOCA evaluations for the AREVA and Westinghouse fuel assemblies and has determined that the fuel assemblies meet the design limits for both mixed core and full core conditions.

3.1.5.3 Additional Testing to Address Information Notice 2012-09

NRC Information Notice 2012-09 identifies a concern about the impact of the change in behavior of the assembly and assembly components during the operational lifetime. This information notice suggests that in order to demonstrate compliance with GDC 27 of 10 CFR Part 50, Appendix A, spacer grid strength, loading, and deformation must be evaluated to determine that the guide tubes (or fuel skeleton structure) remain sufficiently straight so as not to impede control rod insertion following a seismic event and/or LOCA. In response to the suggestions in the information notice, the licensee and AREVA have performed additional testing and evaluations. Characterization tests (free and forced vibration) were done on two PSL-2 specific full scale assemblies – one non-irradiated (or BOL condition) and the other irradiated (or EOL condition). Also, characterization tests were performed on individual spacer grids of both non-irradiated and simulated irradiated conditions.

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]] The tests performed on the simulated irradiated assembly had a **[[**
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Effects of irradiation spacer grid impact testing was simulated in the spacer grid impact testing according to Addendum 1 of Reference 17 for both non-irradiated and simulated irradiated configurations. This testing was done with **[[** **]]** to simulate the effects of irradiation. These modifications were made to simulate **[[**

]]. Based on the testing conducted in a hot cell on actual irradiated grids, it has been demonstrated that there are two primary effects that exhibit a significant effect on the grid dynamic characteristics: **[[**

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The testing protocol has been shown to demonstrate a good conservative agreement with the results of the tests performed in the hot cell.

The NRC staff has reviewed the additional testing of irradiated and non-irradiated grid straps and found that the licensee has appropriately responded to the concerns raised in NRC Information Notice 2012-009.

3.1.6 Nuclear Design

The purpose of the nuclear design analysis is to verify that the cycle-specific reload design and the key safety parameters are properly addressed in the reload analysis. The impact of the transition from Westinghouse CE 16x16 fuel to AREVA CE HTP™ 16x16 fuel on the nuclear design bases and methodologies for PSL-2 is evaluated in this section.

The nuclear design methodologies are described in NRC-approved References 21 through 23. Major safety parameters that are biased in the safety analysis and calculated as part of the neutronic design analysis are listed in Table 3-1 of Reference 9. These parameters include moderator temperature coefficient (MTC), Doppler temperature coefficient (DTC), shutdown margin, LHGR, maximum ejected rod total power peaking factor F_Q, delayed neutron fraction, and total deposited enthalpy.

The nuclear design analysis used the methodology as described in NRC-approved Reference 21. The effect of extended burnup on nuclear design parameters for AREVA fuel has been approved in Reference 24. A representative first transition core and two representative follow-on core designs for PSL-2 fuel transition have been developed by the licensee and AREVA. The loading patterns are depleted at a core power of 3,020 megawatts thermal (MWt). The first transition cycle contains fresh AREVA St. Lucie Plant CE 16x16 HTP™ fuel with once-burnt and twice-burnt Westinghouse CE 16x16 fuel. The second transition cycle contains fresh and once-burnt AREVA fuel and twice burnt CE/Westinghouse fuel. The third cycle contains only AREVA fuel. Table 3.1 of Reference 9 lists key information from the core design analyses and is repeated below.

Table 3.1.6-1: 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

Key:
- EFPD: Effective Full Power Days
- HFP: Hot Full Power
- ARO: All Rods Out
- F_r^T: Assembly Radial Peaking Factor
- F_Q: Total Power Peaking Factor

The variations in F_r^T and F_Q are typical of normal cycle-to-cycle variations as the fuel loading pattern changes between cycles. Compliance with the TS peaking factors will be assured by the licensee for each cycle.

The staff has verified the methodology and the results from nuclear design analysis during the transition from Westinghouse/CE fuel to AREVA fuel and confirmed that the peaking factors and key safety parameters shall be maintained within their specified limits using only AREVA methodologies and codes. The key safety parameters and peaking factor limits will be verified on a cycle-specific basis using the Westinghouse nuclear design methodology and/or the AREVA methodologies and codes, as described in the COLR, similar to the current PSL-1 analysis.

3.1.7 Thermal and Hydraulic Design

The thermal-hydraulic analysis of AREVA CE 16x16 HTP™ fuel is performed using approved methodologies for departure from nucleate boiling (DNB) calculations (References 25 and 16). The S-RELAP5 code was used in the transient reactor analysis, and the XCOBRA-IIIC code was used to calculate the minimum departure from nucleate boiling ratio (MDNBR) using the HTP™ and Biasi critical heat flux (CHF) correlations. RODEX2-2A was developed to perform calculations for fuel rods under normal operating conditions (References 27 and 28). RODEX2-2A was used to establish the fuel centerline melt (FCM) LHGR 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 nozzle effects. The Biasi CHF correlation (Reference 26) is used to calculate the DNBR for post-scram reactor conditions. Using a 95/95 statistical approach, AREVA determined a safety limit DNBR of 1.141 for HTP™ CHF correlation (Reference 29) and **[[ ]]** for Biasi correlation. **[[**

]] For mixed cores, AREVA applies a 2 percent penalty to DNBR safety limits as required by the safety evaluation report in Reference 26.

The NRC-approved methodology for evaluating DNB behavior in a mixed core is described in Reference 26 and uses the steady-state XCOBRA-IIIC code model with core boundary conditions at the time of MDNBR from the S-RELAP5 transient reactor analysis. Statistical DNBR analysis is performed using the methodology described in Reference 30 subject to two conditions: (1) the methodology is approved only for CE reactors with a protection system as described in Reference 30, and (2) although the statistical analysis methodology uses specific variables, if additional variables are needed, AREVA will reevaluate the methodology and document the changes in the variables.

Protection against FCM SAFDL is expressed as a limit on LHGR allowed. This FCM limit is explicitly calculated for the AREVA fuel transition. Since the gadolinia fuel rods have reduced thermal conductivity, the FCM limit may be set by the gadolinia fuel.

The NRC staff requested that the licensee justify that the legacy methodologies (References 11 and 12) are appropriately applicable to the AREVA CE 16x16 fuel design in analyzing rod bow impact on thermal margin analysis (Reference 6). The original rod bow methodology (XN-75-32, Supplements 1, 2, 3, and 4) was revised when the burnup limits were increased to the currently approved 62 MWd/KgU peak rod limit in Reference 24. The NRC staff concurred that the applicability of the rod bow correlation and resulting DNB and LHGR penalties would be a conservative application over predicting the rod bow, and thus, resulting in a conservative rod bow penalty. The NRC staff reviewed the applicability of this correlation when M5 cladding material implementation report (Reference 11) was submitted. The staff concluded that the "growth characteristics of M5 will not have a detrimental effect on rod-bow." Subsequently, AREVA has completed a lead assembly program for a CE 16x16 fuel design. In this program, the fuel rods were M5® clad, had a 0.382 inch rod OD, and had a rod pitch of 0.506 inch (the same dimensions as the PSL-2 fuel). Measurements and inspections on this test assembly after each cycle and after EOL discharge showed no noticeable rod distortion in the visual inspections. This performance was well within the correlation predictions. The NRC staff has determined that there is continued validity and conservatism of the rod bow methodology confirmed for the CE 16x16 design with M5® cladding.

3.1.8 Thermal Hydraulic Compatibility

NRC staff requested additional information for the methodology and results of mechanical and thermal compatibility of the AREVA fuel with the co-resident fuel and core internals. Mechanical compatibility is discussed in Section 3.1.3 of this SE. This section discusses the methodology and results for the thermal hydraulic compatibility of AREVA fuel with the co-resident fuel.

AREVA performed an analysis to evaluate the pressure drops associated with the AREVA and Westinghouse fuel designs during the transition. The Westinghouse fuel assemblies have a lower overall resistance (lower pressure drop) to coolant 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. This difference is small and within previous AREVA transition experience. This results in the DNB-limiting AREVA HTP™ assembly receiving less flow in mixed core analyses relative to a full core of AREVA HTP™ fuel. This difference in flow is explicitly included in the thermal-hydraulic analyses. Pressure drop tests were conducted for both the AREVA PSL-2 fuel assembly and for a co-resident Westinghouse fuel assembly at the same AREVA test facility. Tables 2-4 and 2-5 of Reference 6 show that the calculated relative flow changes under normal conditions through resident Westinghouse fuel and AREVA fuel during transition cycles compared well to a full core of the respective fuel types.

Mixed core DNB performance is evaluated using XCOBRA-IIIC at a power level to achieve MDNBR. As more AREVA HTP™ fuel assemblies are inserted into the core in subsequent reloads, less flow will be diverted into the lower pressure drop Westinghouse assemblies; increased flow in the AREVA HTP™ assemblies will result in an increase in DNB performance. Transition DNB analyses were performed using an XCOBRA-IIIC model representative of a single batch of AREVA HTP™ fuel (88 AREVA HTP™ assemblies), which will bound the DNB

performance of a full core of AREVA HTP™ fuel. In addition to modeling mixed core explicitly, the NRC-approved 2 percent penalty is applied to the AREVA HTP™ correlation.

The Westinghouse co-resident fuel is expected to be at least 5 percent lower in power peaking and receive increased RCS flow due to slightly lower pressure drop as compared to the AREVA HTP™ fuel. This combination of lower peaking and higher flow will make the co-resident Westinghouse fuel non-limiting for DNB during the transition cycles. This will be verified every cycle as part of the reload process to confirm that the Westinghouse fuel continues to meet the respective fuel design limits during the transition cycles using applicable Westinghouse methodology.

The NRC staff has reviewed the thermal-hydraulics compatibility analysis for PSL-2 transition cycles with respect to the pressure drops in regards to the AREVA HTP™ and the co-resident fuel, as well as the DNBR performance, and has determined that the thermal-hydraulics and thermal margin performance are acceptable.

3.1.9 Accident and Transient Analyses

This section provides a brief description of methodology, input parameters, and results from a few PSL-2 UFSAR Chapter 15 events affected by the fuel transition. This section discusses transient analysis listed below:

- Pre-Trip Steam System Piping Failure (Mail Steam Line Break (MSLB) – Pre-Trip) (UFSAR Section 15.1.5)
- Post-Trip Steam System Piping Failure (MSLB-Post-Trip) (UFSAR Section 15.1.6)
- Uncontrolled CEA Bank Withdrawal (CEAW) from a Subcritical or Low Power Startup Condition (UFSAR Section 15.4.1)
- Uncontrolled CEAW at Power (UFSAR Section 15.4.2)
- Spectrum of CEA Ejection Accidents (UFSAR Section 15.4.8)
- Small Break Loss-of-Coolant Accident (SBLOCA) (UFSAR Section 15.6.5.3)

3.1.9.1 Main Steam Line Break (Pre-Trip)

SRP (NUREG-0800) Section 15.1.5 provides the criteria for the review of the licensee's safety analysis for the main steamline break (MSLB) accident. The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow, which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the RCS and results in a reduction of coolant temperature and pressure. The negative moderator temperature coefficient and the cooldown of the reactor system causes an increase in core reactivity. The core reactivity increase may cause a loss of reactor core shutdown margin and a resulting increase in reactor power. If the plant is at power, the reactor is automatically tripped, and the main steam and feedwater line isolation valves are automatically closed. Decay heat is removed as necessary through the unaffected steam generators (SGs) by venting steam from the secondary system safety and relief valves.

The pre-scram portion of the MSLB analysis is concerned with the behavior prior to and just after reactor scram where there is a potential for fuel failure to occur. The analysis is terminated shortly after a scram occurs. The pre-scram phase of the accident can challenge the DNB and FCM limits due to potential increase in power and the variable high power (VHP) trip power input decalibration for both the ex-core detector and the ΔT power signals. This event is analyzed for a spectrum of break sizes from 0.1 ft^2 to 6.8533 ft^2 and MTC values from -8 percent mil (pcm)/degrees Fahrenheit (°F) to -33 pcm/°F. Three scenarios for a break location are used: (1) a break of a steamline inside containment (asymmetric inside break), (2) a break of a steamline outside containment but upstream of the common turbine header (asymmetric outside break), and (3) a break located at the common steamline header at the turbine inlet (symmetric break). The detailed description of this analysis is described in Section 4.5.1 of Reference 31.

The major input parameters are initial operating conditions, reactivity feedbacks, reactor protection system (RPS) trips and delays, gap conductance and fuel thermal properties, SG tube plugging percentage, and single volume containment (for break inside containment).

The acceptance criteria for the pre-scram MSLB accident are:

- RCS pressure and main steam system should be maintained below 110 percent of design values;
- Fuel cladding integrity shall be maintained by ensuring that the MDNBR remains above the 95/95 DNBR limit for PWRs; and
- No more than a small fraction of the fuel elements are damaged, and the offsite release of the radioactive material complies with the requirements of 10 CFR 50.67.

The MSLB accident analyses were performed using approved methodology per Reference 25 using the S-RELAP5 code for modelling the reactor system and the XCOBRA-IIIC for calculating the MDNBR using the HTP™ CHF correlation. Cases at end-of-cycle (EOC) HFP initial conditions at maximum TS core inlet temperature and minimum TS RCS flow rate were analyzed.

Results for each break location are listed in Table 4.8, and the sequence of events for the limiting case is given in Table 4.7 of Reference 31. Figures 4.41 through 4.47 of Reference 31 provide the plots for key system parameters for the overall limiting case.

The NRC staff reviewed the methodology, input parameters, and results for the PSL-2 pre-trip MSLB and found that the results have satisfied the acceptance criteria for the accident analysis.

3.1.9.2 Main Steam Line Break (Post-Trip)

For the post-scram analysis, the MSLB event is initiated by a postulated break in a main steamline coincident with reactor scram. The SG pressures and temperatures will decrease rapidly following the initiating event, and the SG pressures drop will initiate a main steam isolation signal, which will initiate as the main steam isolation valves (MSIVs) close and terminate the blowdown from the SG with the intact main steam line. The cooldown of the RCS will insert positive reactivity from both moderator and fuel temperature reactivity feedbacks,

particularly at EOC conditions with a most-negative MTC. The analysis assumes that the most reactive control rod is stuck out of the core, and as a result, the radial neutron flux and the flux distribution will be highly peaked in the stuck rod region and will be cooled primarily with coolant delivered from the cold legs of the affected loop. The event will be terminated by the injection of boron from high pressure safety injection pumps and/or by the dryout of the affected SG, which will stop the RCS cooldown (Reference 31).

The major input parameters for the post-trip MSLB are initial conditions (one set with rated power with a maximum core inlet temperature (CIT) with largest potential cooldown, and a second set of conditions initiated from hot zero power (HZP) with minimum allowed TS shutdown margin), break size and location, break flow, reactivity feedback, gap conductance, SG tube plugging, RCS flow, and MSIV and main feedwater parameters. This event is primarily driven by moderator feedback as a result of the cooldown of the RCS. The MTC analysis value of -33 pcm/°F was modeled, which bounds the most negative TS limit of -32 pcm/°F. Minimum scram worth, appropriate for the assumed initial condition, was assumed. For the post-scram analyses, the most reactive rod was assumed to be stuck out of the core.

The acceptance criteria for post-trip MSLBs are the same as those for the pre-trip MSLB analysis listed in the previous section of this SE.

The MSLB accident analyses were performed using approved methodology per Reference 25 using the S-RELAP5 code for modelling the reactor system to determine neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures), and produce an estimated time of MDNBR. The asymmetry in the core is simulated by dividing the core into a sector adjacent to the affected loop and a sector adjacent to the unaffected loop. The S-RELAP5 core fluid boundary conditions and average rod surface heat flux were then input to the XCOBRA-IIIC code, which was used to calculate the MDNBR using the Biasi correlation. The event was analyzed from both HZP and HFP conditions to assess the potential amount of fuel failure. Offsite power available and loss of offsite power cases were considered. For the loss of offsite power cases, loss of offsite power was assumed at event initiation.

The sequence of events is summarized in Table 4.10 of Reference 31 for the HZP cases. Table 4.11 of Reference 31 summarizes the sequence of events for cases initiated from HFP, with and without offsite power. Table 4.12 of Reference 31 summarizes the results of the analyzed post-trip MSLB. The greatest challenge to the FCM limit occurred for the case initiated from HZP with offsite power available, and the greatest MDNBR challenge occurred for the case initiated from HFP with offsite power available. Transient response of key system parameters for cases initiated from HZP and HFP cases are illustrated in Figures 4.48 through 4.68 of Reference 31.

The NRC staff reviewed the methodology, input parameters, and the results for the PSL-2 post-trip MSLB and found that the results have satisfied the acceptance criteria for the accident analysis.

3.1.9.3 Uncontrolled Control Element Assemblies Bank Withdrawal from a Subcritical or Low Power Startup Condition

SRP (NUREG-0800) Section 15.4.1 provides the criteria for review of the licensee's safety analysis for evaluating the effects and consequences of an uncontrolled CEA withdrawal from a subcritical or low-power condition to assure conformance with the requirements of GDC 10, 17, 20, and 25. A summary description of this analysis is described in Section 4.18 of Reference 31.

This event is initiated by a continuous CEA withdrawal that could result from a malfunction in the reactor regulating system or control element drive system and is initiated from a Mode 2 startup (critical) condition at zero power. The event is characterized by a large and rapid positive reactivity insertion that can challenge the DNB and FCM SAFDLs. Reactor trip occurs on a VHP trip signal; however, the power excursion is mitigated by Doppler reactivity feedback prior to reactor trip.

The major input parameters for the analysis consist of HZP CIT and minimum TS RCS flow rate for four-pump operation, reactivity feedbacks, RPS trips and delays, pressurizer pressure control system parameters, CEA withdrawal characteristics, and gap conductance.

The acceptance criteria for this anticipated operational occurrence (AOO) event are:

- The thermal margin limit, DNBR, is not less than the 95/95 DNB correlation limit[1], and
- Fuel centerline melt limit is precluded in the most adverse location in the core.

The analysis is performed with the approved non-LOCA methodology in Reference 25. For the CEAW event at subcritical state, the S-RELAP5 code was used to model the key system components and calculate neutron power, fuel thermal response, surface heat transport, fluid conditions, and an estimated time of MDNBR. The core fluid boundary conditions and average rod surface heat flux are then input to the XCOBRA-IIIC code, which is used to calculate the MDNBR using the HTP™ CHF correlation.

The sequence of events is listed in Table 4.21 of Reference 31. The event transient responses are plotted in Figures 4.98 through 4.103 of Reference 31. The results with respect to the acceptance criteria are shown in Table 4.22 of Reference 31. The MDNBR is found to be above the 95/95 limit for the HTP™ CHF correlation. The peak cladding temperature (PCT) is calculated to be less than the FCM temperature.

The NRC staff reviewed the methodology, input parameters, and the results for the PSL-2 uncontrolled CEAW at subcritical state and found that the results have satisfied the acceptance criteria for the accident analysis.

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[1] SRP Section 4.4 states, "For departure from nucleate boiling ratio (DNBR), CHFR [critical heat flux ratio] or CPR [critical power ratio] correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs."

3.1.9.4 Uncontrolled Control Element Assemblies Bank Withdrawal at Power

SRP (NUREG-0800) Section 15.4.2 provides the criteria for review of the licensee's safety analysis for evaluating the effects and consequences of an uncontrolled CEA withdrawal at power condition to assure conformance with the requirements of GDC 10, 17, 20, and 25. A summary description of this analysis is described in Section 4.19 of Reference 31.

An inadvertent CEA bank withdrawal at power could be caused by two potential initiators: (1) operator error or (2) a malfunction of either the CEAs or of the control element drive mechanism, which results in an uncontrolled, continuous CEA bank withdrawal. The positive reactivity addition from the CEA withdrawal results in a power transient. A combination of heat extraction from the SG and increase in reactor power can cause reactor power, RCS temperature, and core heat flux to decrease the margin to DNB and FCM SAFDLs and the RCS overpressure limit. However, the RPS is designed to terminate any such transient before thermal margin and RCS overpressure limits are reached.

The major input parameters for the analysis consist of full power, 90 percent rated thermal power (RTP), 65 percent RTP, and 20 percent RTP, maximum CIT and minimum TS RCS flow rate for four-pump operation, reactivity feedbacks, scram worths, RPS trips and delays, pressurizer pressure control system parameters, CEA withdrawal characteristics, and gap conductance.

The acceptance criteria for this AOO event are:

- Fuel cladding integrity should be maintained by ensuring that the SAFDLs are not exceeded, and
- The RCS and main steam pressure should be less than 110 percent of the design values.

The CEAW at power event analyses were performed using the non-LOCA methodology described in Reference 25. Reactor system analysis code S-RELAP5 is used to model the key system components and calculate neutron power, fuel thermal response, surface heat transport, RCS flow rates, temperatures, and pressures, and an estimated time of MDNBR. The RCS boundary conditions and rod heat flux are input to the XCOBRA-IIIC, which is used to calculate the MDNBR. Two reactivity feedback matrices of cases – one for most-positive with most-positive MTC and least-negative DTC, and the other for most-negative with most-negative MTC and most-negative DTC are used in the analyses.

Calculations were performed for full power and power levels using beginning-of-cycle (BOC) and EOC neutron kinetics values. Table 4.24 of Reference 31 lists the peak LHGRs and MDNBR for all the cases analyzed. The peak LHGR does not challenge the LHGR limit for this event. The MDNBR is found to be above the 95/95 limit for the HTP™ CHF correlation. The transient response for the various parameters such as reactor power, rod surface heat flux, pressurizer pressure, pressurizer coolant level, total RCS flow rate, and the reactivity feedback as a function of time are shown in Figures 4.104 through 4.110 in Reference 31.

The NRC staff reviewed the methodology, input parameters, and the results for the PSL-2 uncontrolled CEAW at power and found that the results have satisfied the acceptance criteria for the accident analysis.

3.1.9.5 Spectrum of Control Element Assembly Ejection Accidents

SRP (NUREG-0800) Section 15.4.8 provides the criteria for review of the licensee's safety analysis for evaluating the effects and consequences of a control rod ejection (CRE) accident for potential damage to the RCS pressure boundary and for whether the fuel damage from a CRE accident could impair cooling water flow. A summary description of this analysis is described in Section 4.25 of Reference 31.

The CEA ejection event is initiated by postulated rupture of control rod drive mechanism housing that results in the full system pressure acting on the drive shaft, which ejects its control rod from the core. The resulting rapid positive reactivity insertion and an increase in radial power peaking could potentially result in fuel rod damage. As the fuel begins to heat up, the DTC feedback mitigates the power excursion, and the eventual scram negative reactivity insertion affects the fuel temperature and fuel rod cladding surface heat flux.

The key input parameters are listed in Table 4.31 of Reference 31 and include the initial power levels (full power, 65 percent, and 20 percent RTP and HZP), maximum CIT, TS minimum RCS flow rate, reactivity feedbacks at BOC and EOC, RPS trips and delays, pressurizer pressure control, ejected rod worth, gap conductance, and single failure.

The acceptance criteria for this event are based on the requirements per regulations according to GDC 13, 28, and 67. Acceptance criteria for the event analysis are:

- Fuel failure due to DNB and FCM should be limited and within limits for radiological analysis;
- Reactivity excursions should not result in peak radial average fuel enthalpy of 230 cal/gm for fuel coolability, and for HZP conditions, 150 cal/gm for fuel failure;
- Pellet-cladding mechanical interaction failure criteria is a change in radial average fuel enthalpy greater than the corrosion dependent limit, as shown in Figure B-1, Section 4.2, Appendix B, of the SRP (NUREG-0800); and
- Peak RCS pressure must remain below a value that would have caused the stresses in the RCS to exceed the faulted condition stress limits.

Transient analysis is performed using the S-RELAP5 code to calculate the neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures). The core fluid boundary conditions and rod surface heat flux are input into XCOBRA-IIIC code to calculate the MDNBR using the HTP™ CHF correlation. Deposited enthalpy calculation was performed using methodology described in Reference 32. CRE analysis is performed for eight different initial conditions: BOC and EOC full power, BOC and EOC 65 percent RTP, BOC and EOC 20 percent RTP, and BOC and EOC HZP. All four reactor coolant pumps (RCPs) are assumed to be in operation in both Modes 1 and 2.

Sequence of events for the CRE analysis is listed in Table 4.32 of Reference 31, and the analysis results are given in Tables 4.33 and 4.34 of Reference 31. BOC full power MDNBR provides the least margin of all other cases. Transient response plots are shown in Figures 4.118 through 4.124 for BOC HFP cases in Reference 31. The BOC 20 percent RTP case presented the most significant challenge to the FCM temperature acceptance criteria. The transient response for BOC 20 percent RTP is illustrated in Figures 4.125 through 4.131.

Fuel coolability acceptance criteria per Section 4.2, Appendix B of the SRP (Reference 33) in terms of HZP, HFP, and part power total deposited enthalpy provided in Tables 4.33 and 4.34 have been met. Since the enthalpy rise for the peak nodes is below the acceptance criteria limit, the pellet-cladding mechanical interaction failure criteria is met. Since the fuel design parameters are not key parameters regarding RCS overpressure, the change in fuel design will not affect the overpressure aspect of this event.

Justification/Validity for Applicability of Legacy CRE Methodology XN-NF-78-44

The NRC staff requested supplemental information from the licensee to justify the use of a legacy CRE methodology (Reference 32) for the AREVA CE 16x16 fuel design. The licensee responded to this request for justification in References 2 and 5. The licensee stated that Reference 32 methodology that is used primarily for energy deposition is independent of fuel pin dimensions due to the fact that the Doppler reactivity terminates the prompt critical excursion. This methodology conservatively assumes adiabatic heatup of the pin, and therefore, conduction losses from different sized pins are ignored.

The power excursion during a CRE event that takes place in a very short time can be modelled analytically by the Nordheim-Fuchs model described in Reference 34. Energy released for the adiabatic heatup that is derived in Reference 34, Section 3-4 is only a function of ejected rod worth, effective delayed fraction, and Doppler temperature coefficient. The approved methodology in Reference 26 uses detailed calculations for deposited energy using a similar approach in conjunction with local power peaking to obtain the local energy deposited.

A range of values for rod outer diameter, ejected rod worth, post-ejection peaking factor, DTC, delayed neutron fraction, and total enthalpy (cal/gm) are listed in Table 2.1 of Reference 2 for PSL-2 BOC and EOC and are compared with a range of values of the same parameters for AREVA's recent analyses for CE 14x14, CE 15x15, Westinghouse 16x16, and Westinghouse 17x17 fuel designs. The parameters used in PSL-2 fall within the range of values used in AREVA analyses for various fuel design. Therefore, the Reference 26 methodology is considered appropriate for PSL-2 CRE event analysis for CE 16x16 fuel design.

The licensee and AREVA further presented an alternative method for justification of their application of CRE event analysis methodology in Reference 32 to CE 16x16 fuel design by comparing the results from a different NRC-approved methodology. The licensee provided AREVA the necessary input to support this comparison between the AREVA control rod ejection (CRE) deposited enthalpy results and the CRE deposited enthalpy results determined by Westinghouse using a different NRC-approved CRE methodology. Table 2.1 of Reference 5 compares the deposited enthalpy results from PSL-2 EPU analysis performed by Westinghouse

and the results from AREVA analysis for BOC (HZP and HFP) and EOC (HZP and HFP) cases. This comparison shows that both NRC-approved methods identify the same case (BOC HFP) to be the limiting condition and that the AREVA method produces a conservative result relative to the Westinghouse method for that case.

The method in Reference 26 is used to determine the deposited energy portion of the CRE analysis. EMF-2310(P)(A) (Reference 25) is used to evaluate the challenge to the DNB and FCM fuel failure acceptance criteria using the S-RELAP5 and XCOBRA-IIIC codes. S-RELAP5 calculates the core average transient response, the hot spot fuel centerline temperature, and thermal hydraulic boundary conditions for the subsequent XCOBRA-IIIC MDNBR calculations. To account for the effects of TCD, COPERNIC methodology (Reference 35) is used to generate the fuel thermal conductivity and heat capacity and fuel pellet-to-clad gap coefficient inputs for the average core and hot spot models.

In the S-RELAP5 calculations, key neutronic parameters are conservatively biased to ensure that bounding results are calculated. These parameters are reviewed and updated each reload cycle to verify that the acceptance criteria continue to be met for the CRE event. AREVA's CRE analysis has shown that the PSL-2 fuel transition does not result in any fuel failure for the calculations submitted as part of the LAR, and significant margin is maintained to the fuel failure limits based on DNB and FCM.

Based on the information submitted by the licensee for the CRE event analysis and the review by the NRC staff, the staff has determined that the PSL-2 CRE event analysis is reasonable and therefore acceptable.

3.1.9.6 Small Break Loss-of-Coolant Accident (SBLOCA)

3.1.9.6.1 Background

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCS primary boundary at a rate in excess of the reactor coolant make-up system's ability to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished. The reactor protection system and ECCS are provided to mitigate these accidents. The NRC staff's review covered the (1) licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) sequence of events; (4) analytical model used for analyses and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of PCT, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection system and ECCS; and (7) operator actions.

The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of an LOCA; (3) GDC 4, insofar as it requires that structures, systems, and components important to safety be protected against dynamic effects, including the effects of missiles, pipe whipping, and

discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit; (4) GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC 35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001, "Review Standard for Extended Power Uprates."

The licensee requested changes (Reference 1) to the facility operating license to allow fuel replacement with AREVA 16x16 HTP™ nuclear fuel utilizing M5 cladding. To facilitate this replacement, PSL-2 must demonstrate acceptable safety performance for full spectrum of design-basis transient thermal hydraulic events that establish the acceptable operating limits and conditions for PSL-2. As part of this evaluation of the design-basis events, the NRC staff review consists of only a review of the design basis SBLOCA analysis performed by AREVA that meets the latest recently approved SBLOCA RELAP5 methodology (Reference 2), and as such, PSL-2 must demonstrate continued acceptable ECCS performance for the replacement AREVA nuclear fuel. This evaluation includes a review of the ECCS licensing methods applicable to the evaluation of small break LOCA for PSL-1 and PSL-2 at a power level of 3029.06 MWt (including a 0.3 percent uncertainty) and a peak LHGR of 13.0 kilowatt/foot (kW/ft).

The NRC staff's evaluations of the small break LOCA are also presented in Section 3.1.9.6.2 of this SE.

Method of NRC Staff Review and Background

PSL-2 consist of a four-loop, PWR designed by Combustion Engineering, Inc. and enclosed within a large, dry containment. The ECCS consists of two high pressure safety injection pumps, two low pressure safety injection pumps, and four safety injection tanks. These high and low pressure pumps, including the safety injection tanks, deliver coolant from the refueling water storage tank to the cold legs. The safety injection tanks are pressurized with a nitrogen cover gas with a pressure of 499.7 pound per square inch absolute (psia). Credit for one of three charging pumps, which are actuated on a safety injection actuation signal, was also assumed in the evaluation of the small break LOCA spectrum. The three charging pumps deliver flow to the cold legs.

The NRC staff reviewed the licensee's application for the results of the small break LOCA analyses, as described in the following sections.

3.1.9.6.2 Evaluation of Small Break LOCA Analysis

The staff evaluation consisted of reviewing the results of the licensee's evaluation of the small break LOCA spectrum using the S-RELAP5 code (References 1, 37, and 6) performed at 3029.06 MWt (including a 0.3 percent uncertainty) and a peak LHGR of 13.0 kW/ft for PSL-2.

Small Break LOCA Behavior

The licensee's submittal (References 1, 6, and 37) for small breaks, using the NRC staff-approved S-RELAP5 code, included a detailed break spectrum analysis with break sizes of 0.022 to 0.491 ft^2. The PCT for the limiting break was calculated to be 2057 °F for the 2.7 inch diameter break (0.03976 $ft^{2)}$ cold leg break. In response to the staff's requests to thoroughly evaluate and locate the limiting break in this region of the spectrum, the licensee performed additional small break analyses (2.5, 2.6, 2.7, 2.8, and 3.0 inches in diameter) using very fine break increments in the 2 to 3 inch diameter break range, and found the 2.7 inch diameter break size to be the most limiting. The staff requested a detailed spectral evaluation for breaks in this region because small changes in break area of approximately 0.05 ft^2 in the vicinity of the peak break can result in increases in PCT in excess of 100 °F.

The NRC staff's review of the licensee's small break LOCA analysis revealed the following conservative assumptions pertinent to the analysis of the small break LOCA spectrum:

- The licensee correctly assumed the hot leg nozzle gaps were fully closed during the small break LOCA. This forces the core steaming rate to vent through the higher resistance loop piping, which maximizes the degree of level depression in the core and maximizes the PCT for the limiting break.

- The kinetics model determining core power prior to reactor trip includes the moderator-density feedback effects and employed an MTC which is the most positive for PSL-2, maximizing core power prior to reactor trip for all small breaks.

- Reactor trip occurs on a low pressurizer pressure thermal margin trip set point, which occurs early during the event. Offsite power is also assumed to occur at this time, since with positive MTCs, core power could increase prior to trip, maximizing primary system pressure, decreasing ECCS injection, and maximizing PCT for all small breaks.

- The upper core barrel leakage paths from the upper plenum/upper head through the core barrel flange to the downcomer was also omitted in the S-RELAP5 mode. This allowed core generated steam to again vent through the external loop piping to the break in the cold leg, maximizing core uncovery and PCT for small breaks.

- The safety injection tank temperature was assumed to be at the TS limiting value of 104 °F.

- Only one charging pump was credited in the analysis, with an assumed maximum injection rate of 35 gallons per minute (gpm). Note that the charging pump injection of about 44 gpm into the RCS represents an injection rate that includes allowance for reactor coolant pump seal leakage. Since a minimum of two charging pumps will be activated on a safety injection actuation signal, credit for only one pump is considered conservative for this evaluation.

- The analysis also employed an axial power distribution that includes an axial peak located highest in the core that meets the axial shape index TS limit for a top peaked power shape.

- Loop seal clearance for break sizes **[[ ]]** is prevented in the S-RELAP5 analysis. This assumption maximizes the loop resistance and thereby maximizes core uncovery and the PCT for the limiting small breaks.

- A **[[ ]]** to preclude artificial top down cooling of the hot assembly and hot rod.

- The S-RELAP5 counter-current flow limitation was applied to the flow paths representing the **[[ ]]** This precludes non-physical draining of the liquid **[[ ]]** during the event.

- The high pressure safety injection head flow curve used in the analysis accounts for the inservice testing (IST) degradation and uncertainties in the head and flow surveillance measurements, plus 1 percent Emergency Diesel Generator under-frequency effects. The maximum injection line flow is also assumed to deliver to the broken cold leg.

It is also noted that the limiting break two-phase level on the secondary side of the SGs exposed a portion of the active tubes in the U-bend region. The staff further notes that at least one SG is needed for heat removal and control of RCS pressure for break sizes less than about 4 inches in diameter. There are three auxiliary feedwater pumps, two motor-driven and one turbine-driven. The S-RELAP5 model assumption of the disabling of a motor-driven auxiliary feedwater pump due to the single failure criterion leaves one motor-driven pump and the turbine-driven pump available for secondary heat removal. The turbine driven pump is conservatively not credited in this analysis.

The NRC staff's calculations showed that the short-term exposure of the active tube bundle that resulted with the bounding auxiliary feedwater flow assumption was still insufficient to affect RCS pressure during the pressure plateau period to cause an earlier core uncovery and higher PCT than that calculated for the limiting 2.7 inch break.

Based on the detailed break size increment evaluations of small breaks in the 2 to 4 inch diameter range, and the supporting conservative assumptions listed and described above as applied to the S-RELAP5 analysis of PSL-2, the staff finds this analysis to be conservative and acceptable.

Severed Injection Line

The licensee also performed an analysis of a severed emergency core cooling injection line. This analysis could potentially represent a more limiting small break LOCA size, since the severed injection line results in more than 25 percent of the broken emergency core cooling injection line (one of four injection ports) spilling to containment. Likewise, this condition results

in much less than 75 percent injection flow into the RCS through the other three intact injection lines, since with the broken injection line pumping against atmospheric pressure and the intact lines feeding an RCS pressure much higher, more flow is pumped through the broken emergency core cooling line.

Based on these considerations, the licensee performed an analysis of a severed injection line, which is a 10.126 inch diameter break on the top of the discharge leg. The PCT was calculated to be 1451 °F, which is less than the PCT for the limiting 2.7 inch diameter break size (0.03976 ft^2 discussed above).

In the analysis, the licensee again credited only minimum flow from the charging, High Pressure Safety Injection (HPSI), and Low Pressure Safety Injection (LPSI) pumps, including safety injection tank discharge to preclude this break from becoming limiting. This bounding assumption maximizes the PCT for the severed injection line event. The staff finds this evaluation conservative and bounding for injection line breaks.

RCP Trip on the Small Break

The effect of an RCP trip on SBLOCA for PSL-2 was previously addressed during the review of the extended power uprate (EPU) LAR in response to RAI SRXB-96 (RAI-2.8.5.6.3-19) (Reference 38). The licensee's response was based on a comparative evaluation with the PSL-1 EPU analyses for the effect of RCP operation (Reference 39). The comparative evaluation determined that the PSL-2 RCP trip criteria of 1736 psia pressurizer pressure for tripping one RCP in each loop and a minimum of 20 °F subcooling for tripping all four RCPs remain unchanged for the EPU. The comparative evaluation and the conclusion, with both PSL-1 and PSL-2 operating at a thermal power level of 3029.06 MWt and favorable minimum HPSI delivery (shown in Table 2-11 of response to SNPB RAI-11 in Section 2.11) for PSL-2, continue to remain applicable for the operation of PSL-2 with the fuel transition to AREVA 16x16 fuel. As a result, there are no significant changes to the plant parameters related to the SBLOCA analysis due to the implementation of the fuel design change to AREVA 16x16 fuel.

Post-LOCA Long-Term Cooling and Boric Acid Precipitation

Lastly, since post-LOCA long-term cooling timing of boric acid precipitation is not impacted by the AREVA transition fuel, no re-analysis of precipitation is required, and the current DBA analysis identifying precipitation time and the required Emergency Operating Procedure (EOP) operator action time to initiate simultaneous injection to preclude boric acid precipitation remains applicable.

3.1.9.6.3 SBLOCA Conclusion

The NRC staff reviewed the licensee's vendor small break LOCA analyses using S-RELAP5, applicable to PSL-2, operating at 3029.06 MWt and 13.0 kW/ft. The staff's review confirmed that the licensee and its vendor have processes to assure that the PSL-2 specific input parameter values and operator action times (where appropriate) that were used to conduct the analyses will assure that 10 CFR.50.46 limits are not exceeded following small break LOCAs.

Furthermore, the staff finds that the analyses were conducted within the conditions and limitations of the NRC-approved AREVA S-RELAP5 small break LOCA methodology, and that the results satisfied the requirements of 10 CFR 50.46(b) at a power level of 3029.06 MWt and a peak LHGR of 13.0 kW/ft.

In areas where the licensee and its contractors used NRC-approved methods in performing analyses related to the proposed nuclear fuel transition to AREVA nuclear fuel, the NRC staff reviewed relevant material to ensure that the licensee's use of these methodologies was consistent with the limitations and restrictions placed on these methods.

Based on its review of the licensee's small break LOCA analyses, the NRC staff concludes that the AREVA S-RELAP5 small break LOCA methodology is acceptable for PSL-2 to demonstrate acceptable ECCS performance and compliance with the temperature and oxidation criteria requirements of 10 CFR 50.46 at 3029.06 MWt and 13.0 kW/ft.

3.2 Reactor Systems

In this part of its review, the NRC staff reviewed the licensee's safety analyses for two classes of transients: AOOs and accidents. For AOOs, the staff reviewed the licensee's analyses to ensure that reactor protective functions served to mitigate AOOs in such fashion that the postulated events are terminated and mitigated without exceeding FCM, DNB, RCS, or main steam system pressure limits, and do not escalate to a more serious event without an independent fault. The staff reviewed two postulated accidents – the locked reactor coolant pump rotor and the large break LOCA analysis to ensure that the analyses acceptably demonstrated that the predicted results met the acceptance criteria for each of the events. For the seized rotor, the acceptance criteria found in SRP Chapter 15, which incorporate all applicable GDCs, are related to gross fuel failure and peak fuel enthalpy. For the LOCA analyses, the acceptance criteria are contained in 10 CFR 50.46(b)(1) through (b)(3).

3.2.1 Anticipated Operational Occurrences

The licensee is implementing the method described by AREVA licensing topical report EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," to analyze AOOs. The NRC staff reviewed the methodology, the PSL-2-specific implementation of the methodology, and the modeling assumptions and results of specific postulated transients, as described in the following sections.

3.2.1.1 Methods Implementation

3.2.1.1.1 General Methodology

The system analysis is performed with S-RELAP5, as described in EMF-2100(P), "S-RELAP5 Models and Correlations Code Manual." The reactor vessel nodalization provides modeling of the key components in the reactor vessel using junctions, volumes, and heat structures. The secondary side includes the tube bundles, feedwater system, separators, steamlines, and turbine simulator.

A complete reactor point kinetics model simulates the production of nuclear power in the core. The model computes both the immediate fission power and the power generated from decay of fission fragments and actinides. The model provides capability to include feedback due to moderator density and fuel temperature changes.

Fuel modeling contributes to the determination of the power and heat flux for the core. The heat flux determines the core coolant heating rate and, ultimately, the temperature response of the RCS to power changes. The power is affected by changes in fuel temperature that determine the Doppler feedback and by the change in the core coolant temperature that determines the moderator feedback. The fuel modeling is based on the RODEX2 and COPERNIC codes, which are addressed in Section 3.1 of this SE.

Fuel parameters that are considered are reflective of both beginning-of-cycle (BOC) and EOC conditions. BOC cases consider the maximum reactivity feedback and a low gap conductance, whereas EOC cases consider the minimum reactivity feedback and a high gap conductance.

The EMF-2310 method divides transients into two categories: fast and slow transients.

Not all events are analyzed on a cycle-specific basis. A disposition of events is prepared, which categorizes each of the events into one of the following four categories:

- Event is reanalyzed
- Event is bounded by another event
- Event is bounded by a previous analysis
- Event is outside the licensing basis of the plant

3.2.1.1.2 DNB Ratio (DNBR) Analysis

Detailed DNBR studies are performed on the hot rod using the XCOBRA-IIIC subchannel code. The code performs a quasi-steady state evaluation of the DNBR using statepoint parameters obtained from the S-RELAP5 code. Some of the S-RELAP5 parameters are biased for the DNBR evaluation in order to ensure that the DNBR result is limiting. The implementation of XCOBRA-IIIC is discussed in a subsequent section of this SE.

3.2.1.2 Increase in Heat Removal by the Secondary System

Events that increase the heat removal by the secondary system are feedwater system malfunctions that result in a decrease in feedwater temperature, feedwater system malfunctions that result in an increase in feedwater flow, excessive increase in secondary steam flow (excess load), and inadvertent opening of an SG relief or safety valve, which are all considered by both the license and the NRC staff to be ANS Condition II events.

All four of these events are analyzed to the following acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110 percent of design.
4. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

These acceptance criteria are consistent with SRP Chapter 15.1 and are, therefore, acceptable for application to PSL-2.

3.2.1.2.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

The feedwater system malfunctions that result in a decrease in feedwater temperature are described in the PSL-2 UFSAR Chapter 15.1.1 as follows:

> In the event of a loss of high pressure feedwater heaters, there could be an immediate reduction in feedwater temperature to the steam generators. At power, the increased sub-cooling will create a greater load demand on the reactor coolant system due to the increased heat transfer in the steam generator.

Both the AREVA and the UFSAR analyses identify the full power loss of high pressure feedwater heaters case as the more limiting transient. A comparison of the initial conditions and assumptions of the AREVA calculation to the PSL-2 UFSAR Section 15.1.1 reveals no significant differences.

A comparison of the credited RPS and engineered safety feature actuation system (ESFAS) actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.1.1 reveals no significant differences. In both analyses, the variable high power trip (VHPT) function is credited to initiate a reactor trip. The AREVA analysis properly accounts for excore detector decalibration (i.e., temperature shadowing) and resistance temperature detector (RTD) thermal lag.

A comparison of the nuclear steam supply system (NSSS) response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.1.1 reveals reasonable agreement. Steam flow rapidly increases, which promotes an increase in primary to secondary heat transfer, which in turn decreases core inlet temperature. With an assumed negative MTC, core power increases past the VHPT setpoint. A reactor scram terminates the power excursion. The results of the AREVA analysis demonstrate the MDNBR and peak LHGR limits are met.
Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the PSL-2 UFSAR, the NRC staff finds this analysis to be acceptable.

3.2.1.2.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The feedwater system malfunctions that result in an increase in feedwater flow are described in the PSL-2 UFSAR Chapter 15.1.2 as follows:

> A change in steam generator feedwater conditions that results in an increase in feedwater flow could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow are a result of a failure of a feedwater control valve, feedwater bypass valve, failure in the feedwater control system, operator error or accidental starting of the auxiliary feedwater system.

Both the AREVA and the UFSAR analyses consider both the hot full power and HZP cases. A comparison of the initial conditions and assumptions of the AREVA calculation to the PSL-2 UFSAR Section 15.1.2 reveals no significant differences.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.1.2 reveals no significant differences. In both analyses, the variable high power trip (VHPT) function is credited to initiate a reactor trip. The AREVA analysis properly accounts for excore detector decalibration (i.e., temperature shadowing) and RTD thermal lag.

A comparison of NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.1.2 reveals reasonable agreement. The increase in feedwater flow creates a mismatch between the energy being generated in the core and the energy being removed by the secondary system. The assumed negative MTC causes a power increase leading to a reactor trip, if the power increase is large enough, or stabilization at an elevated power level if the power increase is not large. The results of the AREVA analysis demonstrate the MDNBR and peak LHGR limits are met.

3.2.1.2.3 Excessive Increase in Secondary Steam Flow (Excess Load)

The excess load event is described in the PSL-2 UFSAR Section 15.1.3 as follows:

> A rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. For this accident the steam flow for EPU [extended power uprate] is assumed to increase by up to 58% of the initial value. This accident could result from one of the following: (1) An administrative violation such as excess loading by the operator, (2) [An] equipment malfunction in the steam dump control, or (3) [A] turbine throttle valve control malfunction.

Both the AREVA analysis and UFSAR identified the hot full power (HFP) excess load scenario as the more limiting case. This scenario results in an increase in main steam flow of up to 58 percent of the initial value.

The AREVA HFP excess load analysis included a parametric study varying MTC (-10 up to -33 percent millirho (pcm)/°F) and turbine control modes. The PSL-2 UFSAR Section 15.1.3

does not provide an explicit analysis because the pre-trip steamline break is more limiting and meets the Condition II acceptance criteria.

The staff reviewed the AREVA calculation initial conditions and assumptions and the credited RPS and ESFAS actuations to confirm their acceptability. The reactor is tripped on the VHPT setpoint and the analysis properly accounts for excore detector decalibration (i.e., temperature shadowing) and RTD thermal lag.

A review of the NSSS response and sequence of events of the AREVA calculation reveals that steam flow rapidly increases, which promotes an increase in primary to secondary heat transfer, which then in turn decreases core inlet temperature. With an assumed negative MTC, core power increases past the VHPT setpoint. A reactor scram terminates the power excursion.

Neither HFP nor HZP excess load events predict a violation of the DNB, FCM, or cladding strain SAFDLs. Therefore, a radiological assessment was not performed.

Based upon its review of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations, the NRC staff finds the excess load analysis acceptable.

3.2.1.2.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The PSL-2 UFSAR does not contain an explicit analyses of this event in Section 15.1.4. It contains an explanation of why the inadvertent opening of an SG relief or safety valve is bounded by the pre-trip steamline break and post-trip steamline break, both meeting the ANS Condition II acceptance criteria. The new AREVA calculation also contains a bounding explanation rather than an explicit analyses. The staff reviewed the AREVA calculation and concludes that this event is bounded by both the excess load and MSLB events.

3.2.1.3 Decrease in Heat Removal by the Secondary System

Events that cause a decrease in heat removal by the secondary system include loss of external load, turbine trip, loss of condenser vacuum, loss of non-emergency alternating current (ac) power to the station auxiliaries, loss of normal feedwater flow, feedwater system pipe break, and transients resulting from the malfunction of one SG. All of these events, except the feedwater system pipe break, are considered to be ANS Condition II events. The feedwater system pipe break event is an ANS Condition IV event.

The ANS Condition II events are analyzed to meet the following acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110 percent of design.
4. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

These acceptance criteria are consistent with SRP Chapter 15.2 and are, therefore, acceptable for application to PSL-2.

Not all of the above events are affected by a change in fuel characteristics, and therefore, only some of these events were re-analyzed for the fuel transition.

3.2.1.3.1 Loss of Condenser Vacuum

The PSL-2 UFSAR Section 15.2.3 describes the loss of condenser vacuum event as follows:

> A complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power with a simultaneous loss of feedwater to both steam generators due to low suction pressure on the feedwater pumps. The atmospheric dump valves and the steam dump and bypass system valves are assumed to be unavailable which minimizes the amount of cooling and maximizes the RCS and Secondary peak pressures during the event.

For the fuel transition, a detailed analysis was only performed for the case that challenges DNB and FCM. A comparison of the initial conditions and assumptions of the AREVA calculation to the PSL-2 UFSAR Section 15.2.3 reveals no significant differences.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.2.3 reveals no significant differences. In both analyses, no credit is taken for a reactor trip on turbine trip, and the high pressurizer pressure trip is credited to initiate a reactor trip.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.2.3 reveals reasonable agreement. After the turbine trip, the primary system temperature and pressure continue to rise until the high pressurizer pressure setpoint is reached and the reactor is tripped. The results of the AREVA analysis demonstrate the MDNBR and peak LHGR limits are met.

Based upon its review of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations and the PSL-2 UFSAR, the NRC staff finds the loss of condenser vacuum analysis acceptable.

3.2.1.3.2 Loss of Non-Emergency Alternating Current Power to the Station Auxiliaries

In both the PSL-2 UFSAR Section 15.2.6 and the AREVA calculation, this event is stated to be bounded by the loss of normal feedwater flow event. The AREVA calculation provides an additional disposition that the MDNBR is bounded by the complete loss of forced reactor coolant flow event. The staff finds the disposition of this event acceptable.

3.2.1.3.3 Loss of Normal Feedwater Flow

The PSL-2 UFSAR Section 15.2.7 describes the loss of normal feedwater flow event as follows:

> A complete loss of main feedwater flow while the reactor is operating at the maximum power level.

The fuel change does not affect the reactor power level, the safety valve capacities or opening setpoints, the SG trip setpoints, or the auxiliary feed capabilities, which are all important parameters affecting the results of this event. Therefore, as the fuel change does not affect these parameters, the current PSL-2 UFSAR analysis remains bounding for the fuel transition.

3.2.1.3.4 Feedwater System Pipe Break

The PSL-2 UFSAR Section 15.2.8 describes the feedwater system pipe break event as follows:

> A break in a feedwater line large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. Depending upon the size and location of the rupture and the plant operating conditions, the event can cause either a cooldown or a heatup of the reactor coolant system. Since the [RCS] cooldown resulting from a secondary system pipe break is covered by the steam line break event, only the RCS heatup aspects are emphasized for the case of feedwater line break.

The fuel transition only affects the DNB aspects of this event, which are bounded by the AREVA MSLB analysis. The current PSL-2 UFSAR feedwater system pipe break analysis for over pressure aspects of this event remain bounding for the fuel transition.

3.2.1.3.5 Transients Resulting from the Malfunction of One Steam Generator

The PSL-2 UFSAR Section 15.2.9 describes the asymmetric SG event as follows:

> A complete loss of steam load to one steam generator from a full power condition. This transient is modeled as an inadvertent closure of the main steamline isolation valve to one steam generator. A concurrent termination of feedwater flow to the affected steam generator is assumed in the analysis to conservatively bound any potential response of the feedwater system. Feedwater isolation to the affected steam generator will result in an increase in the vessel inlet temperature asymmetry during the transient, which is conservative with respect to demonstrating that the DNB design basis is satisfied.

A comparison of the initial conditions and assumptions of the AREVA calculation to the PSL-2 UFSAR Section 15.2.9 reveals no significant differences.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.2.9 reveals no significant differences. In both analyses, the high SG delta pressure trip is credited to initiate a reactor trip.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.2.9 shows reasonable agreement. After the MSIV closure, the turbine header pressure decreases and causes increased steam flow in the unaffected SG. This in turn causes a cooldown in the unaffected RCS loop resulting in asymmetric coolant temperatures entering the core. The reactor trip on high SG delta pressure will occur before any significant increase in augmented radial peaking can develop. The results of the AREVA analysis demonstrate that the MDNBR and peak LHGR limits are met.

Based upon its review of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations and the PSL-2 UFSAR, the NRC staff finds the asymmetric SG event analysis acceptable.

3.2.1.4 Loss of Forced Coolant Flow

Events that cause a loss of forced coolant flow include partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, reactor coolant pump shaft seizure (locked rotor), and reactor coolant pump shaft break. These events fall into three different ANS event types. Therefore, the acceptance criteria will be identified in each specific section below.

3.2.1.4.1 Partial Loss of Forced Reactor Coolant Flow

Both the AREVA calculation and the PSL-2 UFSAR Section 15.3.1 disposition this event as being bounded by the complete loss of forced coolant flow event. The staff finds this disposition acceptable.

3.2.1.4.2 Complete Loss of Forced Coolant Flow

The PSL-2 UFSAR Section 15.3.2 describes the complete loss of forced coolant flow event as follows:

> A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all [RCPs]. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in [DNB] with subsequent fuel damage if the reactor were not tripped promptly.

The acceptance criteria for this ANS Condition II event include:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110 percent of design.
4. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

These acceptance criteria are consistent with SRP Chapter 15.3.1 and are, therefore, acceptable for application to PSL-2.

A comparison of the initial conditions and assumptions of the AREVA calculation to the PSL-2 UFSAR Section 15.3.2 reveals no significant differences.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.3.2 reveals no significant differences. In both analyses, the low reactor coolant loop flow trip is credited to initiate a reactor trip.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.3.2 reveals reasonable agreement. The loss of forced reactor coolant flow causes a rapid increase in core coolant temperature. A reactor trip on low reactor coolant loop flow starts a cooldown before DNB occurs. The results of the AREVA analysis demonstrate the MDNBR and peak LHGR limits are met.

Based upon its review of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations and the PSL-2 UFSAR, the NRC staff finds the loss of forced reactor coolant flow event analysis acceptable.

3.2.1.4.3 Reactor Coolant Pump Shaft Seizure (locked rotor)

The RCP locked rotor event is described in the PSL-2 UFSAR Section 15.3.3 as follows:

> An instantaneous seizure of an RCP rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. The consequences of a postulated pump shaft break accident are similar to the locked-rotor event. With a broken shaft, the impeller is free to spin, as opposed to it being fixed in position during the locked-rotor event. Therefore, the initial rate of reduction in core flow is greater during a locked-rotor event than in a pump shaft break event because the fixed shaft causes greater resistance than a free-spinning impeller early in the transient, when flow through the affected loop is in the positive direction. As the transient continues, the flow direction through the affected loop is reversed. If the impeller is able to spin free, the flow to the core will be less than that available with a fixed-shaft during periods of reverse flow in the affected loop. Because peak pressure, cladding temperature, and DNB occur very early in the transient, the reduction in core flow during the period of forward flow in the affected loop dominates the severity of the

results. Consequently, the bounding results for the locked-rotor transients also are applicable to the RCP shaft break.

Classified as an ANS Condition IV postulated accident, this event is analyzed to the following acceptance criteria:

1. If fuel failure is predicted, the radiological consequences must not exceed the limits defined in Regulatory Guide 1.183, Table 6.

2. Pressure in the RCS and main steam system should be maintained below 120 percent of the design values in accordance with the ASME Boiler and Pressure Vessel Code and consider potential brittle as well as ductile failures.

These acceptance criteria are consistent with SRP Chapter 15.3.3 and are, therefore, acceptable for application to PSL-2.

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals no significant differences. Both analyses targeted the BOC for the most positive MTC.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.3.3 reveals no significant differences. Both analyses credit the low total RCS coolant flow trip, including a response delay time. One minor difference is that the AREVA analysis specifically models actuation of pressurizer sprays and power operated relief valves. Due to their timing, these actuations have no impact on thermal margin degradation and minimum DNBR.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 14.16 reveals reasonable agreement. In all cases, the minimum DNBR remained greater than the SAFDL. An evaluation of DNB propagation was not used for this event. This is acceptable since no fuel rods are predicted to experience DNBR.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the PSL-2 UFSAR, the NRC staff finds the locked rotor analysis acceptable.

3.2.1.4.4 Reactor Coolant Pump Shaft Break

As discussed in the preceding section, the consequences of a reactor coolant pump shaft break are bounded by the consequences of the locked rotor event. The staff finds this disposition acceptable.

3.2.1.5 Reactivity and Power Distribution Anomalies

Both the CEA drop event and a chemical volume control system (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant will be covered in this section and are Condition II events.

The acceptance criteria for these events include:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110 percent of design.
4. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

These acceptance criteria are consistent with SRP Chapter 15.4 and are, therefore, acceptable for application to PSL-2.

3.2.1.5.1 Control Element Assembly Drop

The CEA drop event is described in the PSL-2 UFSAR Section 15.4.3 as follows:

> A Control Element Assembly (CEA) Drop Event is a Condition II event that is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of rods from a CEA subgroup to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor may occur due to the skewed power distribution representative of a CEA drop configuration. Since this is a Condition II event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following a CEA Drop Event.

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals minor differences. For example, the UFSAR analysis uses a range of CEA worths of 100 pcm to 1000 pcm, while the AREVA calculation uses a range of 25 pcm to 1000 pcm. Both analyses targeted the EOC for the most negative MTC.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.4.3 shows no significant differences. Both analyses result in the plant establishing a new equilibrium condition at the original power level but at a reduced RCS temperature and pressure.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.4.3 shows reasonable agreement. In all cases, the minimum DNBR remained greater than the SAFDL, and LHGR remained below the limit.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the PSL-2 UFSAR, the NRC staff finds the CEA drop analysis acceptable.

3.2.1.5.2 CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

The CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant event is described in the PSL-2 UFSAR Section 15.4.6 as follows:

> Any event caused by a malfunction or an inadvertent operation of the CVCS that results in a dilution of the active portion of the RCS. The active portion of the RCS is defined as that volume of water that circulates through the core. For example, when in shutdown cooling (SDC), no credit is allowed for the volume of water in the stagnant portions of the RCS. A dilution of the RCS can be the result of adding water, which has a boron concentration that is less than the system boron concentration.

This event is analyzed to ensure that the boron dilution alarm system is activated in time to allow operator action to terminate the event and ensure protection of the minimum shutdown margin.

The staff reviewed the initial conditions and assumptions, the credited RPS and ESFAS actuations, and the NSSS response and sequence of events of the AREVA calculation. Operator errors and system malfunctions that allow the operator the shortest time for corrective action have been analyzed from plant conditions of startup, power operation (automatic and manual), hot standby, hot shutdown, cold shutdown, and refueling.

Results of the analyses showed that the operator has 30 minutes to take corrective action if a boron dilution incident occurs during refueling and 15 minutes if a boron dilution event occurs during startup, hot standby, hot shutdown, and cold shutdown. The minimum DNBR for this event is bounded by other events, such as CEA withdrawal at power and CEA withdrawal at startup. The reactor coolant and main steam system pressures remained less than 110 percent of design.

Based upon its review of the initial conditions and assumptions, RPS and ESFAS actuations, and the sequence of events of these AREVA calculations, and the fact that the boron dilution event requirements are verified each refueling cycle, the NRC staff finds this analysis acceptable.

3.2.1.5.3 Inadvertent Loading of a Fuel Assembly

The fuel transition does not affect the ability of any of the existing processes and surveillance requirements to preclude significant misloads from going undetected. The staff finds this disposition acceptable.

3.2.1.6 Increase in Reactor Coolant Inventory

The events that cause an increase in RCS inventory are the inadvertent operation of the ECCS and the CVCS malfunction that increases RCS inventory. The inadvertent ECCS actuation is not analyzed because the shutoff head of the Safety Injection pumps is lower than the RCS pressure during Mode 1 operation.

The CVCS malfunction that increases RCS inventory is not affected by the change in fuel design parameters of the fuel transition. There are no changes to the high pressurizer pressure trip or pressurizer high level alarm or operator actions that are depended on in the UFSAR analysis of record. The staff finds that the current UFSAR analysis remains bounding for the fuel transition.

3.2.1.7 Decrease in Reactor Coolant Inventory

Several of the decrease in reactor coolant inventory events are not affected by the change in fuel design parameters of the fuel transition or are bounded by other events. The break in instrument line or other lines from the RCPB that penetrate the containment is bounded by the inadvertent opening of a pressurizer safety or relief valve with respect to DNB. The RCS mass release aspect is not affected by the fuel transition, and the current UFSAR analysis remains bounding. The current SG tube rupture analysis in the UFSAR also remains bounding. There are no changes to reactor power, reactor trip setpoints, safety valve opening setpoints or tolerances, or the TS limit for primary to secondary leakage. The staff finds the disposition of the break in instrument line or other lines from the RCPB that penetrate the containment event and the SG tube rupture event analyses acceptable.

3.2.1.7.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The Inadvertent opening of a pressurizer safety or relief valve event is described in the PSL-2 UFSAR Section 15.6.1 as follows:

> An accidental depressurization of the reactor coolant system could occur as a result of one of the following: an inadvertent opening of both of the pressurizer Power Operated Relief Valves (PORVs), an inadvertent opening of a single Pressurizer Safety Valve (PSV), or a malfunction of the pressurizer spray system.

The event was analyzed to the following Condition II acceptance criteria:

1. Fuel cladding integrity shall be maintained by ensuring SAFDLs are not exceeded.
2. Fuel centerline melting shall not occur.
3. Peak primary and secondary pressure shall remain below 110 percent of design.
4. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

These acceptance criteria are consistent with SRP Chapter 15.6.1 and are, therefore, acceptable for application to PSL-2.

A comparison of the initial conditions and assumptions of the AREVA calculation to the UFSAR reveals no significant differences. The DNBR case is analyzed by AREVA for the fuel transition.

A comparison of the credited RPS and ESFAS actuations of the AREVA calculation to the PSL-2 UFSAR Section 15.6.1 reveals no significant differences. Both analyses credit the thermal margin low pressure trip for preventing fuel damage.

A comparison of the NSSS response and sequence of events of the AREVA calculation to the PSL-2 UFSAR Section 15.6.1 reveals reasonable agreement. In all cases, the AREVA calculation demonstrates that minimum DNBR remained greater than the SAFDL and LHGR remained below the limit. The pressurizer overfill case, which is terminated by operator action, is not affected by the fuel transition, and the analysis of record remains bounding.

Based upon its review of the supporting AREVA calculations, comparison of the initial conditions and assumptions, RPS and ESFAS actuations, and sequence of events of these AREVA calculations to the PSL-2 UFSAR, the NRC staff finds the inadvertent opening of a pressurizer safety or relief valve analysis acceptable.

3.2.1.8 Primary System Deviation Events and Anticipated Transients Without Scram

None of the events in the PSL-2 UFSAR Section 15.8 challenge the DNBR SAFDL or are significantly affected by the change in fuel design parameters of the fuel change. The PSL-2 UFSAR Section 15.8 primary system deviation events remain bounding. The change in fuel design parameters has no effect on the anticipated transients without scram events described in the PSL-2 UFSAR Section 15.9. The staff finds the disposition of these events acceptable.

3.2.1.9 Station Blackout

The station blackout event is defined in the PSL-2 UFSAR Section 15.10 as follows:

> The Station Blackout event results from a loss of offsite power followed by failure of both standby diesel generators to start. For Unit 2, this event results in a loss of all onsite ac power except that supplied by inverters from the two safeguards batteries. This provides power to the 120V AC (safeguards) instrument power and other required DC loads. The SBO [station blackout] cross-tie can be used to supply additional power to Unit 2 to power selected equipment, however, there was no credit taken for that in the SBO analysis.

The complete loss of forced reactor coolant flow event bounds the initial part of the SBO. The rest of the event is governed by operator actions, RCP seal leakage, and the secondary side performance of the main steam safety valves (MSSVs) and the auxiliary feedwater turbine driven pump, none of which are affected by the changes in fuel design parameters of the fuel transition. The SBO event was not re-analyzed for the fuel transition. The staff finds the disposition of the SBO event acceptable.

3.2.2 Large Break Loss-of-Coolant Accidents (LBLOCAs)

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection system and ECCS are provided to mitigate these accidents.

The NRC's acceptance criteria are based on the following:

(1) 10 CFR 50.46, which establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance;
(2) 10 CFR Part 50, Appendix K, which establishes required and acceptable features of evaluation models for heat removal by the ECCS during and after the blowdown phase of an LOCA;
(3) GDC 4, which requires that structures, systems, and components important to safety be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit;
(4) GDC 27, which requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and
(5) GDC 35, which requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented.

Specific review criteria are contained in SRP Sections 6.3 and 15.6.5.

The LBLOCA event is described in the PSL-2 UFSAR Section 15.6.5.1 as follows:

> A pipe break with a total break area equal to or greater than 1.0 ft^2. As identified in Section 15.0.6.4, a large break LOCA is classified as an ANS Condition IV plant condition, i.e., a limiting fault.

3.2.2.1 Methodology Implementation and the Analytical Model

The licensee's best estimate, LBLOCA analyses supporting the fuel transition was performed by AREVA and documented in ANP-3346(P) Revision 1, "St. Lucie Unit 2 Fuel Transition Realistic Large Break LOCA Summary Report." The analysis was conducted in accordance with EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," which received approval as documented in an NRC SE dated April 9, 2003 (ADAMS Accession No. ML030760312). EMF-2103 is a best estimate code and uses a statistical method based on order statistics to produce an estimate of the upper tolerance limit for predicted peak cladding temperature (PCT), consistent with the "high level of probability" statement contained in

10 CFR 50.46(a)(1)(i). Upper tolerance limit estimates are also produced for the maximum local oxidation and hydrogen generation. The NRC SE approving EMF-2103(P)(A), Revision 0, included 13 conditions and limitations restricting its use. The licensee provided Table 3-4 in ANP-3346(P), which listed the conditions and limitations and included information to demonstrate that adherence to each had been attained. The NRC staff reviewed this information and concluded that the licensee's implementation of EMF-2103 adheres to each of the conditions and limitations.

Subsequent to its approval of Revision 0 to EMF-2103(P)(A), the NRC staff has found that certain modeling assumptions and constitutive relationships contained in the EMF-2103 methodology are not suitable for demonstrating compliance with the 10 CFR 50.46(b) acceptance criteria, as described in a draft SE dated April 3, 2007.[2] Therefore, the staff also considered the conditions and limitations in this draft SE, and the corresponding departures from NRC-approved EMF-2103(P)(A), Revision 0, required to adhere to these conditions and limitations. The licensee provided information to address these issues in Section 5, "Generic Support for Transition Package," and Section 6, "Recent NRC Request for Additional Information (RAI) and AREVA Responses."

The power assumed in the analyses, 3029.06 MWt, includes a 0.3 percent measurement uncertainty recapture relative to the current rated thermal power. This is a departure from the previously approved methodology and is acceptable because it is conservative in that the previously approved methodology permitted ranging the assumed power level, meaning that some cases could have initiated at a power level less than 3029.06 MWt, had the analysis been performed using the previously approved methodology. It is also acceptable because it is consistent with the NRC staff's position that parametrically ranging the assumed initial power level is inconsistent with 10 CFR 50.46 requirements, whereas deterministically including uncertainty in the assumed initial power level is acceptable.

The realistic LBLOCA (RLBLOCA) analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than **[[ ]]** and the **[[ ]]** before the rod is allowed to quench. During its review of EMF-2103(P), Revision 1, the NRC staff determined that the S-RELAP5 evaluation model could allow rod quench to occur once the temperature drops below the minimum film boiling temperature, regardless of the void fraction in the channel. Contrarily, NUREG-0915 demonstrated that the void fraction must also be less than 0.95 for rod rewet to occur. To address this concern for the St. Lucie Plant, the licensee stated that the **[[ ]]** before the rod is allowed to quench. T_{min} is a sampled parameter, and the licensee confirmed that T_{min} was never sampled **[[ ]]**. The staff finds this departure from the methodology acceptable because the departure provides for analytic predictions that

[2] The referenced SE was never formally issued; the vendor withdrew the topical report revision that it supported. Therefore, there is no publicly available copy of this report. Nonetheless, the NRC staff has continued to use adherence to the conditions and limitations listed in that SE as a basis for its approval of requests to implement EMF-2103, Revision 0, as an interim review approach, until a second revision to EMF-2103 is submitted to the NRC for review and approval. While EMF-2103(P)(A) is an acceptable evaluation model as described in 10 CFR 50.46, the NRC staff requires these deviations for plant-specific application of the methodology.

are not only more consistent with observed data, but also more conservative than predictions obtained using the previous NRC-approved methodology.

The RLBLOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than **[[ ]]** percent of the total heat transfer at and above a void fraction of **[[ ]]**. This departure from the approved methodology accounts for experimentally observed phenomena that appear to inhibit droplet contact with heated fuel rods at high void fractions. Thus, this departure is conservative relative to the approved methodology because it corrects for any potential to over-predict heat transfer through conduction to entrained droplets, which experimental observations have shown not to come in contact with the fuel rods at such high void fractions. The net effect of this conservatism would serve to increase the overall predicted PCT compared to evaluations performed using the previous NRC-approved methodology.

The analyses addressed the availability of offsite power correctly by ranging each case separately. This is acceptable because it satisfies GDC 35 of 10 CFR Part 50, Appendix A, in that each distribution type has been accounted for separately with its own set of cases, thereby addressing possible concerns associated with the mixing of two separate statistical spectra. The NRC staff finds this treatment acceptable because it is consistent with the staff's position regarding compliance with GDC 35 as noted in ANP-3346(P).

Downcomer boiling is caused by metal heat release from vessel and core barrel walls to fluid in the downcomer gap. Metal heat from the vessel lower head and structures in the lower plenum also contribute to downcomer boiling. As heat is released to the downcomer fluid, its temperature is gradually increased, eventually subcooled, and saturated boiling takes place. Voids generated by these processes displace water in the downcomer and reduce the driving head that forces water into the core during the reflood phase of a large break LOCA. This loss in head can significantly reduce the core flooding rate, and increase the peak cladding temperature.

The NRC staff has historically identified differences in results between staff confirmatory calculations and those produced using the AREVA RLBLOCA evaluation model, attributable to downcomer boiling modeling, which cause significant differences in peak cladding temperature results. As discussed in Section 1.0 of ANP-3346(P), AREVA attributes this to an underprediction of cold leg condensation efficiency. To correct for this, AREVA has identified appropriate multipliers to force fluid entering the downcomer to saturated conditions following the deployment of the safety injection tanks. The staff finds this departure from the previously approved methodology acceptable because: (1) the artificially saturated fluid conditions will conservatively reduce both the downcomer driving head and the core flooding rate, which becomes conducive to portions of the fuel remaining in a vapor-cooled environment, thus presenting a greater challenge to clad surface cooling; and (2) conditions in the downcomer following safety injection tank discharge are expected to be slightly subcooled, meaning that assuming fully saturated conditions is conservative.

The licensee's treatment of downcomer boiling is stated to depend on the heat release model and on the ability to track steam rising through the downcomer. The heat release model in

S-RELAP5 was validated by a sensitivity study on wall mesh point spacing and through benchmarking against a closed form solution. The steam tracking was validated through both an axial and an azimuthal fluid control volume sensitivity study done at low pressures. The licensee states that the studies demonstrate that S-RELAP5 appropriately predicts the delivery rate of energy to the downcomer liquid volume and contains sufficient downcomer nodalization detail to track the distribution of any steam formed. The staff therefore finds that the licensee has appropriately considered downcomer boiling in its analyses.

The staff continues to have concerns that AREVA's omission of a model representing fuel rod swell, rupture, and pellet relocation is unjustified and possibly non-conservative. However, since the predicted peak cladding temperatures at PSL-2 do not exceed 1800 °F, the NRC staff finds that there is reasonable assurance that blowdown cladding ruptures would not occur during a postulated LOCA at PSL-2, and thus, the model is acceptable because it provides an acceptable representation of the LBLOCA progression without modeling a blowdown cladding rupture.

NRC Information Notice 2009-23 describes an issue concerning the ability of legacy thermal-mechanical fuel modeling codes to predict the exposure-dependent degradation of fuel thermal conductivity accurately. Some legacy codes, including RODEX-3A, non-conservatively over-predict fuel thermal conductivity at higher burnups. A safety concern with fuel thermal conductivity degradation (TCD) in a LOCA would be that fuel temperatures modeled incorrectly would affect the heat transfer to the cladding surface, causing the LOCA evaluation model to predict potentially erroneous PCTs. To correct for this issue, AREVA has applied a polynomial transformation that is used to bias the fuel pellet centerline temperature based on empirical data collected supporting the more recent RODEX4 fuel performance code. The staff finds that this polynomial bias adequately accounts for fuel thermal conductivity degradation.

Consistent with NRC regulatory guidance for realistic LOCA evaluation models described in Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," the licensee uses the ANS-1979 standard for decay heat power in light-water reactors. In the analysis provided, the uncertainty for the decay heat parameter is set to zero, and no sampling is done on this parameter, resulting in the decay heat being used with a 1.0 multiplier. This is a change from the EMF- 2103 evaluation model. The decay heat in the analysis is always the 1979 ANS standard for decay heat from U^{235} with fully saturated decay chains, corresponding to infinite operation, assuming 200 MeV per fission. The infinite operation decay chain was compared to several finite decay in order to demonstrate that uncertainty is properly accounted for in the infinite chain by bounding the finite chains. It was shown that 2 seconds after shutdown, the infinite chain bounds all of the finite chains. During the first 2 seconds when the infinite chain is not bounding, it was determined that the difference in energy deposited between the infinite chain and the bounding finite chain would have an insignificant effect on clad temperature because stored energy will have a much greater influence on PCT than early in the transient. The NRC staff finds this treatment of decay heat acceptable because the licensee appropriately incorporated an acceptable model identified in Regulatory Guide 1.157 and demonstrated that uncertainty was accounted for using a bounding method.

The licensee stated, "[Florida Power & Light Company] and the LBLOCA Analysis Vendor have an ongoing process to ensure that all input variables and parameter ranges for the PSL-2 realistic large break loss-of-coolant accident are verified as conservative with respect to plant operating and design conditions." These statements, together with the NRC staff's review of the results of the RLBLOCA analyses, provide reasonable assurance that the AREVA methodology with deviations applies to PSL-2, pursuant to 10 CFR 50.46(c)(2), and that the licensee has applied it properly.

Determination of Break Locations and Break Sizes

The analyses ranged in area between the minimum break area and an area of twice the size of the broken pipe. The licensee stated that A_{min} (minimum break area) was calculated to be 29.9 percent of the double-ended guillotine break area. This information demonstrates that the total number of sampled cases is appropriate because the phenomenology dominating the limiting aspects of the event for all sampled break areas is consistent. That is, a certain number of sampled cases is appropriate, because the limiting results of the accident for pipe ruptures ranging from about 20 to 100 percent of the double-ended pipe rupture size are all limited by dispersed flow film boiling ahead of the quench front. If the sampled break area included a greater range (i.e., break sizes less than 20 percent of the double-ended guillotine rupture), additional phenomenology would govern the limiting events. Additional cases would be required to provide the necessary high level of statistical confidence that a bounding upper tolerance limit had been identified.

The licensee stated that the worst break location was identified by deterministic studies and determined to be in the cold leg between the RCP and the reactor vessel (RV) for the RCS loop containing the pressurizer. The NRC staff reviewed the methodology and confirmed this to be true. The method did not consider slot breaks because PSL-2 does not have any loop seals with bottom elevation below the top elevation of the core. The NRC reviewed this information and confirmed that Condition 8 of the SE approving EMF-2103, which requires that the licensee submit the results of the plant-specific analyses, including the calculated worst break size, PCT, and local and total oxidation, was satisfied. On this basis, the NRC staff finds the licensee's conclusion that the PSL-2 PCT-limiting transient is a double-ended cold leg guillotine break acceptable, because uncertainties related to break type and size were appropriately included in the modeling approach.

Postulated Initial Conditions and Sequence of Events

The current rated thermal power for PSL-2 is 3020 MWt plus 0.3 percent power measurement uncertainty. The assumed reactor core power used in the RLBLOCA analysis is 3029.06 MWt, which represents the current rated thermal power and the 0.3 percent power measurement uncertainty. The RLBLOCA analysis also assumed a SG tube plugging level of 20 percent in all SGs, a total LHGR of 13.0 kW/ft (no axial dependency), a total peaking factor (F_Q) up to a value of 2.504, and a nuclear enthalpy rise factor (Fr) up to a value of 1.81 (including 6 percent measurement uncertainty and 3.5 percent control rod insertion uncertainty).

The single failure assumption was based on the approved RLBLOCA evaluation model (EM), EMF-2103(P)(A), Rev. 0. The single failure scenario is a diesel loss with fully functional containment sprays at TS minimum temperature and pumped ECCS injection at TS maximum temperature. Containment pressure is indirectly sampled through containment volume. A sensitivity study on the limiting case for a maximum ECCS injection scenario is provided in Section 6.6 of ANP-3346P Rev. 0. The maximum ECCS scenario resulted in a slightly lower PCT as the RLBLOCA EM single failure; therefore, the RLBLOCA EM single failure remains the most limiting.

Results

The RLBOCA analyses were performed to demonstrate that the system design would provide sufficient ECCS flow to transfer the heat from the reactor core following an LBLOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than would compromise cladding ductility and result in excessive hydrogen generation. The NRC staff reviewed the analyses to assure that they reflected suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities were available such that the safety functions could be accomplished, assuming a single failure, for LBLOCAs considering the availability of onsite and offsite electric power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available). The acceptance criteria for ECCS performance, provided in 10 CFR 50.46, were used by the staff in assessing the acceptability of the AREVA EMF-2103 methodology for PSL-2.

In its submittal, the licensee provided the analysis results for the best estimate large break loss-of-coolant accident analyses for the fuel transition conditions, which were produced in accordance with the EMF-2103 methodology. The licensee's results for the calculated PCTs, the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table, along with the acceptance criteria of 10 CFR 50.46(b).

Parameters	**Fresh UO2 Fuel**	**Once Burned UO2 Fuel**	**10 CFR 50.46 Limits**
Cladding Material	M5	M5	Zircaloy (Exemption for M5 requested)
Peak Cladding Temperature	1732 °F	1639 °F	2200 °F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	3.337%	3.382%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	0.0139%	NA	1.0% (10 CFR 50.46(b)(3))

The licensee's analytic limiting local maximum oxidation is a combination of the pre-transient and transient local oxidation. The transient oxidation contribution was shown to decrease from the BOL value to a negligible value at EOL. This result is expected because fuel is generally more susceptible to transient oxidation at the BOL. The results demonstrate that the sum of pre-transient plus transient local oxidation remains below 17 percent at all times in life for the AREVA fuel.

The limiting core-wide oxidation is based on the maximum values of core-wide transient oxidation computed for the case set. Because the oxidation is 0.0139 percent, there is significant margin to the regulatory limit, and the NRC staff finds that the licensee has adequately demonstrated that the core-wide oxidation will remain less than 1 percent.

3.2.2.2 Conclusion

The NRC staff has reviewed the licensee's analyses of the LBLOCA event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the fuel transition conditions and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the RPS and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the applicable LOCA requirements following implementation of the proposed fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable.

3.3 Radiation Protection and Consequences

The NRC staff's review of the radiation protection and consequences included a review of the radiological impact of the proposed transition to AREVA fuel on previously analyzed DBA radiological consequences.

3.3.1 Evaluation

By letter dated September 29, 2008 (ADAMS Accession No. ML082060400), the NRC approved a license amendment to fully implement an AST methodology at PSL-2. By letter dated September 24, 2012 (ADAMS Accession No. ML12235A463), the NRC approved a license amendment that re-analyzed the AST methodology for an increase in the maximum steady state reactor core power level at PSL-2. To support this license amendment, the licensee did an analysis that determined the PSL-2 bounding source term, which is used for the radiological DBAs that assume fuel failures. This analysis was performed based on the release fractions given in Regulatory Guide 1.183. The core isotopic activity was calculated using ORIGEN-2.1 computer code, which is in accordance with Regulatory Guide 1.183 guidance, and therefore, acceptable to the NRC staff. In the analysis, RADTRAD NAI computer code was used to calculate a TEDE at the exclusion area boundary for the limiting 2-hour period, at the low population zone outer boundary for the duration of the accident, and in the control room for 30 days for each analyzed DBA.

A modification to the licensing basis fuel type can have the potential to change the core isotopic activity distribution assumed for post-accident conditions. In its supplemental letter dated June 2, 2015, the licensee stated that the fuel type change does not change the fuel rod dimensions, rod configuration, and core design parameters, as compared to the fuel currently used in the PSL-2 core analyzed for the radiological consequences. In addition, the licensee

stated that the following fuel parameters and core design parameters used in the isotopic calculations remain unchanged:

- Pellet diameter
- Cladding outside diameter
- Cladding thickness
- Active fuel height
- Rod configuration (16x16)
- Total number of fuel assemblies in the core
- Core power level
- Maximum radial peaking factor
- Maximum assembly power level
- Peak assembly average burnup
- Core average burnup
- Fuel enrichment (minimum and maximum)
- Maximum mass of uranium per assembly

The licensee concluded that the fuel parameters and the core design parameters listed above used in developing a bounding source term for radiological DBAs, remains unchanged for the fuel design change proposed in this LAR. Thus, no additional source term analyses need to be performed in support of the proposed fuel transition.

The NRC staff has reviewed the licensee's radiological consequence analyses of the AREVA fuel on the AST source term. The staff finds that there are no changes to the NRC-approved methodology stated in the PSL-2 current licensing basis. In addition, the current Westinghouse fuel-based source term input for the radiological DBAs does not need to be updated because the fuel parameters and the core design parameters listed above will remain the same for the new AREVA fuel. Therefore, the AREVA fuel-based source term would be the same as the Westinghouse fuel-based source term. Furthermore, the PSL-2 DBAs regulatory dose limits are unaffected and still meet the regulatory requirements in 10 CFR 50.67 and GDC 19.

3.3.2 Conclusion

The NRC staff reviewed the analysis used by the licensee to assess the radiological impacts of the transition from Westinghouse fuel to AREVA fuel at PSL-2. The staff finds that the licensee used methods consistent with regulatory requirements and guidance identified above. The staff finds, with reasonable assurance, that the licensee's estimates of the exclusion area boundary, low-population zone, and control room doses will remain unchanged and continue to comply with these criteria. Therefore, the proposed change and associated TS changes are acceptable with regard to the radiological consequences of postulated DBAs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida

Department of Health) of the proposed issuance of the amendment on March 25, 2016. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (80 FR 32620; June 9, 2015). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

Proprietary documents were used for the NRC staff's review. For these references, non-proprietary versions, where available, are provided in the list below and are publicly available.

1. FPL Letter L-2014-366 to USNRC, "Application for Technical Specification Change and Exemption Request Regarding the Transition to AREVA Fuel," December 30, 2014 (ADAMS Accession No. ML15002A091).

2. FPL Letter L-2015-091, "Supplemental Information for Technical Specification Change and Exemption Request Regarding the Transitioning to AREVA Fuel," FPL, March 23, 2015 (Attachment 2, ANP-3396P, "St. Lucie Unit 2 Fuel Transition Supplemental Information to Support the LAR) (ADAMS Accession No. ML15084A011).

3. FPL Letter L-2015-166, "RAI Reply for Application for Technical Specification Change and Exemption Request Regarding the Transitioning to AREVA Fuel," June 2, 2015 (ADAMS Accession No. ML15161A316).

4. FPL Letter L-2015-177, "Resubmittal of the AREVA Small Break LOCA Summary Report Within the Application for Technical Specification Change and Exemption Request Regarding the Transitioning to AREVA Fuel," June 18, 2015 (ADAMS Accession No. ML15181A290).

5. FPL Letter L-2015-206, "RAI Response to SNPB-RAI-1 for the Technical Specification LAR and Exemption Request Regarding the Transitioning to AREVA Fuel," July 30, 2015 (ADAMS Accession No. ML15219A184).

6. FPL Letter L-2015-259, "RAI Response to SRXB-RAI-1 and SNPB RAI-2 thru RAI-20 for the Technical Specification LAR and Exemption Request Regarding the Transitioning to AREVA Fuel," October 2, 2015 (ADAMS Accession No. ML15279A226).

7. FPL Letter L-2015-279, "RAI Response Clarification for SRXB-RAI-1 and SNPB RAI-2 thru RAI-20 for the Technical Specification LAR and Exemption Request Regarding the Transitioning to AREVA Fuel," November 3, 2015 (ADAMS Accession No. ML15322A134).

8. FPL Letter L-2015-300, "RAI Response for SNPB RAI-9 for the Technical Specification LAR and Exemption Request Regarding the Transitioning to AREVA Fuel," FPL, December 8, 2015 (Attachment 1, ANP-3456P, "Response to SLU2 NRC SNPB RAI-9," Attachment 2, ANP-3352P, Revision 1, "St. Lucie Unit 2 Fuel Transition License Amendment Request, Technical Report") (ADAMS Accession No. ML15356A184).

9. ANP-3352P, Revision 1, "St. Lucie Unit 2 Fuel Transition License Amendment Request Technical Report," AREVA, Inc., November 2015 (ADAMS Accession No. ML15356A189).

10. ANP-3352P, Revision 0, "St. Lucie Unit 2 Fuel Transition License Amendment Request Technical Report," AREVA, Inc., December 2014 (ADAMS Accession No. ML15002A093).

11. BAW-10240(P)(A), Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods," Framatome ANP Inc., May 2004 (ADAMS Accession No. ML042800314).

12. BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," Framatome Cogema Fuels, June 2003 (ADAMS Accession No. ML15162B052 (non-public)), as noted by letter dated April 19, 20014 (ADAMS Accession No. ML15162B047).

13. ANP-3440P, Revision 1, "St. Lucie Unit 2 Fuel Transition: Response to NRC Questions SRXB-RAI-1 and SNPB RAI-2 through RA-20 – Technical Report," AREVA, Inc., September 2015 (ADAMS Accession No. ML15279A228).

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

15. EMF-92-116(P)(A), Revision 0, Supplement 1(PA), Revision 0, "Generic Mechanical Design Criteria for PWR Reactor Fuel," AREVA, Inc., February 2015 (ADAMS Accession No. ML15148A159 (non-public)). EMF-92-116(NP)(A), Revision 0 (ADAMS Accession No. ML15148A158).

16. Letter, Pedro Salas (AREVA NP Inc.) to NRC, "Response to a Request for Additional Information Regarding EMF 92-116(P)(A), Revision 0, Supplement 1, Revision 0," November 11, 2013, and August 7, 2014 (ADAMS Accession No. ML13317B917 and ADAMS Package Accession No. ML14220A469, respectively).

17. BAW-10133P-A, Revision 1, Addenda 1 and 2, "Mark-C Fuel Assembly LOCA-Seismic Analyses," Framatome Cogema Fuels, October 2000 (ADAMS Accession No. ML003767624).

18. BAW-10172(P)(A), Revision 0, "Mark-BW Mechanical Design Report," December 1989 (ADAMS Accession No. ML040960584 (non-public)).

19. ANP-3456P, Revision 0, "Response to SLU2 NRC SNPB RAI-9 Licensing Report," AREVA, Inc., December 2015 (ADAMS Accession No. ML15356A187 (non-public)).

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LIST OF ACRONYMS

ACRONYM	DEFINITION
ac	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BOC	Beginning-of-Cycle
BOL	Beginning-of-Life
CE	Combustion Engineering
CEA	Control Element Assemblies
CEAW	CEA [Control Element Assemblies] Bank Withdrawal
CHF	Critical Heat Flux
CIT	Core Inlet Temperature
COLR	Core Operating Limits Report
CVCS	Chemical Volume Control System
°F	Degrees Fahrenheit
DBA	Design-Basis Accident
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOL	End-of-Life
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System
FAP	Fuel Alignment Plate
FCM	Fuel Centerline Melt
FPL	Florida Power & Light Company
gpm	Gallons Per Minute
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GWd/MTU	Gigawatt-Days per Metric Ton Uranium Metal
HFP	Hot Full Power
HMP™	High Mechanical Performance
HPSI	High Pressure Safety Injection
HTP™	High Thermal Performance
HZP	Hot Zero Power
ID	Inner Diameter
kW/ft	Kilowatt per Foot
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection

ACRONYM	DEFINITION
LTP	Lower Tie Plate
LWR	Light-Water Reactor
MDNBR	Minimum Departure from Nucleate Boling Ratio
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MWt	Megawatts Thermal
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
pcm	Percent Mil
PCT	Peak Cladding Temperature
PORV	Power Operated Relief Valve
PSV	Pressurizer Safety Valve
PWR	Pressurized-Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
rem	Roentgen Equivalent Man
RLBLOCA	Realistic Large Break Loss-of-Coolant Accident
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SBO	Station Blackout
SDC	Shutdown Cooling
SE	Safety Evaluation
SG	Steam Generator
PSL-1	St. Lucie Plant, Unit No. 1
PSL-2	St. Lucie Plant, Unit No. 2
SRP	Safety Review Plan
SSE	Safe Shutdown Earthquake
TCD	Thermal Conductivity Degradation
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UTP	Upper Tie Plate
VHP	Variable High Power
VHPT	Variable High Power Trip

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA by RGladney for/

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 182 to Renewed NPF-16
2. Safety Evaluation (non-public)
3. Safety Evaluation (public)

cc w/ Enclosures 1 and 2: Addressee only
cc w/ Enclosures 1 and 3: Distribution via Listserv

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