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

December 12, 2025

Mr. Robert Coffey
Executive Vice President, Nuclear
and Chief Nuclear Officer
Florida Power & Light Company
700 Universe Blvd.
Mail Stop: EX/JB
Juno Beach, FL 33408

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT NO. 213 TO REVISE TECHNICAL SPECIFICATION 5.6.3, "CORE OPERATING LIMITS REPORT," BY UPDATING THE LISTING OF NRC-APPROVED ANALYTICAL METHODS USED TO DETERMINE THE CORE OPERATING LIMITS (EPID L-2024-LLA-0124)

Dear Mr. Coffey:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 213 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant (St. Lucie), Unit No. 2. The amendment is in response to your application dated September 11, 2024, as supplemented by letter dated May 29, 2025.

The amendment revises St. Lucie, Unit No. 2, Technical Specification 5.6.3, "Core Operating Limits Report," by updating the listing of NRC-approved analytical methods used to determine the core operating limits. Specifically, changes to the fuel thermal-mechanics, core thermal-hydraulics, emergency core cooling, nuclear design, and select design-basis event analyses are proposed using NRC-approved advanced codes and methods in support of a St. Lucie, Unit No. 2, transition to 24-month fuel cycles.

Enclosure 2 to this letter contains proprietary information. When separated from Enclosure 2, this document is DECONTROLLED.

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

Sincerely,

/RA/

Natreon Jordan, 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. 213 to NPF-16
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

cc: 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. 213
Renewed License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (FPL, the licensee), dated September 11, 2024, as supplemented by letter dated May 29, 2025, 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.

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, in part, to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, 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 within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: December 12, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 213

ST. LUCIE PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following page of Renewed Facility Operating License No. NPF-16 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
5.6-6	5.6-6
5.6-7	5.6-7
5.6-8	5.6-8
5.6-9	5.6-9
	5.6-10

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. 213, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. DELETED

D. Antitrust Conditions

FPL shall comply with the antitrust conditions in Appendices C and D to this renewed license.

E. Fire Protection

Florida Power & Light Company (FPL) St. Lucie Plant Unit 2 shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated March 22, 2013, and May 2, 2017, and supplements dated June 14, 2013, February 24, 2014, March 25, 2014, April 25, 2014, July 14,

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

43. 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).

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

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

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

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

48. WCAP-16045-P-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

49. ANP-10297P-A, Revision 0, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," February 2013.

50. ANP-10338P-A, Revision 0, "AREA™ - ARCADIA® Rod Ejection Accident," December 2017.

51. ANP-10297, Revision 0, Supplement 1P-A, Revision 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," December 2020.

52. 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.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

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

54. 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.

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

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

57. 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.

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

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

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

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

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

63. EMF-2103(P)(A) Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," June 2016.

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

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

65. EMF-2328(P)(A), Revision 0, Supplement 1(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016.

66. ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods," November 2021.

67. ANP-10311 P-A, Revision 1, “COBRA-FLX: A Core Thermal-Hydraulic Analysis Code," October 2017.

68. ANP-10311, Revision 1, Supplement 1P-A, Revision 0, ”COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation," March 2023.

69. ANP-10339P-A, Revision 0, “ARITA - ARTEMIS/RELAP Integrated Transient Analysis Methodology," October 2023.

70. BAW-10227P-A, Revision 2, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," January 2023.

71. [Not used]

72. ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018.

73. ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," November 2020.

74. BAW-10084P-A, Revision 3, "Program To Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

c. The core operating limits shall be determined assuming operation up to RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

5.6.4 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.5 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.6, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

5.6 Reporting Requirements

5.6.5 Steam Generator Tube Inspection Report (continued)

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. NPF-16 FLORIDA POWER & LIGHT COMPANY (FPL) ST. LUCIE NUCLEAR PLANT, UNIT NO. 2 (ST. LUCIE, UNIT NO. 2) DOCKET NO. 50-389

Application (i.e., initial and supplements)	
• September 11, 2024 (ML24255A118) • May 29, 2025 (ML25150A005)	Principal Contributors to Safety Evaluation • Robert Beaton • Richard Fu • Logan Gaul • Kevin Heller

1.0 PROPOSED CHANGE

By letter L-2024-138 dated September 11, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24255A118), as supplemented by letter L-2025-108 dated May 29, 2025 (ML25150A005), Florida Power & Light Company (FPL or the licensee) submitted a license amendment request (LAR) to Renewed Facility Operating License No. NPF-16 for St. Lucie Nuclear Plant, Unit No. 2 (St. Lucie, Unit No. 2). The LAR requested revisions to technical specifications (TS) to update the listing of U.S. Nuclear Regulatory Commission (NRC)-approved analytical methods used to determine the core operating limits. Specifically, changes to the fuel thermal-mechanics, core thermal-hydraulics, emergency core cooling, nuclear design, and select design-basis event analyses are proposed using NRC-approved advanced codes and methods in support of a St. Lucie, Unit No. 2, transition to 24-month fuel cycles.

From February 24, 2025, through July 15, 2025, the NRC staff conducted a regulatory audit to support its review of the LAR, as discussed in the staff's audit plan dated February 18, 2025 (ML25041A239), and audit summary dated September 16, 2025 (ML25251A079).

2.0 REGULATORY EVALUATION

2.1 Proposed TS Changes

The licensee's proposed changes to TS 5.6.3, "Core Operating Limits Report," are described below where additions are in **bold red text** while deletions are in ~~strikethrough text~~.

Enclosure 2

TS 5.6.3, "CORE OPERATING LIMITS REPORT"

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

49. **ANP-10297P-A, Revision 0, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," February 2013.** ~~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.~~

50. **ANP-10338P-A, Revision 0, "AREA™ - ARCADIA® Rod Ejection Accident," December 2017.**~~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.~~

51. **ANP-10297, Revision 0, Supplement 1P-A, Revision 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," December 2020.**~~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).~~

63. EMF-2103(P)(A) ~~Revision 0~~**Revision 3**, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," ~~April 2003~~**June 2016**.

65. EMF-2328(P)(A) Revision 0, Supplement 1(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016.

66. ANP-10349P-A Revision 0, "GALILEO Implementation in LOCA Methods," November 2021.

67. ANP-10311 P-A, Revision 1, COBRA-FLX: A Core Thermal-Hydraulic Analysis Code," October 2017.

68. ANP-10311, Revision 1, Supplement 1P-A, Revision 0, "COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation," March 2023.

69. ANP-10339P-A, Revision 0, "ARITA - ARTEMIS/RELAP Integrated Transient Analysis Methodology," October 2023.

70. BAW-10227P-A, Revision 2, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," January 2023.

71. [Not Used]

72. ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018.

73. ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," November 2020.

74. BAW-10084P-A, Revision 3, "Program To Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995.

2.2 Applicable Regulations and Guidance

The NRC staff considered the following regulatory requirements and guidance during its review of the LAR.

Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.36, "Technical specifications," provides regulatory requirements related to TSs. Pursuant to 10 CFR 50.36(c)(1)-(5), TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. Specifically, the regulation in 10 CFR 50.36(c)(5) requires, in part, that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner.

Key regulatory requirements specified in 10 CFR 50.46(a) that are relevant to the proposed license amendment include:

- Each pressurized light-water reactor (LWR) fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must perform analysis of core cooling performance under postulated loss-of-coolant accident (LOCA) conditions using an acceptable evaluation model (EM).

- An acceptable LOCA EM must be used that either applies realistic methods with an explicit accounting for uncertainties or follows the prescriptive, conservative requirements of Appendix K, "ECCS [emergency core cooling system] Evaluation Models," to 10 CFR Part 50.

- Core cooling performance must be analyzed for a number of postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

The following ECCS acceptance criteria in 10 CFR 50.46(b)(1) though (b)(5) state, in part:

1. *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2,200°F [degrees Fahrenheit].

2. *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. . .

3. *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The acceptance criteria in 10 CFR 50.46(b)(1) through (5) above are referred to as the peak cladding temperature (PCT) criterion, the maximum local oxidation (MLO) criterion, the hydrogen generation (or core wide oxidation (CWO)) criterion, the coolable geometry criterion, and the long-term cooling criterion, respectively.

The 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” applicable to this LAR are as follows:

- General Design Criterion (GDC) 11, “Reactor inherent protection,” states that “[t]he reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.”

- GDC 12, “Suppression of reactor power oscillations,” states that “[t]he reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.”

- GDC 27, “Combined reactivity control system capability,” states, in part, 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.

- GDC 28, “Reactivity limits,” states, in part, that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

- GDC 35, “Emergency core cooling,” states:

  A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any

> loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Appendix K to 10 CFR Part 50 establishes required and acceptable features of EMs for heat removal by the ECCS after the blowdown phase of a LOCA. It consists of the following two parts:

- required and acceptable features of LOCA EMs and,
- documentation required for LOCA EMs.

The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic (T-H) behavior.

The second part specifies requirements for the documentation of LOCA EMs, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

<u>Regulatory Guidance</u>

The NRC staff relied on the following sections of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition” (SRP), in its review of this LAR:

- Section 4.2, “Fuel System Design,” Revision 3, March 2007
- Section 4.4, “Thermal and Hydraulic Design,” Revision 2, March 2007
- Section 15, “Introduction – Transient and Accident Analyses,” Revision 3, March 2007

Regulatory Guide (RG) 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” Revision 0, June 2020, details acceptable methods and procedures to use when analyzing a postulated control rod drop accident.

RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, July 2000, provides methods acceptable to the NRC on acceptable applications of alternative source terms, including the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and acceptable radiological analysis assumptions for use in conjunction with the accepted alternate source term.

Generic Letter (GL) 86-06, “Implementation of TMI [Three Mile Island] Action ITEM II.K.3.5, ‘Automatic Trip of Reactor Coolant Pumps,’” dated May 29, 1986, discusses the staff's conclusions regarding the Combustion Engineering Owners Group (CEOG) submittals on reactor coolant pump trip in response to GLs 83-10a and b, "Resolution of TMI Action ITEM II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,'" and provides guidance concerning implementation of the reactor coolant pump trip criterion.

2.3 Description

St. Lucie, Unit No. 2, is a Combustion Engineering (CE) pressurized-water reactor (PWR) with two hot legs, two steam generators, and four cold legs. The St. Lucie, Unit No. 2, reactor core is fueled with UO_2 and UO_2-Gd_2O_3 pellets enclosed in zircaloy tubes pressurized with helium and fitted with welded end plugs. The feed fuel is a Framatome CE-16 HTP™ fuel design, which consists of a 16x16 assembly configuration with M5 clad fuel rods, Zircaloy-4 MONOBLOC Corner Guide tubes, an Alloy 718 HMP spacer at the lowermost axial elevation, Zircaloy-4 HTP spacers in all other axial elevations, a FUELGUARD™ lower tie plate, and the AREVA reconstitutable upper tie plate.

The LAR proposes revisions to St. Lucie, Unit No. 2, TS 5.6.3, "Core Operating Limits Report," by updating the listing of NRC-approved analytical methods used to determine the core operating limits. Specifically, changes to the fuel thermal-mechanics, core thermal-hydraulics, emergency core cooling, nuclear design, and select design-basis event analyses are proposed using NRC-approved advanced codes and methods in support of a St. Lucie, Unit No. 2, transition to 24-month fuel cycles. This includes the addition of 13 new analytical methods, including:

- ANP-10297P-A, Revision 0, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," February 2013
- ANP-10297, Revision 0, Supplement 1P-A, Revision 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," December 2020
- ANP-10311 P-A, Revision 1, "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code," October 2017
- ANP-10311, Revision 1, Supplement 1P-A, Revision 0, "COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation," March 2023
- ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," November 2020
- ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018
- ANP-10338P-A, Revision 0, "AREA™ -ARCADIA® Rod Ejection Accident," December 2017
- ANP-10339P-A, Revision 0, "ARITA-ARTEMIS/RELAP Integrated Transient Analysis Methodology," October 2023
- ANP-10349P-A Revision 0, "GALILEO Implementation in LOCA Methods," November 2021
- BAW-10084P-A, Revision 3, "Program To Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995
- BAW-10227P-A, Revision 2, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," January 2023

- EMF-2103(P)(A) Revision 3, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” June 2016
- EMF-2328(P)(A) Revision 0 Supplement 1(P)(A) Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” December 2016

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee’s application to determine whether the proposed changes are consistent with the guidance, regulations, and plant-specific design and licensing basis discussed in Section 2.2 of this safety evaluation (SE). The NRC staff reviewed the licensee’s statements in the LAR, attachments to the LAR, and the relevant sections of the St. Lucie, Unit No. 2, Updated Final Safety Analysis Report (UFSAR). The staff reviewed the limitations and conditions for all proposed topical reports (TRs) to ensure they are met satisfactorily.

The NRC staff’s technical evaluation is organized into the following sections as follows:

- Fuel design is discussed in Section 3.1
- LOCAs are discussed in Section 3.2
- Non-LOCA transients are discussed in Section 3.3
- Control rod ejection accident is discussed in Section 3.4

3.1 Fuel Design

The fuel design is described in Framatome Technical Report ANP-4090P, “St. Lucie Unit 2 24-Month Fuel Analysis License Amendment Request,” Revision 1, August 2024 (Reference 26).

The fuel system design bases must reflect four objectives: (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. The LAR evaluates mechanical design-related aspects of the fuel system design basis.

3.1.1 Mechanical Analyses

The Framatome CE-16 HTP fuel assembly design for St. Lucie, Unit No. 2, was analyzed in accordance with the NRC-approved generic mechanical design criteria in EMF-92-116(P)(A) (Reference 12) in conjunction with NRC-approved TRs BAW-10240(P)(A) (Reference 13), and ANP-10337P-A (Reference 14). BAW-10240(P)(A) (Reference 13) incorporates the M5 cladding material properties into the EMF-92-116(P)(A) mechanical design methodology (Reference 12). Both TRs are currently included in the St. Lucie, Unit No. 2, TSs as approved core operating limits methodologies. ANP-10337P-A (Reference 14) is being added to the St. Lucie, Unit No. 2, TSs and is discussed further below. All the mechanical design criteria were shown to be met up to the licensed fuel rod burnup limit of 62 Gigawatt Days per Metric Ton Uranium (GWd/MTU).

The mechanical design evaluations are performed using the NRC-approved design methods and evaluated to the NRC approved generic design criteria (Reference 12). These generic criteria are consistent with the specified acceptable fuel design limits (SAFDLs) identified in Section 4.2 of the SRP (Reference 15).

Framatome performed fuel seismic evaluations as part of the 24-month cycle project. The NRC-approved methodology defined in ANP-10337P-A (Reference 14) was used for the evaluations. This method was generically approved for PWRs and was noted explicitly as applying to CE 14x14 and 16x16 HTP fuel designs.

The ANP-10337P-A methodology was reviewed for applicability to St. Lucie, Unit No. 2, along with any limits and conditions from the NRC's SE. Framatome determined that none of the limits and conditions prevent application to St. Lucie, Unit No. 2.

<u>Limitation and Condition (L&C) 1</u>:

> Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1 of ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in the TR, the key elements of which are:
> **[[**
>
> **]]**

<u>Licensee evaluation</u>:

> Dynamic grid crush tests were performed in compliance with this requirement.
>
> **[[**
>
> **]]**

The NRC staff finds this L&C is met through the use of dynamic grid crush tests and subsequent processing of test data.

Limitation and Condition 2:

> For fuel assembly designs where spacer grid applied loads are limited based on allowable grid permanent deformation (as opposed to buckling), the following limits from Table 4-1 of the TR [ANP-10337P] apply:
>
> a. For all OBE [operating basis earthquake] analyses, allowable spacer grid deformation is limited to design tolerances and must permit functional gage insertion.
>
> b. For SSE [safe shutdown earthquake], LOCA, and combined SSE+LOCA analyses, allowable spacer grid deformation limited to 1mm for interior assemblies and 3mm for exterior assemblies, uniformly distributed, measured from face-to-face. The ultimate or buckling load must not be exceeded. The deformation must also permit functional gage insertion.

Licensee evaluation:

> Under combined OBE and SSE/LOCA excitations, grid impact loads have been determined to keep within the elastic limits and limits defined by 1 mm of deformation, respectively.

The NRC staff finds that this L&C is met through demonstration that grid deformation meets the requirements.

Limitation and Condition 3:

> The modification or use of the codes CASAC and ANSYS (or other similar industry standard codes) are subject to the following limitations:
>
> a. CASAC computer code revisions, necessitated by errors discovered in the source code, needed to return the algorithms to those described in ANP-10337P (as updated by RAIs) are acceptable.
>
> b. Changes to CASAC numerical methods to improve code convergence or speed of convergence, transfer of the code to a different computing platform to facilitate utilization, addition of features that support effective code input/output, and changes to details below the level described in ANP-10337P would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used in licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B.
>
> c. ANSYS or other industry standard codes may be used if they are documented in an auditable manner to meet the quality assurance

requirements of 10 CFR Part 50, Appendix B, including the appropriate verification and validation for the intended application of the code.

Licensee evaluation:

The applicable CASAC versions used are fully consistent with requirements "a" and "b". ANSYS or other industry codes were not used in this analysis.

The NRC staff finds that this L&C is met through licensee assurances that codes used meet the criteria of the L&C.

Limitation and Condition 4:

This methodology is limited to applications that are similar to the current operating fleet of PWR reactor and fuel designs. The core geometry should be comparable to the current fleet, in terms of dimensions, dimension tolerances, fuel assembly row lengths, and the gaps between fuel assemblies. Fuel designs should be comparable to the current fleet, in terms of materials, geometry, and dynamic behavior.

Licensee evaluation:

The St. Lucie Unit 2 reactor is part of the "current fleet" of PWR reactors in place at the time of approval of ANP-10337P-A.

The NRC staff finds that this L&C is met as St. Lucie, Unit No. 2, was part of the "current fleet" at the time the referenced methodology was adopted.

Limitation and Condition 5:

ANP-10337P established generic fixed damping values intended to be used for all PWR designs. All applications of this methodology to new fuel assembly designs must consider the continued applicability of the fixed damping values of this methodology. If new materials, new geometry, or new design features of a new fuel assembly design may affect damping, additional testing and/or evaluation to determine appropriate damping values may be required.

Licensee evaluation:

The damping values per ANP-10337P-A are applicable to the CE16 HTP Design.

The NRC staff finds the L&C is met as the damping values provided in the referenced methodology are applicable to the CE16 HTP design.

Limitation and Condition 6:

The ANP-10337P methodology includes the generation of fuel rod loads but does not provide a means to demonstrate compliance for fuel rod performance under externally applied loads (to applicable acceptance criteria). Applications of this methodology must provide an acceptable demonstration of fuel rod performance.

<u>Licensee evaluation</u>:

> Fuel rod performance under faulted conditions is in compliance. Fuel rod cladding stress intensity limits derived from American Society of Mechanical Engineers (ASME) Sec III Division 1; subsection NB based on biaxial strength at operating temperature with modification made for M5 cladding per BAW-10227P-A, Revision 2. [Reference 10].

NRC staff has reviewed the licensee's description of the relevant analysis and found that this L&C has been satisfied.

<u>Limitation and Condition</u>:

> As indicated in ANP-10337P when orthogonal deflections from separate core locations are artificially superimposed to calculate component stresses, the component stresses must be compared against the design criteria associated with control rod positions.

<u>Licensee evaluation</u>:

> Orthogonal deflections from same core locations are superimposed to calculate component stresses. The component stresses are analyzed against the appropriate service limits for the locations considered.

NRC staff agrees the licensee approach satisfies the L&C.

<u>Limitation and Condition 8</u>:

> In accordance with RG 1.92, the combination of loads for non-grid component evaluation should ideally be based on three orthogonal components (two horizontal and one vertical). When a 2D combination of loads as described in Section 8.1.2 Item i of ANP-10337P is performed, this must be consistent with the licensing basis of the plant to which it is being applied.

<u>Licensee evaluation</u>:

> The analysis performed is in accordance with RG 1.92 and combines loads based on three-orthogonal components.

The NRC staff finds this L&C to be satisfied based on the licensee approach to the combination of loads for no-grid components.

<u>Limitation and Condition 9</u>:

> The linear viscoelastic grid impact model of this methodology is limited to impact forces that correspond to permanent grid deformations no greater than 1mm.

Licensee evaluation:

> Under combined seismic and LOCA excitations grid impact loads have been determined to keep within the limits defined by 1 mm of deformation.

The NRC staff finds 24-Month cycle design is in full compliance with this L&C.

The mechanical criteria (SAFDLs) for the fuel rod and fuel assembly are listed in Table 2-2 of Enclosure 2 in this LAR with the corresponding section number from the criteria TR (Reference 12) and with the results from the Framatome 24-month cycle analyses. These results are based on Framatome's representative 24-month cycle core designs. Each of these criteria are either reanalyzed or confirmed for every reload cycle, so future results could vary depending on cycle-specific inputs or other input changes.

The NRC staff finds that The Framatome CE-16 HTP fuel design has been analyzed in accordance with NRC-approved mechanical design criteria using representative 24-month cycle inputs. All the design criteria were shown to be met up to the licensing fuel rod burnup of 62 GWd/MTU under normal and faulted operating conditions.

3.1.2 Thermal-Mechanical Analyses

In compliance with the SAFDLs identified in Section 4.2 of the SRP (Reference 15) fuel rod thermal-mechanical performance is evaluated using the NRC-approved codes and methodologies (C&Ms) documented in References 16 through 19. Revision 2 of the M5 topical report (Reference 16), GALILEO (Reference 17), and CROV (Reference 18) are part of the Framatome advanced C&Ms and are being added to the St. Lucie, Unit No. 2, TS listing of approved core operating limits methodologies. Their applicability to St. Lucie, Unit No. 2, is discussed below. ARITA (Reference 11) is also being added to the St. Lucie, Unit No. 2, TS and is discussed in the non-LOCA ARITA section (see Section 3.3 of this SE).

The methodologies described in Revision 2 of the M5 topical report (Reference 16), GAILILEO (Reference 17), CROV (Reference 18), ARITA (Reference 19), ARCADIA (nuclear simulator, Reference 20), and AREA (Reference 21) are collectively referred to as a suite of Advanced Codes and Methods. Brief descriptions of GALILEO (Reference 17), Revision 2 of M5 (Reference 16), and CROV (Reference 18) are provided in the following sections along with justification for their application to St. Lucie, Unit No. 2.

GALILEO ANP-10323P-A, Revision 1 (Reference 17), is used to perform calculations to demonstrate compliance with the rod internal pressure and cladding corrosion criteria per SRP Section 4.2 (Reference 15). GALILEO is also used to initialize fuel rod data for the downstream evaluation of the cladding collapse (CROV) and fatigue criteria according to the requirements established in Section 4.2 of the SRP. In addition, GALILEO is used to provide initial conditions for other fuel methodologies, such as AREA (discussed in Section 3.4 of this SE) and the LOCA analyses (discussed in Section 3.2 of this SE).

> **[[**

]].

Section 5.0 of the GALILEO SE lists four (4) topical report L&Cs. Adherence measure for each of the L&Cs is evaluated as follows. GALILEO is generically approved by the U.S. NRC for use in PWR designs using Low Enriched Uranium (LEU) fuel loading and is therefore applicable to St. Lucie, Unit No. 2.

Limitation and Condition 1:

> The application of GALILEO should assume fuel failure when the predicted fuel temperatures exceed the fuel melting temperatures as calculated by GALILEO due to the lack of properties for molten fuel in GALILEO and other properties such as thermal conductivity and fission gas release.

Licensee evaluation:

> Overheating of fuel pellets is addressed in Framatome's ARITA and AREA Summary.

The NRC staff agrees this L&C will be evaluated in the analyses where it is applicable.

Limitation and Condition 2:

> The ability to make changes to both the mean and standard deviation of model parameter uncertainty values without NRC review and approval is not approved. Because of the complex interaction between parameters in fuel performance codes, the NRC staff does not approve the ability to make changes to the model parameters without NRC approval.

Licensee evaluation:

> The applicable model uncertainties are consistent with the GALILEO topical report (i.e. no changes).

The NRC staff agrees this L&C is met because the model uncertainties are unchanged.

Limitation and Condition 3:

> The peak axial node burnup for the fuel rod is limited to **[[ ]]** GWd/MTU.

[1] Condition I and Condition II events are defined in the SE for the GALILEO Revision 1 Safety Evaluation (Reference 5). These Conditions are designed to model AOOs, and adjust local peaking factors while keeping the rod average power unchanged to evaluate these AOOs. Condition I events skew the steady-state axial power shape to a local peaking factor of 1.5 for 4 hours, while Condition II events skew the steady-state axial power shape to a local peaking factor of 1.7 for 1 minute.

Licensee evaluation:

> Peak nodal burnups remain below **[[ ]]** GWD/MTU for analyses that support fuel rod SAFDL compliance.

The NRC staff agrees the licensee will ensure peak nodal burnups will remain below **[[ ]]** GWD/MTU.

Limitation and Condition 4:

> No methodology has been approved for providing initial data or conditions for ECCS analysis.

Licensee evaluation:

> Emergency Core Cooling System (ECCS) analyses is not part of the fuel rod analysis. **[[ ]]**

The NRC staff agrees the L&C is met for fuel design consideration, and the use of GALILEO for LOCA analysis will be addressed in Section 3.2 of this SE.

BAW-10227P-A, Revision 2 (Reference 16), provides updates to the M5 material characteristics. As such, material models of physical properties, mechanical behavior, oxidation and hydrogen pick-up fractions, irradiation growth and creep are defined. In addition, BAW-10227P-A, Revision 2 supports M5 as a fuel rod cladding material to **[[ ]]**

Section 4.0 of the M5 topical report SER (Reference 16) lists one L&C.

Limitation and Condition 1:

> When applying the methodology described in BAW-10227P, Revision 2, **[[ ]]** licensees shall ensure that changes to expected fatigue cycles are appropriately captured in the fatigue evaluation.

Licensee evaluation:

> SLU2 fuel rod average burnup is limited to **[[ ]]** and is inherently accounted for in the fatigue analysis.

The NRC staff finds this L&C to be satisfied because the methods adopted by the licensee are limited to **[[ ]]** and the licensee does not have approval to exceed this burnup limit and is inherently accounted for in the fatigue analysis.

BAW-10084P-A, Revision 3 (Reference 18), defines the criteria and methodology for cladding creep collapse. GALILEO is used to initialize fuel rod data for the cladding collapse evaluation and the CROV code simulates cladding creep-down deformations as a function of time (exposure of the fuel rod). CROV has been approved by the NRC for use with Framatome fuel designs with M5 cladding (Reference 16).

There is no explicit section for limitations and conditions within the [Technical Evaluation Report] TER for CROV (Reference 18); however, per Section 5.0 of the TER, there is one L&C to consider. The limitation and condition states that the cladding temperature limit shall be less than or equal to 700°F (for the strain rate equation to be applicable). The St. Lucie, Unit No. 2, creep collapse analysis maintained cladding temperatures that were compliant with the licensing limit.

Therefore, the NRC staff finds that this L&C is met.

Thermal-Mechanical fuel rod analyses results are based on Framatome's 24-month cycle project work. Each of these criteria are reanalyzed or confirmed for every reload cycle, so future results could vary depending on cycle-specific inputs or other input changes.

In conclusion, fuel rod thermal-mechanical performance has been analyzed in accordance with NRC-approved design criteria and methodologies using representative cycle design inputs. All the design criteria were shown to be met up to the licensing fuel rod average burnup of 62 GWd/MTU under normal, AOO, and faulted operating conditions.

3.1.3 Thermal-Hydraulic Analyses

The existing NRC-approved methods for thermal-hydraulic (T-H) analyses for St. Lucie, Unit No. 2, as specified in the COLR reference list in TS 5.6.3, continue to be applicable as the physical fuel assembly and core design configuration remains unchanged with the transition to 24-month cycles, resulting in no remarkable changes to the thermal-hydraulic response of the fuel.

In compliance with the SAFDLs identified in Section 4.2 of the SRP, T-H evaluations are performed using the NRC-approved C&Ms.

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

- There is at least a 95% probability at a 95% confidence level that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during Modes 1 (power operation) and 2 (startup), operational transients, or any condition of moderate frequency.
- No fuel melting during any anticipated normal operating condition, operational transients, or any conditions of moderate frequency.

The hydraulic characteristics of the 24-month cycle fuel design are unchanged from the recent fuel deliveries to St. Lucie, Unit No. 2. Therefore, parameters related to hydraulic compatibility are not impacted by the 24-month cycle fuel design. Rod bow and setpoint analyses are included in the ARITA methodology (discussed in Section 3.3 of this SE), and the AREA methodology (discussed in Section 3.4 of this SE) demonstrate compliance with SAFDLs and represent Framatome advanced C&Ms being added to the St. Lucie, Unit No. 2, TS listing of approved core operating limits methodologies.

The NRC staff finds that the use of existing T-H methodologies is acceptable for the transition to 24-month cycles, with the inclusion of ARITA (see Section 3.3 of this SE) and AREA (see Section 3.4 of this SE) methodologies.

3.1.4 Nuclear Design

The effects of transitioning to Framatome's ARCADIA code system on the nuclear design bases and methodologies for St. Lucie, Unit No. 2, are evaluated by the licensee, as demonstrated on the implementation of 24-month cycle. The approved Framatome neutronics methodology and primary physics codes used for these analyses are described in References 21, 23, and 24. These codes and their Topical Reports are part of the Framatome advanced C&Ms and are being added to the St. Lucie, Unit No. 2, TS listing of approved core operating limits methodologies. The nuclear design codes are based on the ARCADIA code system. References 21, 23, and 24 are the U.S. NRC-approved ARCADIA Topical Reports outlining the approved Framatome neutronics methodology and codes.

The Limits and Conditions (L&C) section of Reference 21 contains six L&Cs, with the third L&C that was originally presented in Reference 23 removed. Each of these L&C's and the requirements of the ARCADIA topical report are addressed below.

Limitation and Condition 1:

> The range applicability of the ARCADIA code system Methodology is restricted to the fuel data provided in the TR, unless additional analysis and benchmarking are conducted to validate the ARCADIA code system to a fuel type not mentioned in the TR.

Licensee Evaluation:

> The licensee stated that although not specifically in the context of St. Lucie, Unit No. 2, CE 16x16 lattice is discussed in Reference 20.

The NRC staff finds that this L&C is met.

Limitation and Condition 2:

> The benchmarks provided in the TR include uncertainty verification for plants that use moveable incore, Rh fixed incore, and Aeroball incore detectors. AREVA will continue to monitor its methods with respect to current cycle designs for its licensing applications. Prior to licensing a new contract, AREVA will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with AREVA fuel with ARCADIA. In addition, application of the ARCADIA® code system to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation. This includes verification of their measurement uncertainties and/or calculation uncertainties by using the appropriate method presented in Section 12 of the TR.

Licensee Evaluation:

> The licensee stated that St. Lucie, Unit No. 2, uses a rhodium fixed incore measurement system which was included in the topical report and an uncertainty analysis was performed. The benchmarking demonstrates acceptable results, which provides evidence that ARCADIA can successfully model the St. Lucie, Unit No. 2, cores, and that

> this capability or proficiency directly encompassed the continued use of Framatome's CE-16 HTP fuel as a result to its similarities to other fuel designs described in the topical report.

The NRC staff satisfies that this L&C is met.

Limitation and Condition 3:

This L&C was removed in Reference 20.

Limitation and Condition 4:

> For any changes made to the stand-alone version of COBRA-FLX that is implemented in the ARCADIA code system (the COBRA-FLX module), AREVA will revalidate the ARCADIA code system output using measured data from multiple plants and cycles.

Licensee Evaluation:

> The licensee stated that for each new release of the ARCADIA code system a change review and validation of the codes is performed. This review is documented and provided to all users of the ARCADIA codes. The review document clearly identifies which features and models are not allowed for licensing analyses.

The NRC staff finds that the ARCADIA modules and codes used in the St. Lucie, Unit No. 2, analyses comply with this L&C.

Limitation and Condition 5:

> The NRC staff finds ARTEMIS™ acceptably models the best estimate neutronic time dependent transient responses (e.g., power response to changes in Doppler, moderator, etc.), and that it is an acceptable tool for use in an evaluation model for non-LOCA SRP Chapter 15 events. However, use of ARTEMIS™ in an evaluation model for such events requires consideration of bounding conditions, inputs, limits, time-step sensitivities, etc., which are not included in Supplement 1. Therefore, as implied for ANP-10297P-A, Revision 0, this SE does not constitute approval of ARTEMIS™ as a stand-alone evaluation model for non-LOCA SRP Chapter 15 events. NRC review and approval of an associated evaluation methodology using ARTEMIS™ is required prior to its use in non-LOCA SRP Chapter 15 event licensing analyses.

Licensee Evaluation:

> The licensee stated that for each new release of the ARCADIA code system a change review and validation of the codes is performed. This review is documented and provided to all users of the ARCADIA codes. The review document clearly identifies which features and models are not allowed for licensing analyses.

The NRC staff finds that the ARCADIA modules and codes used in the St. Lucie Unit 2 analyses comply with this L&C.

Limitation and Condition 6:

Any changes made to the ARCADIA® code system must:

a. ensure the validation suite acceptance criteria (Table 10-2 of Supplement 1) remain applicable,
b. be consistent with the methodology described in ANP-10297P, as supplemented, and
c. not invalidate the NRC staff's SE.

In instances where it is unclear if a change is consistent with the approved methodology, Framatome may submit descriptions of a change to the NRC for confirmation that the change is within the scope of the approved methodology, as discussed in Section 3.9.3 of this SE.

Licensee Evaluation:

The licensee stated that for each new release of the ARCADIA code system a change review and validation of the codes is performed. This review is documented and provided to all users of the ARCADIA codes. The review document clearly identifies which features and models are not allowed for licensing analyses.

The NRC staff finds that the ARCADIA modules and codes used in the St. Lucie, Unit No. 2, analyses comply with this L&C. Application of this L&C resulted in Framatome receiving NRC confirmation that a change to the ARCADIA code system was within the scope of approved methods as given by the NRC in Reference 23.

Key parameters are calculated and verified as part of the neutronics core design analysis using the ARCADIA code system. Margin to key safety parameter limits (Table 5-3 of Enclosure 2 in the LAR) is maintained during the St. Lucie, Unit No. 2, transition from 18-month to 24-month operating cycles, with the evaluations performed based on the NRC approved analytic C&Ms (i.e., ARCADIA code system - References 21, 23 and 24) and the continued utilization of the Framatome CE-16 HTP fuel. The NRC staff reviewed Table 5-3 and additional details provided during the regulatory audit and found that the margin to key safety parameter limits is not significantly impacted or reduced by the implementation of 24-month operating cycles in St. Lucie, Unit No. 2. The key safety parameters evaluated for St. Lucie, Unit No. 2, as it transitions to 24-month operating cycles show little, expected, and/or predictable change relative to the current reload designs. The variations in these parameters are similar to the normal cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle. Changes to the core power distributions and peaking factors are also within the normal cycle-to-cycle variations expected in core loading patterns.

In conclusion, a comprehensive nuclear core design evaluation, specific to St. Lucie, Unit No. 2, is performed with the NRC-approved Framatome ARCADIA code system. This code system has been confirmed to be applicable to St. Lucie, Unit No. 2. The nuclear core design analysis of the submittal core design(s) for the St. Lucie, Unit No. 2, implementation of 24-month operating cycles with the Framatome St. Lucie CE-16 HTP fuel has confirmed that peaking factor and key safety parameters can be maintained within their specified limits using Framatome methodologies and codes.

The key safety parameters generated with the submittal core design were used in the applicable analyses and evaluated to meet the acceptance criteria. The NRC staff finds that the analyses performed have adequately accounted for the effects of the proposed implementation of 24-month operating cycles at St. Lucie, Unit No. 2, on the nuclear design and has demonstrated that the fuel design limits will not be exceeded. Therefore, 24-month cycles were analyzed against the existing UFSAR safety analyses, and it was determined that the safety analyses will continue to be met with ample Nuclear Design margins to accommodate the implementation or transition to 24-month operating cycles, including cycle-to-cycle variations, at St. Lucie, Unit No. 2.

3.1.5 Fuel Design Conclusion

St. Lucie, Unit No. 2, 24-Month fuel analysis LAR TR justifies use of Framatome's NRC-approved advanced C&Ms for use at St. Lucie, Unit No. 2, and shows acceptability for 24-month cycle operation of the Framatome CE-16 HTP. Demonstration of the evaluation methodologies has been performed with a representative core design, which was developed to provide key safety parameters to support the Framatome analyses. This provides assurance that the plant licensing bases are met for the continued operation with Framatome CE-16 HTP fuel and 24-month cycles.

3.2 Loss-of-coolant accidents

The NRC regulations require that licensees of operating LWRs analyze a spectrum of accidents including LOCAs to assure adequate core cooling under the most limiting set of postulated design-basis conditions. LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant system (RCS) primary boundary at a rate in excess of the reactor coolant make-up 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 small and large-break LOCA analysis are discussed below in Sections 3.2.1 and 3.2.2, respectively.

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

The licensee performed an SBLOCA analysis as documented in Framatome report ANP-4055P, "St. Lucie Unit 2 24-Month Cycle Small Break LOCA Analysis," Revision 0, November 2023.

3.2.1.1 SBLOCA Description

The postulated SBLOCA is defined in the methodology as a break in the RCS pressure boundary with an area less than or equal to 10 percent of the cold leg pipe area. The reactor protection system (RPS) and ECCS are provided to mitigate these accidents. The most limiting break location for SBLOCA analysis performed is in the cold leg pipe on the discharge side of the reactor coolant pump (RCP). This break location results in the largest amount of RCS inventory loss, the largest fraction of ECCS fluid discharged out the break, and the largest pressure drop between the core exit and the top of the downcomer. The SBLOCA event progression develops in the following distinct phases: (1) subcooled depressurization (also known as blowdown), (2) natural circulation, (3) loop seal clearing, (4) core boil-off, and (5) core recovery and long-term cooling. The duration of each of these phases is break size and system dependent. The licensee provided a detailed description of each of the phases in Framatome report ANP-4055.

3.2.1.2 SBLOCA Methodology

The licensee performed the SBLOCA analysis using the NRC-approved SBLOCA methodology documented in TR EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001, Supplement 1(P)(A) to TR EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016, and TR ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods," November 2021. The SBLOCA EM uses a deterministic approach based on the requirements of 10 CFR 50 Appendix K to determine the expected PCT, MLO, and CWO response. The licensee used the evaluation model for event response of the primary and secondary systems as well as the hot fuel rod. The licensee used the following two computer codes:

- The GALILEO code to determine the burnup dependent initial fuel rod conditions for the system calculations.
- The S-RELAP5 code to predict the primary and secondary system T-H and hot rod transient response.

The S-RELAP5 code is used in the NRC-approved SBLOCA methodology as documented in TR EMF-2328(P)(A). The use of S-RELAP5 and the RODEX2A fuel performance code (FPC) is specified for SBLOCA analysis as described in Supplement 1 to TR EMF-2328(P)(A). However, in TR ANP-10349P-A, Framatome supplemented the approved EMs and implemented the GALILEO FPC in S-RELAP5. In the SE to TR ANP-10349P-A, the NRC staff concluded that the GALILEO FPC is an acceptable supplement for RODEX2A for SBLOCA evaluation models. The NRC staff, therefore, finds the GALILEO and S-RELAP5 codes appropriate for use with the applied methods.

3.2.1.3 SBLOCA Analysis

The goal of the analysis is to demonstrate that the ECCS will continue to satisfy the acceptance criteria given in 10 CFR 50.46(b)(1) through (b)(4). A break spectrum analysis for SBLOCA was performed for breaks of varying diameters of up to 10 percent of the flow area for the cold leg pump discharge consistent with the EM. The spectrum analyzed included a break size range from 1.0 to 9.49 inches in diameter, with a break size interval sufficient to establish a PCT trend.

In addition to the cold leg pump discharge break spectrum analysis, the licensee performed sensitivity studies as required by the methodology for a delayed RCP trip, a break in an attached pipe, ECCS temperature, and a **[[ ]]** For the delayed RCP trip, an operator action time of 2 minutes following event initiation is analyzed to evaluate the adequacy of the specified trip criteria. During the regulatory audit, the staff asked about the basis for the operator action time of 2 minutes. The licensee stated that it was based on recommendations in CEOG Report CEN-268, Revision 1, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients," dated May 1987 (ML20214V685). GL 86-06, titled "Implementation of TMI Action ITEM II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'" (ML031150282), states that the CEOG studies have shown that every Combustion Engineering plant's UFSAR ECCS analysis demonstrates compliance with 10 CFR 50.46 if operator action to trip the RCPs is taken within 2 minutes after the RCP trip criterion is reached. Based on its review of the above information, the NRC staff finds that the delayed RCP trip time of 2 minutes is acceptable.

The licensee performed an analysis of the ruptures in attached piping that compromise the ability to inject emergency coolant into the RCS. The piping study analyzed breaks in the accumulator line and high-head safety injection (HHSI) line. The ECCS temperature sensitivity study analyzed the sensitivity to ECCS fluid temperatures different from those used in the break spectrum analysis. **[[ ]]**

The licensee performed SBLOCA analysis to support plant operation at a core power level of 3,029.06 MWt (including measurement uncertainty), a peak linear heat generation rate (LHGR) of 13.0 kW/ft, a radial peaking factor of (F_r) of 1.81 (1.65 plus uncertainty), and up to 20 percent steam generator (SG) tube plugging per SG.

3.2.1.4 SBLOCA Results

The licensee's SBLOCA break spectrum analysis for St. Lucie, Unit No. 2, resulted in a limiting PCT of 1,831°F for a 2.6-inch diameter cold leg pump discharge break. The same break produced the limiting CWO of 0.14 percent. A 2.5-inch diameter cold leg pump discharge break produced the limiting MLO of 9.52 percent. The total MLO value includes **[[ ]]**

For the attached piping break study, a double-ended guillotine break of a safety injection tank (SIT) line was analyzed. This case resulted in a PCT of 1,476°F, transient MLO of 0.15 percent and CWO of 0.002 percent. These results are all less limiting than the SBLOCA break spectrum.

For the delayed RCP trip study, the spectrum of cold and hot leg breaks includes break sizes from 1.00 to 9.49 inches. The RCP trip criteria is modeled as the occurrence of the RCS pressure trip following the loss of subcooling margin. The results indicate that there is at least 2 minutes for operators to trip all four RCPs after the trip criteria is reached and still maintain margin to the acceptance criteria in 10 CFR 50.46(b)(1-4).

For the ECCS temperature sensitivity, **[[ ]]**

[[ ]]

Based on the above, the NRC staff finds the analysis results demonstrate the adequacy of the ECCS to satisfy the criteria given in 10 CFR 50.46(b)(1) to (b)(3). Further, maintaining compliance to 10 CFR 50.46(b)(1) to (b)(3) criteria also ensures the 10 CFR 50.46(b)(4) criteria on maintaining the core amenable to cooling will be satisfied.

3.2.1.5 Compliance with NRC Staff Imposed Limitations and Conditions on SBLOCA Methodologies

Section 3.5 of Framatome report ANP-4055 discusses the limitations and conditions for the SBLOCA methodology. For each of the three methodologies used in the SBLOCA analysis, the limitations and conditions for each are discussed below.

For TR EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001, there was one condition imposed on the use of S-RELAP5 as follows:

> That while it has been shown in Reference 53 [NUREG/CR-4945] that the thermal-hydraulic phenomena observed for breaks up to 10 percent of the cold leg flow area are the same, if the code is used for break sizes larger than 10 percent of the cold leg flow area additional assessments must be performed to ensure that the code is predicting the important phenomena which may occur.

The licensee performed analysis on a spectrum of cold leg break sizes from 1.0-inch diameter to 9.49-inch diameter (10 percent of cold leg pipe area). However, in the accumulator line pipe break sensitivity, the break area was equivalent to a 10.13-inch diameter (safety injection tank line which connects to the cold leg) which is larger than 10 percent of the cold leg flow area. This is addressed in Section 8.2 of Supplement 1 to EMF-2328 where a full double-ended break in SIT piping is considered a special exception for breaks in attached piping. Therefore, the NRC staff finds this limitation and condition has been met.

For Supplement 1(P)(A) to TR EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016, the SE stated:

> The NRC staff mentions that it is necessary for all SBLOCA submittals utilizing the Reference 1 [EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2012] methodology identify the critical break size, at and below which, only one loop seal clears of liquid. The NRC staff further requires that the largest small break that depressurizes to a pressure just above the SIT actuation pressure be included in the break spectrum evaluation. The **[[ ]]** diameter break increment resolution is expected to capture this particular break size; however, it is mentioned and emphasized here since it is important to locate this break size as it could be the limiting small break.

Tables 4-1 and 4-2 of ANP-4055 provide results for the cold leg pump discharge SBLOCA break spectrum. Part of these tables provides the loop seal clearing times. The tables show that the **[[**

]]

As to the accumulator actuation, a footnote in Table 4-1 of ANP-4055 states **[[**

]] Therefore, the NRC staff finds the above limitations and conditions have been met.

For TR ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods," November 2021, Section 4.0, "Limitations and Conditions," of its SE states the following:

> The demonstrated range of applicability of the methodology, specifically RLBLOCA and SBLOCA (EMF-2103 Rev 3, and EMF-2328 Rev 0 and Supplement 1) and applicable range (not related to the thermo-mechanical method) of applicability of GALILEO topical report (ANP-10323P) shall be implemented in the supplement EM (ANP-10349).

TR EMF-2103 is for the LBLOCA analysis and is discussed below in Section 3.2.2.5, "Compliance with NRC Staff Imposed Limitations and Conditions on LBLOCA Methodologies," of this SE. The range of applicability for EMF-2328 and its supplement were discussed above. The applicable range of ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," November 2020, are provided in Section 1.2, Limits of Code Applicability," of its SE. Table 3.2.1 below describes each item along with how the licensee meets the applicable conditions.

Table 3.2.1 – Range of Applicability for ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors"

Range of Applicability for ANP-10323P-A, Revision 1	**Licensee response**
Pressurized water reactor designs using Low-Enriched Uranium (LEU) fuel loading	This analysis was performed for the St. Lucie, Unit No. 2, plant, which is a PWR, using LEU fuel.
Rod average burnups up to **[[ ]]** gigawatt-days per metric ton of uranium (GWd/MTU) for Zircaloy-4 and up to **[[ ]]** GWd/MTU for M5 cladding	The fuel burnups applied in this analysis do not exceed the rod average burnup of **[[ ]]**
Zircaloy-4 and M5 cladding	The analysis supports operation with $M5_{Framatome}$ cladding.
Rod diameter between **[[ ]]** mm and **[[ ]]** mm	This analysis was performed using fuel with a rod outside diameter of 9.7 mm.
Uranium ^{235}U enrichments up to 5 weight percent (wt%)	The ^{235}U enrichments applied in this analysis do not exceed 5 weight percent.
Gadolinia concentrations up to 10 wt%	Gadolinia fuel is not analyzed as part of the SBLOCA methodology. Therefore, this parameter is not subject to the limitation for this LOCA analysis.
Nominal true pellet density ranging from **[[ ]]** percent of the theoretical density of UO_2	The initial pellet density is **[[ ]]** percent of the theoretical density of UO_2.
Fuel grain sizes ranging from **[[ ]]** microns (mean linear intercept)	This analysis was performed using fuel pellets with a grain size of **[[ ]]**
Pellets manufactured by dry conversion and ammonium diuranate	The fuel pellet manufacturing process for the fuel design considered in this analysis is dry conversion and ammonium diuranate.
Fuel temperature up to the melting point to the approved burnup range	This is related to thermo-mechanical methods and is not subject to the limitation for this LOCA analysis.

Range of Applicability for ANP-10323P-A, Revision 1	Licensee response
Cladding strain up to the approved transient clad strain limit	This is related to thermo-mechanical methods and is not subject to the limitation for this LOCA analysis.
Internal rod pressure up to pressures that protect from clad lift-off and hydride reorientation	This is related to thermo-mechanical methods and is not subject to the limitation for this LOCA analysis.
Fuel rod power not to exceed levels as limited by fuel melt, cladding strain, and rod pressure criteria	This is related to thermo-mechanical methods and is not subject to the limitation for this LOCA analysis.

As shown in the table above, the licensee has met all the criteria in the range of applicability for ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors." It should be noted that the last four items in the table are related to thermo-mechanical methods and are not subject to the limitation on appliable range for the LOCA analysis as stated in Section 4.0, "Limitations and Conditions," of the safety evaluation approving ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods." Therefore, based on the review of the licensee responses to the limits of code applicability summarized above, the NRC staff finds all limitations and conditions satisfied for the methodologies used in the SBLOCA evaluations.

3.2.1.6 SBLOCA Conclusion

The NRC staff reviewed the information in the licensee's submittal pertaining to the analysis of the SBLOCA event to support plant operation at a core power level of 3,029.06 MWt (includes measurement uncertainty), a peak LHGR of 13.0 kW/ft, a radial peaking factor (F_r) of 1.81 (1.65 plus uncertainty), and up to 20 percent SG tube plugging per SG.

The NRC staff's review verified that SBLOCA break spectrum analysis results meet the limiting PCT limits and the total MLO and CWO limits set by 10 CFR 50.46(b)(1) through (b)(3). The NRC staff finds the delayed RCP trip study performed by the licensee to be acceptable as it demonstrates that there is at least 2 minutes for operators to trip all four RCPs after the trip criteria is met. The NRC staff finds the results from analysis of the ruptures in attached piping to be acceptable as they are less limiting than the limiting break spectrum case. **[[**

]] The NRC staff finds that the licensee's analysis showed it will continue to meet GDCs 4, 27, and 35 of Appendix A to 10 CFR 50, the requirements of Appendix K to 10 CFR 50, and the limits set by 10 CFR 50.46(b)(1) through (b)(3).

3.2.2 Large Break Loss-of-Coolant Accident (LBLOCA)

The LBLOCA analysis is described in Framatome report ANP-4056P, "St. Lucie Unit 2 24-Month Cycle Large Break LOCA Analysis," Revision 1, February 2024.

3.2.2.1 LBLOCA Description

During normal plant operation at full power, the LBLOCA is initiated by a postulated rupture of the RCS primary piping. The most limiting break is an instantaneously occurring break in the cold leg piping between the RCP and the reactor vessel. A worst-case single failure is also assumed to occur during the accident. The single failure for this analysis, as defined in the EM, assumes the loss of one emergency diesel generator (EDG), which takes out one train of ECCS pumped injection without the loss of containment spray.

The LBLOCA is described in three phases: the blowdown phase, the refill phase, and the reflood phase. The licensee described these phases in Section 3.2, "Description of LBLOCA Event," in ANP-4056.

3.2.2.2 LBLOCA Methodology

The NRC-approved TR EMF-2103P-A, Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," June 2016, describes the Framatome methodology developed for the realistic evaluation of a LBLOCA for PWRs with recirculation (U-tube) SGs. It covers Westinghouse 3-loop and 4-loop plant designs and Combustion Engineering (CE) plants, all with fuel assembly lengths of 14 feet or less and ECCS injection to the cold legs. Since St. Lucie, Unit No. 2, is a CE designed PWR with recirculation SGs, fuel length less than 14 feet, and ECCS injection into the cold legs, this methodology is applicable to St. Lucie, Unit No. 2, for the LBLOCA analysis. The EM in TR EMF-2103P-A for the LBLOCA response of the RCS, secondary system, and the fuel rod used in the analysis is based on the use of the following two computer codes:

- GALILEO for computation of the initial fuel stored energy, fission gas release, and the transient fuel-cladding gap conductance.
- S-RELAP5 code for the thermal-hydraulic system calculations, which includes ICECON for containment response.

The use of S-RELAP5 and the COPERNIC FPC is specified for RLBLOCA analysis as described in TR EMF-2103P-A. However, in TR ANP-10349P, Framatome supplemented the approved EMs and implemented the GALILEO FPC in S-RELAP5. In the SE to TR ANP-10349P, the NRC staff concluded that the GALILEO FPC is an acceptable supplement for COPERNIC for RLBLOCA evaluation models. The NRC staff therefore finds the GALILEO and S-RELAP5 codes appropriate for use with the applied methods.

3.2.2.3 LBLOCA Analysis

The licensee's LBLOCA analysis is based on a statistical realistic LOCA EM in accordance with the methodology in TR EMF-2103P-A instead of conservative EMs specified by 10 CFR 50, Appendix K. For performing the statistical analysis, the licensee created **[[**

]] The licensee sampled each key input parameter over a range established through code uncertainty assessment or expected operating limits provided either by TSs or plant data. The licensee considered the key LOCA parameters listed in Table A-6 of TR EMF-2103P-A, and the uncertainty range associated with each of these parameters given in Table A-7 of TR EMF-2103P-A. The analysis uses the fuel swelling, rupture, and relocation (FSRR) model to determine if cladding rupture occurs and evaluate the consequences of FSRR on the transient response.

Table 4-1 of ANP-4056 shows the plant parameters and ranges used for the analysis. The analysis assumes full-power operation at a core power level of 3,029.06 MWt (including measurement uncertainty), an LHGR of 13.0 kW/ft, an F_R of 1.81 (including measurement uncertainty), and up to 20 percent SG tube plugging per SG. This analysis also addresses typical operational ranges or TS limits for items such as pressurizer pressure and level, accumulator pressure, temperature, and level, loop flow, and containment pressure and temperature.

During the review, NRC staff noticed that the sampled range for the LHGR (as shown in Figure 4-1 of ANP-4056) exceeded the peak value of 13.0 kW/ft (as specified in Table 4-1 of ANP-4056). This was discussed during the regulatory audit and Framatome staff noted that **[[**

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As the sampled region extends into the conservative region, NRC staff finds this acceptable.

3.2.2.4 LBLOCA Results

Table 4-4 of ANP-4056 provides the St. Lucie, Unit No. 2, results of the licensee's analysis for compliance with 10 CFR 50.46(b)(1), (b)(2), and (b)(3). Table 3.2-2 below (extracted from ANP-4056) shows the upper tolerance limit (UTL) for 95/95 simultaneous coverage/confidence results for PCT, MLO, and CWO for **[[ ]]** cases.

Table 3.2.2 – LBLOCA Upper Tolerance Limit for 95/95 Simultaneous Coverage/Confidence Results

Criteria	**St. Lucie Unit 2**		**Acceptance Criteria**
	[[ ]]	**[[ ]]**	
PCT (°F)	1,587	**[[ ]]**	≤ 2,200
MLO (%)	2.30	**[[ ]]**	≤ 17
CWO (%)	0.012	**[[ ]]**	≤ 1

The results in Table 3.2.2 above shows the limiting **[[ ]]** results for 95/95 simultaneous coverage/confidence meet the 10 CFR 50.46(b) criteria with a PCT of 1,587°F, MLO of 2.30 percent, and a total CWO of 0.012 percent. The PCT of 1,587°F occurred in a fresh UO_2 rod w/8 percent gadolinia. Therefore, the NRC staff finds that the results of the licensee's LBLOCA analysis demonstrate that the ECCS is adequate to support the 10 CFR 50.46(b)(1), (b)(2), and (b)(3) acceptance criteria.

3.2.2.5 Compliance with NRC Staff Imposed Limitations and Conditions on LBLOCA Methodologies

For the application of the EMF-2103P-A methodology, there are 11 limitations and conditions listed in Section 4.0 of the NRC staff's SE for TR EMF-2103P-A. The licensee's compliance statements for these are provided in Section 3.7 of ANP-4075.

Table 3.2.3 – Limitations and Conditions for EMF-2103P-A, Revision 3

Limitation		Licensee Response
1	This EM was specifically reviewed in accordance with statements in EMF-2103, Revision 3. The NRC staff determined that the EM is acceptable for determining whether plant-specific results comply with the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3). AREVA did not request, and the NRC staff did not consider, whether this EM would be considered applicable if used to determine whether the requirements of 10 CFR 50.46(b)(4), regarding coolable geometry, or (b)(5), regarding long-term core cooling, are satisfied. Thus, this approval does not apply to the use of SRELAP5-based methods of evaluating the effects of grid deformation due to seismic or LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.	The analysis applies only to the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3).
2	EMF-2103, Revision 3, approval is limited to application for 3-loop and 4-loop Westinghouse designed nuclear steam supply systems (NSSSs), and to Combustion Engineering-designed NSSSs with cold leg ECCS injection, only. The NRC staff did not consider model applicability to other NSSS designs in its review.	St. Lucie, Unit No. 2, is a Combustion Engineering-designed NSSS with cold leg ECCS injection.
3	The EM is approved based on models that are specific to AREVA proprietary M5® fuel cladding. The application of the model to other cladding types has not been reviewed.	The analysis supports operation with $M5_{Framatome}$ cladding.

Limitation		Licensee Response
4	Plant-specific applications will generally be considered acceptable if they follow the modeling guidelines contained in Appendix A to EMF-2103, Revision 3. Plant-specific licensing actions referencing EMF-2103, Revision 3, analyses should include a statement summarizing the extent to which the guidelines were followed, and justification for any departures. <u>Additional Discussion</u> Should NRC staff review determine that absolute adherence to the modeling guidelines is inappropriate for a specific plant, additional information may be requested using the RAI process. For example, if a specific plant shows heightened PCT sensitivity to containment parameters, the NRC staff may request additional information seeking justification for the application of the containment modeling guidelines to that particular plant.	The modeling guidelines contained in Appendix A of EMF-2103(P)(A), Revision 3 were followed completely for the analysis described in this report [ANP-4056].
5	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from data that extend to currently licensed fuel burnup limits (i.e., rod average burnup of **[[ ]]**. Thus, the approval of this method is limited to fuel burnup below this value. Extension beyond rod average burnup of **[[ ]]** would require a revision or supplement to EMF-2103, Revision 3, or plant-specific justification.	The analysis burnups applied in this analysis do not exceed the rod average burnup of **[[ ]]**
6	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from currently available data. Should new data become available to suggest that fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data. Such a request would be tendered by a letter from the NRC to AREVA identifying the newly available data and requesting an update to the model, or an assessment to demonstrate that such an update is not needed.	The analysis uses the approved EMF-2103(P)(A), Revision 3 relocation packing factor application. **[[ ]]**

Limitation		Licensee Response
7	The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. To account for the use of the C-P [Cathcart-Pawel] correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness. <u>Additional Discussion</u> Should the NRC staff position regarding the application of the 17 percent Baker-Just acceptance criterion to the C-P correlation change, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.	The MLO UTL is less than 13 percent.
8	In conjunction with Limitation 8 [7] above, C-P oxidation results will be considered acceptable, provided plant-specific **[[** **]]**. If second-cycle fuel is identified in a plant-specific analysis, whose **[[** **]]**, the NRC staff reviewing the plant-specific analysis may request technical justification or quantitative assessment, demonstrating that **[[** **]]**. <u>Additional Discussion</u> This limitation ensures that the safety analysis retains sufficient margin to the ECR analytic limit to **[[** **]]**.	**[[** **]]** Therefore, all second cycle fuel rod **[[** **]]**

	Limitation	Licensee Response
9	The response to RAI 13 states that all operating ranges used in a plant-specific analysis are supplied for review by the NRC in a table like Table 8-8 of EMF-2103, Revision 3. In plant-specific reviews, the uncertainty treatment for plant parameters will be considered acceptable if plant parameters are **[[** **]]**, as appropriate. Alternative approaches may be used, provided they are supported with appropriate justification. <u>Additional Discussion</u> This limitation ensures that the safety analysis adequately covers the range of permissible plant operation, as discussed in Section 3.4.2 of this SE. However, this limitation should not be construed to imply that exceeding limiting values by any amount is acceptable; sampling distributions for plant parameters should be realistic and justifiable.	**[[** **]]**
10	**[[** **]]**	**[[** **]]** were not used in this analysis.

	Limitation	Licensee Response
11	Any plant submittal to the NRC using EMF-2103, Revision 3, which is not based on the first statistical calculation intended to be the analysis of record must state that a re-analysis has been performed and must identify the changes-that were made to the evaluation model and/or input in order to obtain the results in the submitted analysis. <u>Additional Discussion</u> Adherence to this process ensures that the fidelity of the chosen tolerance level is preserved in the analysis.	The present analysis is the first statistical application of EMF-2103(P)(A), Revision 3 for this plant.

The NRC staff reviewed the licensee response to all 11 limitation and conditions above in Table 3.2.3 and finds that the licensee has satisfactorily met each limitation and condition for use of TR EMF-2103P-A, Revision 3.

For TR ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods," November 2021, Section 4.0, "Limitations and Conditions," of its SE states the following:

> The demonstrated range of applicability of the methodology, specifically RLBLOCA and SBLOCA (EMF-2103 Rev 3, and EMF-2328 Rev 0 and Supplement 1) and applicable range (not related to the thermo-mechanical method) of applicability of GALILEO topical report (ANP-10323P) shall be implemented in the supplement EM (ANP-10349).

With one exception, the licensee response to the limitations and conditions for the use of GALILEO for LBLOCA are all identical to the responses for SBLOCA as described above in Section 3.2.1.5 of this SE. The one difference is the use of gadolinia fuel in the LBLOCA evaluations. Given that the licensee states that the gadolinia concentration analyzed does not exceed 10 wt percent, the NRC staff finds that the specific limitation related to gadolinia concentration is met. Therefore, based on the review of the licensee responses to the limits of code applicability summarized above, the NRC staff finds all limitations and conditions satisfied for the methodologies used in the LBLOCA evaluations.

3.2.2.6 LBLOCA Conclusion

The NRC staff reviewed the information in the licensee's submittal pertaining to the analysis of the LBLOCA event to support plant operation at a core power level of 3,029.06 MWt (includes measurement uncertainty), a peak LHGR of 13.0 kW/ft, a radial peaking factor (F_r) of 1.81 (1.65 plus uncertainty), and up to 20 percent SG tube plugging per SG.

The NRC staff's review verified that LBLOCA break spectrum analysis results meet the limiting PCT limits and the total MLO and CWO limits set by 10 CFR 50.46(b)(1) through (b)(3). In addition, the NRC staff finds that the licensee's analysis demonstrate it will continue to meet GDCs 4, 27, and 35.

3.3 Non-LOCA transients

The non-LOCA analysis is documented in Framatome report ANP-4105P, “St. Lucie Unit 2 24-Month non-LOCA Summary Report,” Revision 2, September 2024.

3.3.1 Non-LOCA Description

The non-LOCA AOOs and postulated accidents (PAs) are described in Chapter 15 of the UFSAR, with the exception of the loss of main feedwater and feedwater line break, which are described in UFSAR Section 10.4.9A. The initial conditions, treatment of measurement uncertainties, component setpoints, component capacities, RPS setpoints and RPS response times are considered and modeled in accordance with the methodology in ANP-10339P-A, “ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology Topical Report” (Reference11). For the change to 24-month cycles, most parameters remain the same, however, the most negative MTC is changed to -36 pcm/°F from the current -32 pcm/°F.

3.3.2 Non-LOCA Evaluation

The non-LOCA AOOs and PAs are reviewed to determine whether the change to 24-month cycles can or will impact the event and whether the event requires re-analysis, whether the event is covered by the existing Analysis of Record (AOR) or existing disposition, whether the event is bounded by another event, or whether the event is not applicable to St. Lucie. Table 3.3.1 below shows the events from the UFSAR and the disposition of the events with consideration of the transition to 24-month cycles.

Table 3.3.1 – Disposition of Events Summary

UFSAR Section	Event Description	Event Condition[2]	Licensee Disposition
15.1	**INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM**		
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	II	Evaluate DNB, Fuel Centerline Melt (FCM) & Transient Cladding Strain (TCS)
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	II	Evaluate DNB, FCM & TCS
15.1.3	Excessive Increase in Secondary Steam Flow (Excess Load)	II	Evaluate DNB, FCM & TCS
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	II	Bounded by 15.1.5 and 15.1.6

[2] Note that “Event Condition” refers to how these events are broken down and are taken from the UFSAR Section 15.06. From the LAR, an Event Condition II is an AOO, and an Event Condition III/IV is a PA. This is provided as AOOs (Event Condition II) have more restrictive acceptance criteria compared to PAs (Event Condition III/IV).

UFSAR Section	Event Description	Event Condition[2]	Licensee Disposition
15.1.5	Pre-Trip Steam System Piping Failure	III/IV	Evaluate DNB & FCM
15.1.6	Post-Trip Steam System Piping Failure	III/IV	Evaluate DNB & FCM
15.2	**DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM**		
15.2.1	Loss of External Electrical Load	---	Bounded by 15.2.3
15.2.2	Turbine Trip	II	Bounded by 15.2.3
15.2.3	Loss of Condenser Vacuum	II	Bounded by AOR
15.2.4	Inadvertent Closure of Main Steam Isolation Valves (MSIVs) (BWR)	II	Not Applicable to St. Lucie Unit 2
15.2.5	Steam Pressure Regulator Failure	II	Not part of the licensing basis
15.2.6	Loss of Non-Emergency Alternating Current (AC) Power to the Station Auxiliaries	II	Bounded by 15.3.2 for DNBR Bounded by 15.2.3 for short term overpressure Bounded by 10.4.9A for decay heat removal
15.2.7	Loss of Normal Feedwater Flow	II	Bounded by 15.3.2 for DNBR Bounded by the AOR for RCS subcooling and peak pressure Bounded by 10.4.9A for decay heat removal
15.2.8	Feedwater System Pipe Break	IV	Bounded by AOR
15.2.9	Transients Resulting from the Malfunction of One Steam Generator		
	Loss of Load to One SG (Single MSIV Closure)	II	Evaluate DNB, FCM & TCS
	Excess Load to One SG	II	Bounded by Loss of Load to One SG
	Loss of Feedwater to One SG	II	Bounded by Loss of Load to One SG
	Excess Feedwater to One SG	II	Bounded by Loss of Load to One SG
15.3	**DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE**		
15.3.1	Partial Loss of Forced Reactor Coolant Flow	II	Bounded by 15.3.2

UFSAR Section	Event Description	Event Condition[2]	Licensee Disposition
15.3.2	Complete Loss of Forced Reactor Coolant Flow	III	Bounded by AOR
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	IV	Bounded by AOR
15.3.4	Reactor Coolant Pump Shaft Break	IV	Bounded by 15.3.3
15.4	**REACTIVITY AND POWER DISTRIBUTION ANOMALIES**		
15.4.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low-Power Startup Condition	II	Evaluate DNB, FCM & TCS
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power	II	Bounded by AOR
15.4.3	CEA Misoperation (Dropped CEA)	II	Bounded by AOR
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	---	Not Applicable to St. Lucie Unit 2
15.4.5	A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor (BWR) Loop that Results in an Increased Reactor Coolant Flow Rate	---	Not Applicable to St. Lucie Unit 2
15.4.6	Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	II	Mode 1 bounded by AOR Modes 2 to 6 Evaluate time to criticality[3]
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	III	Precluded by plant processes and surveillance requirements.
15.4.8	Spectrum of RCCA Ejection Accidents	IV	SAFDLs evaluation is provided in Reference 5.1. Overpressure bounded by AOR
15.5	**INCREASE IN REACTOR COOLANT INVENTORY**		
15.5.1	Inadvertent Operation of the ECCS During Power Operation	II	Event is not credible (the shutoff head of the injection pumps is much less than the RCS pressure during power operation)
15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	II	Bounded by AOR
15.6	**DECREASE IN REACTOR COOLANT INVENTORY**		

[3] The method to determine time to criticality is described in UFSAR Section 15.4.6.2. The NRC staff reviewed this and determined that the method does not change with the increase to 24-month fuel cycles.

UFSAR Section	Event Description	Event Condition[2]	Licensee Disposition
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	II	Bounded by AOR
15.6.2	Break in Instrument Line or Other Lines from the Reactor Coolant Pressure Boundary that Penetrate the Containment	II	Bounded by AOR
15.6.3	Steam Generator Tube Rupture	IV	Bounded by AOR
15.6.4	Spectrum of BWR Steam System Piping Failures Outside of Containment	---	Not applicable
15.6.5	Loss of Coolant Accidents	III/IV	Addressed in Section 3.2 of this SE.
15.7	**RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT**	III	Outside scope of this section.[4]
15.9	**ANTICIPATED TRANSIENTS WITHOUT SCRAM**	---	Bounded by AOR
15.10	**STATION BLACKOUT**	---	Bounded by AOR
10.4.9A	**LOSS OF MAIN FEEDWATER**	---	Bounded by AOR
10.4.9A	**FEEDWATER LINE BREAK**	---	Bounded by AOR

As noted in Table 3.3.1, several analyses require re-analysis based on the transition to 24-month cycles due to the neutronic and fuel design characteristic changes. The licensee indicated that all evaluations performed as specified in Table 3.3.1 use the ARITA – ARTEMIS/RELAP methodology (Reference 11) with the exception of the UFSAR Chapter 15.1.5 post-scram main steam line break event. This event was analyzed in accordance with the EMF-2310 Methodology (Reference 24) as well as the ARCADIA methodology (Reference 3) and the COBRA-FLX methodology (Reference 8).

The NRC staff has reviewed the disposition of the non-LOCA events from Chapter 15 and Chapter 10.4.9A of the UFSAR as well as reviewed more detailed dispositions during the regulatory audit and agrees with the dispositions as noted in the table above. The staff found that the dispositions above and the more detailed dispositions and analyses reviewed during the regulatory audit were appropriately evaluated and considered the non-LOCA UFSAR events to ensure the DNBR SAFDL, the FCM SAFDL, and the TCS SAFDL are not exceeded as a result of extending the cycle length to 24 months.

[4] As stated by the licensee in the "No Significant Hazards Consideration" Section of the LAR: *The proposed change does not affect the type or quantity of radioactive effluent that may be released off-site or increase individual or cumulative occupational exposures resulting from any accident, and thereby cannot increase the consequences of an accident.*

3.3.3 Compliance with NRC Staff Imposed Limitations and Conditions on Non-LOCA Transient/Accident Methodology

Table 3.3.2 – Limitations and Conditions for ANP-10339P-A, Revision 0, "ARITA-ARTEMIS/RELAP Integrated Transient Analysis Methodology," October 2023"

	Limitation on ANP-10339P-A, Revision 0	**Summary of Licensee Response**
1	For plants with a licensing basis that deviates from the SRP guidance current as of the issuance of this SE, the licensee shall assess the compatibility of the proposed ARITA methodology with unique or legacy aspects of the licensing basis in plant-specific implementation submittals. In cases where the plant-specific licensing basis deviates substantially and non-conservatively from current SRP guidance, the licensee's implementation submittal shall propose modifications, as necessary, to the ARITA methodology or licensing basis to assure adequate conservatism. (Reference to Section 3.1.2 of Reference 19)	The ARITA analysis process includes a review of the existing plant licensing basis against the event descriptions included in the SRP as part of the assessment of applicability of the ARITA method for that event. Examples of results of this review are provided in the demonstration analyses available for audit.
2	Plant-specific implementation submittals for the ARITA methodology **[[** **]]** Alternatively, submittals may **[[** **]]** (Reference to Section 3.1.3 of Reference 11)	**[[** **]]**
3	For each analyzed event, plant-specific implementation submittals for the ARITA methodology must identify: (1) the FOMs [figures of merit] considered, (2) the **[[** **]]** discussed in Framatome's response to RAI 11, and (3) the method of determining **[[** **]]** discussed in Framatome's response to RAI 11. (Sections 3.4.2.1 and 3.4.2.4 of Reference 11)	The FOMs for each event analyzed as part of this 24-month cycle design are provided. Further, **[[** **]]**

	Limitation on ANP-10339P-A, Revision 0	Summary of Licensee Response
4	The ARITA methodology does not modify the scope of the postulated misloading events considered within the plant's licensing basis. In addition, as applicable, licensees must justify credit for any equipment, surveillances, and associated acceptance criteria that are not included in TSs, but which are credited with the detection of misloading errors. (Section 3.4.2.5 of Reference 11)	The ARITA methodology is not being applied to postulated misloading events as part of this submittal.
5	Proposed values for all **[[ ]]** used in analyses performed with the ARITA methodology must be included in each plant-specific implementation submittal, including appropriate justification. The licensee must assure that all **[[ ]]** used with the ARITA methodology are acceptable for each operating cycle. Once the **[[ ]]** in the implementation submittal have been approved by the NRC staff, the licensee shall use those values in its analysis unless (1) the values need to be changed to maintain a conservative analysis or (2) the values are changed to reflect an updated plant design configuration with an appropriately conservative margin. (Sections 3.5.1.1 and 3.5.1.7 of Reference 11)	Licensee-controlled parameters including **[[ ]]** used in analyses will continue to be managed in accordance with existing procedures and 10 CFR 50.59. The **[[ ]]** used in the ARITA analyses are available for audit.
6	In the absence of plant-specific data or other information **[[ ]]** licensees implementing the ARITA methodology shall **[[ ]]** (Section 3.5.1.2, PCS-7d and SEC-11c, and Section 3.5.1.3 of Reference 11)	**[[ ]]**

	Limitation on ANP-10339P-A, Revision 0	Summary of Licensee Response
7	For events that consider overfill as a FOM, **[[ ]]** licensees implementing the ARITA methodology shall either **[[ ]]** (Section 3.5.1.2, PCS-20a of Reference 11)	No events considering pressurizer or steam generator overfill as a FOM are being analyzed with the ARITA methodology as part of this submittal.
8	Absent justification for the insignificance of primary system and secondary system metal heat capacity on applicable FOMs, licensees implementing the ARITA methodology shall **[[ ]]** irrespective of evaluation model variant, where the primary system significantly cools down relative to its initial condition, **[[ ]]** (Section 3.5.1.2, PCS-27b of Reference 11)	**[[ ]]**
9	In the absence of plant-specific data or other information conclusively demonstrating the ability to determine the reactor pressure vessel upper head fluid temperature to within **[[ ]]** by the ARITA methodology, licensees adopting the ARITA methodology shall **[[ ]]** (Section 3.5.1.2, PCS-28 of Reference 11)	**[[ ]]**
10	The parameter SEC-2b, **[[ ]]** (Section 3.5.1.2, SEC-2b of Reference 11)	**[[ ]]**

	Limitation on ANP-10339P-A, Revision 0	Summary of Licensee Response
11	Licensees using the ARITA methodology must either (1) justify that the comparisons in ANP-10339P for the MB-2 MSLB testing or other relevant comparisons of the ARITA methodology against experimental data accurately or conservatively represent their plants with respect to the **[[** **]]** or (2) otherwise demonstrate that the **[[** **]]** is representative or conservative for each event where it is relevant to the prediction of applicable FOMs. (Section 3.5.1.2, SEC-6 of Reference 11)	**[[** **]]**
12	The parameter SEC-21a, **[[** **]]** In the event that this prescribed approach prevents stable code execution, licensees shall describe and justify any alternative approach taken. (Section 3.5.1.2, SEC-21a of Reference 11)	An analysis was performed to establish interfacial shear multipliers for the treatment of parameter SEC-21a, Boiler Region Void Uncertainty, consistent with the requirements of this L&C.
13	Licensees using the ARITA methodology shall provide justification for the applicability of parameter **[[** **]]** (Section 3.5.1.2, SEC-21a of Reference 11)	An analysis has been performed to establish interfacial shear multipliers for an extended range of lower pressure to be used in analysis as needed and consistent with the requirements of L&C 12.

Limitation on ANP-10339P-A, Revision 0		Summary of Licensee Response
14	Licensees using the ARITA methodology must ensure that assumptions regarding operator actions remain conservative with respect to the FOMs calculated by the ARITA methodology. (Section 3.5.1.2, HUM-1a of Reference 11)	Operator actions for all transients using the ARITA method will be accurately or conservatively credited.
15	For any event within the scope of the ARITA methodology that exhibits a prompt critical response: (1) the licensee shall perform any necessary analysis for the event using the coupled evaluation model variant and (2) justification must be provided at the time of application for the **[[ ]]** (Section 3.5.1.2, GCN-2a and GCN-2b of Reference 11)	For any event within the scope of the ARITA methodology that exhibits a prompt critical response: (1) the coupled evaluation model variant will be used and (2) justification will be established for the **[[ ]]**
16	The parameter **[[ ]]** (Section 3.5.1.2, GCN-3a of Reference 11)	The parameter **[[ ]]**
17	Unless otherwise justified, the parameter **[[ ]]** consistent with the AREA methodology presented in ANP-10338P-A. (Section 3.5.1.2, LCN-15a3 of Reference 11)	The parameter **[[ ]]**

	Limitation on ANP-10339P-A, Revision 0	Summary of Licensee Response
18	This SE does not approve the uncertainty equation specified in Section 9.1.3.2 of ANP-10339P for determining the **[[** **]]** licensees may (1) provide adequate justification for the equation Framatome proposed in Section 9.1.3.2 of ANP-10339P, (2) conservatively estimate the magnitude of the **[[** **]]** according to the equation $\frac{\delta F_Z}{\lvert F_Z \rvert} = \sqrt{\left(\frac{\delta F_Q}{F_Q}\right)^2 + \left(\frac{\delta F^N_{\Delta H}}{F^N_{\Delta H}}\right)^2}$ or (3) justify other approaches for determining the **[[** **]]** in their implementation submittals. (Section 3.5.1.2, LCN-15b1 of Reference 11)	Justification is provided that the uncertainty equation specified in Section 9.1.3.2 of ANP-10339P for determining the **[[** **]]** can be used to obtain an estimate of the **[[** **]]** for the present application.
19	This SE does not approve the proposed uncertainty approach for parameter **[[** **]]** licensees may (1) conservatively apply **[[** **]]** to this parameter, or (2) justify alternative approaches (e.g., via citation of additional representative data) for determining **[[** **]]** in their implementation submittals. (Section 3.5.1.2, LCN-15b2 of Reference 11)	The parameter **[[** **]]**
20	**[[** **]]** may not be representative of all contemporary or future fuel designs. **[[** **]]** prior to the use of these fuel designs with the ARITA methodology or revise **[[** **]]** to maintain a conservative analysis in accordance with Limitation and Condition 5 (Section 3.5.1.2, TH-5, and TH-8 of Reference 11)	The current implementation does not include a hydraulically mixed core. If a hydraulically mixed core is introduced in a future reload using the ARITA method, then **[[** **]]** will be performed at that time

	Limitation on ANP-10339P-A, Revision 0	**Summary of Licensee Response**
21	Unless otherwise justified, the **[[** **]]** (Section 3.5.1.2, TH-9 of Reference 11)	The current implementation does not include a hydraulically mixed core. If a hydraulically mixed core is introduced in a future reload using the ARITA method, then the **[[** **]]** at that time
22	For uncertainty parameters **[[** **]]** (Section 3.5.1.2, FRR-12 and Section 3.5.1.4 of Reference 11)	For uncertainty parameters **[[** **]]**

	Limitation on ANP-10339P-A, Revision 0	**Summary of Licensee Response**
23	Unless otherwise justified, PORVs and PSVs will be **[[** **]]** (Section 3.5.1.2, **[[** **]]**	Power Operated Relief Valves (PORVs) and Pressurizer Safety Valves (PSVs) **[[** **]]**
24	Unless otherwise justified, licensees applying the ARITA methodology shall **[[** **]]** in accordance with the treatment prescribed per GCN-7a. (Section 3.4.2.2 of Reference 11)	The single rod withdrawal and misaligned rod events are not part of the licensing basis of St. Lucie, Unit No. 2.
25	In the absence of plant-specific data or other information **[[** **]]** when addressing parameter GCN-7a, licensees shall **[[** **]]** (Section 3.5.1.3 of Reference 11)	**[[** **]]**

	Limitation on ANP-10339P-A, Revision 0	Summary of Licensee Response
26	The **[[** **]]** proposed for the ARITA methodology have been reviewed and approved **[[** **]]** The use of engineering analyses, on a case-by-case basis, **[[** **]]** is not precluded by this limitation and condition. (Section 3.5.1.6 of Reference 11)	The scope of this implementation does not include **[[** **]]**
27	Licensees using the ARITA methodology **[[** **]]** Licensees shall consider **[[** **]]** on a cycle-specific basis per Limitation and Condition 5, above. (Section 4.3 of Reference 11)	St. Lucie, Unit No. 2, will continue to collect measured data per existing plant programs/procedures. Where those measured values are relevant to the uncertainty and bias values used in the ARITA methodology, these will be assessed over time as an integral aspect of the reload engineering evaluation, which includes a review of plant parameters used in safety analysis against values applicable to the upcoming reload. This scope is generally performed on a reload basis and addressed in the reload 50.59 process. Representative documents showing the scope of the reload plant parameter review will be available for audit during the NRC review period.
28	For each analyzed event, the applicability range of the ARITA methodology shall be limited to the range over which its constituent computational codes and models have been assessed and validated to provide acceptable predictions of relevant phenomena and processes. (Section 5.1 of Reference 11)	All events evaluated with the ARITA method will include a review of the range of applicability of constituent codes and models vs. transient conditions. **[[** **]]**

The NRC staff reviewed the licensee response to the 28 limitations and conditions above in Table 3.3.2 and finds that the licensee has satisfactorily met each limitation and condition for use of TR ANP-10339P-A, Revision 0. However, some of the limitations and conditions required confirmation of information or responses to RAIs before the staff could conclude the limitations and conditions were satisfactorily met. Several other limitations and conditions required additional considerations and examination of audit documentation for the staff to conclude the licensee satisfactorily met the limitations and conditions. The discussion of these limitations and conditions and the licensee response is discussed further in the following subsection.

3.3.3.1 Discussion of Compliance with Limitations and Conditions for ANP-10339P-A, Revision 0

Limitation and Condition 2

Limitation and Condition 2 requires plant-specific submittals for the ARITA methodology **[[**

]] Section 2.1.1 of ANP-4105P, submitted as part of the licensee's license amendment request, states **[[**

]]

The NRC SRP 4.2 identifies a TCS damage threshold and a TCS failure threshold. In part, the damage threshold is to ensure that cladding is not so deformed during AOOs that the fuel rod can't be returned to service. The failure threshold is applicable to postulated accidents, wherein TCS is employed as a surrogate figure of merit for the pellet-cladding mechanical interaction (PCMI) failure mechanism. While SRP 4.2 specifies a 1 percent TCS criterion for both thresholds, applicants are free to propose an alternative with appropriate justification. In the present case, the licensee indicates that **[[**

]]

As discussed in NUREG/KM-0019 (which documents the underlying technical basis for RG 1.236) and the NRC staff's safety evaluation for PWROG-21001-P-A, PCMI is cladding brittle fracture due to hydrogen embrittlement under reactivity insertion transient conditions. These events are characterized by short duration high mechanical loading. Rod ejection accidents, with their high enthalpy short-width pulses, are the most limiting reactivity insertion event; the high mechanical loading is a result, in part, of the rapid thermal expansion of the pellet and its subsequent pressing up against the cladding.

Failure threshold curves as a function of peak radial average fuel enthalpy rise versus excess cladding hydrogen content are defined in RG 1.236. These curves place limits on peak radial average fuel enthalpy rise for reactivity initiated accidents. Generally, for excess cladding hydrogen < 100 weight parts per million (wppm), the peak radial average fuel enthalpy rise is limited to 150 calories per gram (cal/g) for both PWR and BWR stress relief annealed and recrystallized annealed cladding.

As mentioned above, the licensee's license amendment request indicates **[[**

]] Several salient

cladding performance metrics, **[[** **]]** were provided in support of this statement. In particular, **[[** **]]** The NRC staff verified these points by auditing documentation demonstrating the performance of **[[** **]]** the NRC staff noted **[[** **]]**

Based on the audited data, **[[** **]]** Therefore, the NRC staff finds there is no need to specify a TCS limit **[[** **]]** for SRP Chapter 15 non-LOCA postulated accidents because **[[** **]]** As a result, the NRC staff finds the licensee meets the intent of Limitation and Condition 2 for the ARITA methodology.

<u>Limitation and Condition 5</u>

Limitation and Condition 5 requires that proposed values for all **[[** **]]** used in analyses performed with the ARITA methodology must be included in each plant-specific implementation submittal, including appropriate justification. The licensee did not include a complete list of **[[** **]]** in the submittal, but the licensee did have documentation readily available for audit that contained values for all **[[** **]]** and the associated justifications. The NRC staff found this to be an acceptable approach because the limitation and condition serves to ensure NRC review of **[[** **]]** is performed and to establish acceptable "baseline" of **[[** **]]** that can be referenced should values be updated in the future based on plant modifications, manifestation of non-conservatisms (e.g., drift in **[[** **]]**) or as a necessity to continue representing plant configuration with appropriate conservatism. To this end, the staff notes that the audited documentation that establishes the acceptable "baseline" of **[[** **]]** is FS1-0067117, Revision 2.0, "St. Lucie 2, 24-Month Project (SLU2) – UFSAR 15 Analytical Input Summary, Generic Input Parameters for ARITA," dated 3/18/2024. Additional documentation associated with this document (e.g., FS1-0053704) was also audited.

Appendix IV of the NRC staff's safety evaluation for ANP-10339P-A identifies a population of uncertainty parameters that **[[** **]]** before they can be used in analyses performed with the ARITA methodology. In most instances, these **[[**

]] In a select few instances, [[ ]]

In auditing the documentation discussed above, NRC staff examined the source and derivation of the proposed values for [[ ]] to ensure they are appropriately determined from representative plant data. There are [[ ]] identified in the NRC staff's safety evaluation for ANP-10339P that require [[ ]] license amendment implementing the ARITA methodology. Of these, the licensee indicated under "Limitation and Condition 27 Method of Adherence" in Section 2.1.1 of ANP-4105P, Revision 2 that the treatment of [[ ]] will utilize an approach discussed in the response to RAI-9 of ANP-10339P-A. This treatment is further discussed in the response to RAI-11 of ANP-10339P-A. Succinctly, this treatment involved the use of [[ ]]

As part of the NRC staff's review of the present application, the treatment of [[ ]] was examined to ensure it appropriately adhered to the approach described in the RAI response, and the staff concluded the approach is appropriately implemented. Because the [[ ]] will be treated in a similar fashion, and because the NRC staff concluded the treatment approach itself, as discussed in the response to RAI-11, is being appropriately applied, the NRC staff finds the treatment for [[ ]] is also acceptable.

Of the remaining [[ ]] license amendment implementing the ARITA methodology, [[ ]] The NRC staff examined the treatment of [[ ]] and concluded that the treatment is acceptable because either [[ ]] is utilized in the treatment. Based on this and the discussions above, the NRC staff finds the licensee meets the intent of Limitation and Condition 5 for the ARITA methodology.

<u>Limitation and Condition 6</u>

Limitation and Condition 6 requires licensees implementing the ARITA methodology to [[ ]] The licensee indicated in Section 2.1.1 of ANP-4105P, Revision 2 that [[ ]] However, the NRC staff found that this response does not explicitly demonstrate adherence to the limitation and condition when analyses are performed with the ARITA methodology. Therefore, the NRC staff requested further clarification via RAI.

The licensee's response indicated that [[

]] Because the response discusses the approach that will be taken and that the more conservative results will be used, the NRC staff finds the licensee meets the intent of Limitation and Condition 6 for the ARITA methodology.

Limitation and Condition 8

Limitation and Condition 8 requires licensees implementing the ARITA methodology **[[**

]] where the primary system significantly cools down relative to its initial condition, irrespective of the evaluation model variant. The licensee identified in Section 2.1.1 of ANP-4105P, Revision 2 the treatment for **[[** **]]** but not all ARITA evaluation model variants were discussed. Therefore, the NRC staff requested further clarification via RAI regarding treatment for the evaluation model variants omitted from the discussion. The licensee's response indicated the **[[** **]]** The response further clarified that, **[[**

]] Because the response clearly discusses applicability of the omitted evaluation model variants and the associated treatment when using them, and the treatment is consistent with the staff's safety evaluation for ANP-10339P-A, the NRC staff finds the licensee meets the intent of Limitation and Condition 8 for the ARITA methodology.

Limitation and Condition 9

Limitation and Condition 9 requires licensees implementing the ARITA methodology **[[**

]] The licensee indicated in Section 2.1.1 of ANP-4105P, Revision 2, that **[[**

]] Additionally, in an audit discussion with the licensee on February 27, 2025, the licensee indicated that **[[**

]] However, the NRC staff found that these responses do not explicitly demonstrate adherence to the limitation and condition when analyses are performed with the ARITA methodology. Therefore, the NRC staff requested further clarification via RAI.

The licensee's response indicated that, **[[**

]]
Because the response discusses the approach that will be taken and the approach is consistent with Framatome's NRC-approved methodologies, the NRC staff finds the licensee meets the intent of Limitation and Condition 9 for the ARITA methodology.

Limitation and Condition 11

Limitation and Condition 11 requires licensees adopting the ARITA methodology either 1) justify the comparisons in ANP-10339P for the Model Boiler-2 (MB-2) MSLB testing or other relevant comparisons of the ARITA methodology against experimental data accurately or conservatively represent their plants with respect to **[[ ]],** or 2) otherwise demonstrate the **[[ ]]** is representative or conservative for each event where it is relevant to the prediction of applicable FOMs. In Section 2.1.1 of ANP-4105P, Revision 2, Framatome discusses **[[ ]]** This same comparison was provided in the ARITA topical report, ANP-10339P-A.

While the results provided in ANP-4105P and ANP-10339P-A demonstrate **[[ ]]** which is conservative for an MSLB event, the NRC staff has concerns with regard to the scaling applicability of the test facility to full-size **[[ ]]** and the assurance that **[[ ]]** will be conservatively predicted for future analyses for all relevant reactor designs. In particular, the scaling analysis report for the test facility indicates there are differences in the design of **[[ ]]** Therefore, the NRC staff requested further justification via RAI regarding the scaling applicability of the **[[ ]]** for the MSLB case.

Framatome's response indicated the **[[**

]]

Framatome's response speaks to each of the underlying concerns associated with scaling applicability as identified in the RAI **[[** **]]** indicating the testing results of the **[[** **]]** for the 100 percent MSLB event. Therefore, the NRC staff finds the licensee meets the intent of Limitation and Condition 11 for the ARITA methodology.

<u>Limitation and Condition 12</u>

Limitation and Condition 12 requires parameter SEC-21a **[[** **]]** The licensee indicated in Section 2.1.1 of ANP-4105P, Revision 2 that an analysis was performed to establish interfacial shear multipliers for the treatment of parameter SEC-21a consistent with the requirements of the limitation and condition and this analysis is available for audit. In audit meetings conducted from February 24, 2025, through March 21, 2025, NRC staff audited the applicable analysis and found the **[[** **]]** 10339P-A. Therefore, the NRC staff finds the licensee meets the intent of Limitation and Condition 12 for the ARITA methodology.

<u>Limitation and Condition 13</u>

Limitation and Condition 13 requires licensees implementing the ARITA methodology provide justification for the applicability of parameter **[[** **]]** The licensee indicated in Section 2.1.1 of ANP-4105P, Revision 2 that an analysis was performed to establish interfacial shear multipliers for an extended range of lower pressures to be used in analyses as needed, and the analysis is available for audit. In audit meetings conducted from February 24, 2025, through March 21, 2025, Framatome indicated that **[[** **]]** To further assess the adequacy of this approach, the NRC staff requested additional information via RAI.

The licensee's RAI response indicated **[[** **]]** The NRC staff noted that the **[[** **]]** Because of this, and because the justification is consistent with the approach presented and discussed for

RAI-43 of ANP-10339P-A, the NRC staff finds the licensee meets the intent of Limitation and Condition 13 for the ARITA methodology.

<u>Limitation and Condition 15</u>

Limitation and Condition 15 requires licensees adopting the ARITA methodology to perform any necessary analysis for events exhibiting a prompt critical response using the coupled evaluation model variant and provide justification for **[[** **]]** The licensee indicated in Section 2.1.1 of ANP-4105P, Revision 2, that for any event that exhibits a prompt critical response, the coupled evaluation model variant will be used and justification provided for **[[** **]]** However, the NRC staff found that this response does not explicitly demonstrate adherence to the limitation and condition when analyses are performed with the ARITA methodology. Therefore, the NRC staff requested further clarification via RAI.

The licensee's RAI response indicated **[[** **]]** In audit meetings conducted from February 24, 2025, through March 21, 2025, NRC staff audited these analyses and verified the results, noting **[[** **]]** Because justification was provided for **[[** **]]** and because the coupled evaluation model variant will be used, the NRC staff finds the licensee meets the intent of Limitation and Condition 15 for the ARITA methodology.

<u>Limitation and Condition 18</u>

Limitation and Condition 18 requires licensees adopting the ARITA methodology conservatively estimate the magnitude of **[[** **]]** according to the supplied equation or justify other approaches, which may include additional justification for the equation proposed in Section 9.1.3.2 of ANP-10339P-A (the ARITA formulation). In Section 2.2.1 of ANP-4105P, Revision 2, Framatome provided additional justification for the ARITA formulation. The justification provided focuses on demonstrating 1) the **[[** **]]** when determined using the ARITA formulation and the formulation in Limitation and Condition 18, provided **[[** **]]** and 2) the ARITA formulation **[[** **]]** can be used to obtain a conservative estimate of **[[** **]]**

Table 2-2 of ANP-4105P, Revision 1, provides a comparison of the results from the different formulations and includes **[[** **]]** as determined based on available measurement data. The results show the ARITA formulation and the Limitation and Condition 18 formulation **[[**

]] Additionally, the table shows the ARITA formulation conservatively estimates **[[** **]]**

The NRC staff notes that the use of measured and predicted peaking factor data is instrumental to this approach. The measured and predicted peaking factor data used comes from topical report ANP-10297, Revision 0, Supplement 1P-A, Revision 1, which documents the NRC-approved ARCADIA core design and analysis methodology. The measured peaking factor data in this topical report **[[** **]]** The use of measured peaking factor data suggests the **[[** **]]** and thus appropriate for use as a common reference for comparison to the different formulations.

In audit meetings conducted from February 18, 2025, through March 21, 2025, NRC staff audited available analysis documents to verify the **[[** **]]** The NRC staff noted **[[** **]]** Based on this verification, the NRC staff finds there is reasonable assurance the **[[** **]]** is representative of **[[** **]]** and that, with respect to this, the ARITA formulation results in a conservative estimate **[[** **]]** for the present application. Therefore, the NRC staff finds the licensee meets the intent of Limitation and Condition 18 for the ARITA methodology.

<ins>Limitation and Condition 19</ins>

Limitation and Condition 19 requires licensees adopting the ARITA methodology conservatively apply **[[**

]] The NRC staff finds this is reasonable; if **[[** **]]** then it follows F_Z **[[** **]]**

With regard to **[[** **]]** Framatome indicated in an audit meeting on February 27, 2025, that, **[[** **]]** Because this involves a separate calculation, Framatome made the calculation available for audit. NRC staff audited the calculation and found it to be consistent with the approach discussed in ANP-10338P-A. Based on the justification provided and the discussion above, the NRC staff finds the licensee meets the intent of Limitation and Condition 19 for the ARITA methodology.

3.3.4 Methodology Departures

Section 2.1.2 of ANP-4105P, Revision 2, identifies 6 departures from the NRC-approved ARITA methodology for the St. Lucie Unit 2 application (the NRC-approved ARITA methodology is documented in ANP-10339P-A and the associated NRC safety evaluation). The NRC staff's review of each of these departures is discussed below.

Different ***[[*** ***]]***

In Section 2.1.2 of ANP-4105P, Revision 2, Framatome identified **[[** **]]** These include **[[** **]]**

In the case of **[[** **]]** and the NRC staff notes this is more conservative: **[[** **]]** This will exacerbate the event response. The NRC staff therefore finds the modification acceptable.

Regarding **[[** **]]** The NRC staff's safety evaluation for ANP-10339P-A, Revision 0, identifies that **[[** **]]** is conservatively captured when **[[** **]]** For the St. Lucie, Unit No. 2, application, **[[** **]]** which adds additional conservatism. Therefore, the NRC staff finds the modification acceptable.

For **[[** **]]** The NRC staff finds

the modification acceptable because **[[** **]]** and, per ANP-10339P-A, Revision 0, **[[** **]]**

Sampling Range for ***[[*** ***]]***

Framatome indicates in Section 2.1.2 of ANP-4105P, Revision 2, that the sampling range **[[** **]]** The NRC staff finds this acceptable because the use of **[[** **]]**

Additional Constraint on ***[[*** ***]]*** *and Additional* ***[[*** ***]]***

Section 2.1.2 of ANP-4105P, Revision 2, as supplemented by L-2025-108 Enclosure 1, Appendix A, indicates an additional constraint **[[** **]]** An additional **[[** **]]**

With regard to the additional constraint on **[[** **]]** Because this ensures that TS limits are adhered to while remaining consistent with the approach in ANP-10339P-A **[[** **]]**, the NRC staff finds this departure from the ARITA methodology acceptable.

With regard to the additional **[[** **]]** Because this will result in the application of either the current approved **[[** **]]** or a more conservative one while further ensuring TS limits are adhered to, the NRC staff finds this methodology departure acceptable.

Adjustment to The Sampling Range for ***[[*** ***]]***

Section 2.1.2 of ANP-4105P, Revision 2, indicates the sampling range for **[[**

]]
The NRC staff finds this methodology departure acceptable because **[[**
]] is generally conservative for analyses and because **[[**

]]

*Inclusion of **[[*** *]]*

Section 2.1.2 of ANP-4105P, Revision 2, indicates **[[**

]] The NRC staff finds
this methodology departure acceptable because **[[**
]] is just as conservative or more conservative than
what is prescribed in the approved ARITA methodology.

*Adjusting **[[*** *]]*

The S-RELAP5 plant input model in the ARITA methodology includes **[[**

]] In the NRC-approved
input model, these **[[** **]]** However, Section 2.1.2 of
ANP-4105P, Revision 2, identifies that, **[[**

]] To address this, **[[**

]] The NRC staff notes discussion of these results were included
in the response to RAI-27 of ANP-10339P-A. The NRC staff also audited associated analyses to
verify the veracity **[[** **]]** and conclude they are
reasonable. Based on this, the NRC staff finds the methodology departure acceptable.

3.3.5 Non-LOCA Transients/Accidents Conclusions

The NRC staff reviewed the information in the licensee's submittal pertaining to the analysis of non-LOCA transients and accidents events to support plant operation for St. Lucie, Unit No. 2. The NRC staff's review verified that the non-LOCA transients and accidents analyses appropriately applied the approved methodology in TR ANP-10339P-A, Revision 0, and the licensee has met all the applicable limitations and conditions. The NRC staff also reviewed the departures from the methodology approved in ANP-10339P-A, Revision 0, and concluded the departures are acceptably justified.

The L&Cs associated with ANP-10297P-A, Revision 0 and ANP-10297, Revision 0, Supplement 1P-A were addressed in Section 3.1 of this SE above.

Table 3.3.3 – Limitations and Conditions for ANP-10323P-A, Revision 1, “GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors,” November 2020

Limitation on ANP-10323P-A, Revision 1		Licensee Response
1	The application of GALILEO should assume fuel failure when the predicted fuel temperatures exceed the fuel melting temperature as calculated by GALILEO due to the lack of properties for molten fuel in GALILEO and other properties such as thermal conductivity and fission gas release.	Fuel failure is assumed when fuel temperatures exceed fuel melting temperatures calculated by GALILEO. Therefore, this L&C is met.
2	The ability to make changes to both the mean and standard deviation of model parameter uncertainty values without NRC review and approval is not approved. Because of the complex interaction between parameters in fuel performance codes, the NRC staff does not approve the ability to make changes to the model parameters without NRC approval.	No changes to the mean and standard deviation of model parameter uncertainty values used in GALILEO are made in ARITA analyses. Therefore, this L&C is met.
3	The peak axial node bum up for the fuel rod is limited to **[[ ]]** GWd/MTU.	In designs where the rod burnup is limited to 62 GWd/MTU, the peak nodal burnup is not expected to exceed **[[ ]]** GWd/MTU. Therefore, the **[[ ]]** GWd/MTU limitation will not be challenged. Therefore, this L&C is met.
4	No methodology has been approved for providing initial data or conditions for ECCS analysis.	This document does not include an ECCS analysis. Therefore, this L&C is not applicable.

The NRC staff reviewed the licensee response to the four limitations and conditions above in Table 3.3.3 and finds that the licensee has satisfactorily met each limitation and condition for use of TR ANP-10323P-A, Revision 1.

Table 3.3.4 – Limitations and Conditions for ANP-10311-P-A, Revision 1, "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code," October 2017

Limitation on ANP-10311-P-A, Revision 1		Licensee Response
1	The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer will not be used for safety-related analysis, and are specifically excluded from this review. Additional limitations are specified for the empirical correlations that will be used in licensing calculations. These empirical correlations are listed in Table 1-2 and Appendix A of the TR (Reference 11) and are summarized as the following: a) water properties (IAPWS-IF97) b) friction factor correlation constants i. Lehman friction factor (with or without Szablewski correction) ii. wall viscosity correction option c) two-phase friction multiplier - homogeneous model only d) bulk void correlation - Chexal-Lellouche (using the full curve fit routine or tables with interpolation) e) subcooled void correlation - Saha-Zuber f) subcooled boiling profile fit correlation - Zuber-Staub g) nucleate boiling forced convection heat transfer correlation - Chen h) post-departure from nucleate boiling (DNB) forced convection heat transfer correlation - Groeneveld 5. 7 i) single-phase convection heat transfer correlations i. Sieder-Tate for normal flow conditions ii. McAdams natural convection correlation for very low flow conditions	The COBRA-FLX models used to support the St. Lucie, Unit No. 2, ARITA analyses are consistent with models and correlations that are specified in this L&C. The fuel rod model and rewetting model for post-critical heat flux (CHF) heat transfer are not used in this analysis. Therefore, this L&C is met.
2	This review examined only the specific models and correlations requested by the applicant, as summarized in Section 2.0 of [the SE approving: ANP-10311 P-A, Revision 1, "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code."] These are the only models and correlations that may be used in licensing calculations with the COBRA-FLX subchannel code. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer shall not be used for safety related analysis and are specifically excluded from this review.	The COBRA-FLX models used to support the St. Lucie, Unit No. 2, ARITA analyses are consistent with models and correlations that were approved in Reference 7. The fuel rod model and rewetting model for post-CHF heat transfer are not used in this analysis. Therefore, this L&C is met.

The NRC staff reviewed the licensee response to the two limitations and conditions above in Table 3.3.4 and finds that the licensee has satisfactorily met each limitation and condition for use of TR ANP-10311-P-A, Revision 1.

Table 3.3.5 – Limitations and Conditions for ANP-10311, Revision 1, Supplement 1P-A, Revision 0, “COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation,” March 2023

Limitation on ANP-10311, Revision 1, Supplement 1P-A, Revision 0		Licensee Response
1	The ORFEO-HMP correlation is limited to the application domain as defined in Tables 2-2, 2-3 and 2-4 of Reference 8. Use of the correlation outside of this domain is not considered in this SE.	The ARITA MDNBR analyses use COBRA-FLX over the application domain defined in Reference 8. Therefore, this L&C is met.
2	The ORFEO-HMP correlation is approved for use in the COBRA-FLX computer code with a design limit of **[[ ]]** Use of the correlation in another computer code or with a lower design limit is not considered in this SE.	The ARITA MDNBR analyses use COBRA-FLX with the modeling options consistent with Reference 8 over the application domain. **[[ ]]** Therefore, this L&C is met.
3	The ORFEO-HMP correlation is approved for use consistent with the method of its use in performing the validation analysis domain.	The ARITA MDNBR analyses use COBRA-FLX with the modeling options consistent with Reference 9 over the application domain. Therefore, this L&C is met.

The NRC staff reviewed the licensee response to the three limitations and conditions above in Table 3.3.5 and finds that the licensee has satisfactorily met each limitation and condition for use of TR ANP-10311, Revision 1, Supplement 1P-A, Revision 0.

There is a single limitation and condition on BAW-10227P-A, Revision 2, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” January 2023, related to using the methodology **[[ ]]** when evaluating fatigue. However, this methodology is used for computing fuel rod bowing where the methodology uses a **[[ ]]** Therefore, the NRC staff finds the limitation and condition on BAW-10227P-A, Revision 2 is not applicable to computation of fuel row bow.

In Framatome report ANP-4105, it states that TR EMF-2310, Revision 1, Supplement 2P-A, Revision 0, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” March 2023, is used for the Chapter 15.1.5 post scram main steam line break event. In its May 29, 2025, supplement, the licensee stated that after additional consideration, FPL determined that EMF-2310, Revision 1, Supplement 2P-A, is no longer necessary and appropriate to remain in the list of analytical methods referenced in the proposed changes to TS 5.6.3. During the regulatory audit, the staff received further clarification that there was a typographical error for a reference number in Framatome report ANP-4105 and that EMF-2310, Revision 1, Supplement 2P-A is not used for the post-scram main steam line break event or any other event that was performed in support of the subject LAR. The correct reference should be to ANP-10311P-A, Revision 1, Supplement 1P-A, Revision 0, "COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation," March 2023, where the limitations and conditions for use are described above.

3.4 Control element assembly (CEA) ejection accident

The NRC staff reviewed the CEA analysis provided in report ANP-4091P, Revision 1 (Reference1), which was provided with the LAR (Enclosure 6 for the proprietary version and Enclosure 12 for the non-proprietary version). For the purposes of this analysis, a rod ejection accident (REA) is comparable to a CEA ejection accident, and the only difference is the plant nomenclature for the reactivity control elements (Framatome and RG 1.236 use "Control Rod" while St. Lucie, Unit No. 2, uses "Control Element Assembly").

3.4.1 Accident Description and Analysis Method

The CEA ejection event is initiated by a postulated rupture of a control rod drive mechanism housing. Such a rupture would theoretically allow the full system pressure to act on the drive shaft, which would eject its control rod from the core. The consequences of the postulated failure would be a rapid positive reactivity insertion, a core power excursion, and an increase in radial power peaking, which potentially leads to localized fuel rod damage. The power excursion would be mitigated by the fuel temperature (Doppler) feedback, and, in some cases terminated by the RPS with a reactor trip in response to changes in neutron flux or system pressure. Although the initial increase in power would occur too rapidly for control rod scram to affect the power increase, the negative reactivity inserted during scram would affect the fuel temperature and fuel rod cladding surface heat flux.

The analysis was performed based on the criteria defined in Regulatory Guide 1.236 and the AREA methodology (Reference 6). From the AREA SE Section 2.0:

> The AREA methodology basically consists of coupling the NRC approved reactor analysis system code ARTEMIS to the NRC approved code S-RELAP5 to account for the RCS response in the rod ejection transient. ARCADIA is a code package that provides a converged code system for neutronic and thermal-hydraulic core design and safety evaluation. The main components of the ARCADIA system are the spectral/lattice code APOLLO2-A and the core simulator ARTEMIS. The core simulator ARTEMIS is a 3-D nodal multigroup reactor burnup code with pin power reconstruction for PWRs. In addition, a thermal-hydraulic program COBRA-FLX that is capable of performing 3-D steady state, transient full-core, and subchannel analyses has been integrated into ARTEMIS to have the capability to solve complex two-phase flow problems. The fuel pin temperature for use with COBRA-FLX are calculated by the fuel rod model (FRM). This model, within the ARTEMIS code system, solves both the static and time-dependent, one-dimensional thermal equations for the fuel rods to compute the fuel temperature distribution and heat flux to the coolant.

In addition, the AREA methodology introduces elements of the current under review fuel performance code GALILEO[5], used to define fuel and clad thermal properties for the FRM in both the neutronics solution and the thermal-hydraulic solution in ARTEMIS.

The criteria within the AREA methodology consists of the following:

- **[[**

  ]]
- The enthalpy rise limit is based on excess hydrogen as defined in RG 1.236. The enthalpy limit used for high temperature cladding failure threshold in RG 1.236 is a function of internal pin pressure with a maximum of 170 calories per gram (cal/g) for internal pressures less than system pressure and a minimum limit of 100 cal/g for internal pressures higher than system pressure.

RG 1.236 has the following restrictions for coolability:

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions.

Methodology departures:

- GALILEO (Reference 5) is used as the fuel performance code in this analysis**. [[**

  ]]
- **[[**

  ]].

The update of the fuel temperature uncertainty is representative of the new revision of GALILEO used and represents a larger uncertainty to bound the fewer data points. The use of RCS depressurization in MNDBR analyses is more conservative than the methodology as described in RG 1.236 and is acceptable for use in this analysis. The NRC staff has reviewed the methodology departures and finds they are acceptable for use at St. Lucie, Unit No. 2, analyses.

---

[5] Note that since the issuance of this SE, GALILEO has been approved by the NRC for use (see Section 3.4.5 for discussion of the Limitations and Conditions for use of GALILEO).

Methodology clarifications:

- The version of GALILEO used in this analysis is provided in Reference 5. The version of GALILEO used in the approved AREA methodology (Reference 6) is a prior revision of GALILEO provided in Reference 4. The newer revision of GALILEO was benchmarked to the prior revision and showed comparable results, with some improvements in the results. These improvements were not factored into this analysis.

  The NRC staff agrees that not considering improvements as a result of code/methodology improvements is a conservative approach and acceptable for use in the St. Lucie, Unit No. 2, AREA evaluation.

- For non-prompt scenarios with uncharacteristic pulse widths (the empirical database is compromised of narrow pulses), **[[**

  ]].

  NRC staff agrees this approach is conservative and acceptable for use in the St. Lucie, Unit No. 2, AREA evaluation.

- The enthalpy limits from Reference 6 are based on prompt critical testing and guard against the clad overheat and PCMI failure. For non-prompt critical ejected rod events the power deposition occurs over a longer time period and failure mechanisms such as DNBR are used. **[[**

  ]]

  The NRC staff reviewed the above information and finds that the approach taken between prompt critical and non-prompt critical cases is consistent with the guidance in RG 1.236 and therefore, acceptable.

- System pressure calculations were not performed as part of the St. Lucie, Unit No. 2, AREA analysis.

  The NRC staff agrees that not performing system pressure calculations is in accordance with the approved AREA methodology, so this is acceptable.

3.4.2 Cycle Inputs

The St. Lucie, Unit No. 2, REA analysis was performed for a full core of Framatome HTP fuel with M5 cladding, and the impact of transitions cycles from 18-months to 24-months is considered through the development of cycle-to-cycle biases.

The analysis is performed for **[[ ]]** times in life (TIL) and at **[[ ]]** power levels for each TIL. The TILs considered are **[[**
]]. The selected power levels are **[[**

]]. Table 2-1 of Reference 1 shows the CEA Insertion limits with respect to power used for the analysis. Table 2-2 of Reference 1 shows the Core Model Parameter Uncertainties applied in the AREA analysis with the depressurization curve supporting the MDNBR analysis provided in Figure 2-1 of Reference 1. Table 2-3 of Reference 1 shows the core model parameter biases, which extend the REA to cover both transition and future cycles.

The NRC staff has reviewed the cycle inputs provided in the LAR and agrees that the inputs used in the analysis agree with the NRC approved AREA methodology (Reference 6) and are acceptable for the CEA ejection analysis.

3.4.3 CEA Ejection Limits Generated by GALILEO

The pellet clad mechanical interaction (PCMI) limits are specified in Section C.3.2 of RG 1.236. The excess hydrogen is calculated using GALILEO (Reference 5). **[[**

]].

3.4.4 Fuel Integrity Summaries

[[

]] The margins reported are based on the calculated value minus the limit, so that a negative number is favorable. A positive value indicates a violation of the limit. Additional details are provided for the cases with the least margin to the limit for fuel melt, fuel rim melt, MDNBR, enthalpy, and enthalpy rise. Limiting cases for enthalpy, enthalpy rise, and fuel rim temperature are considered for those cases which are **[[**

[[

]].

The results reported in Tables 4-2 through 4-6 of Reference 1 are summarized below in Table 3.4.1, which provides limiting criteria for power level, cycle burnup, **[[**

]]. Because no prompt critical cases were identified, the parameters below that are limiting for prompt critical cases are not reported, specifically the maximum enthalpy, maximum enthalpy rise, and maximum fuel rim temperature.

Table 3.4.1 - Measure of Conservatism for Limiting Results

[[

]]

The NRC staff reviewed the St. Lucie, Unit No. 2, CEA analysis as described in Reference 1. The NRC staff determined that the analysis was performed according to the NRC-approved AREA methodology and is consistent with RG 1.236, with the clarifications and departures as described above. The fuel related acceptance criteria for this event are evaluated to support the transition to 24-month cycles. The AREA methodology implementation sufficiently addresses transition and equilibrium cycles of 24-months with the development of core model parameter biases. The NRC staff determined that the CEA analysis provides adequate margin to limits for fuel temperature, fuel rim temperature, MDNBR, enthalpy, and enthalpy rise which assures that any fuel rod failures are below the limits provided in RG 1.236.

3.4.5 Compliance with NRC Staff Imposed Limitations and Conditions

The AREA-ARCADIA Rod Ejection Accident methodology (Reference 6) is being added to TS 5.6.3. This methodology was approved by the NRC with three Limitation and Conditions, which are addressed below in Table 3.4.2.

Table 3.4.2 - Limitations and Conditions for ANP-10338P-A, Revision 0, "AREA – ARCADIA Rod Ejection Accident," December 2017

Limitation on ANP-10338-P-A, Revision 0		Licensee Response
1	The AREA methodology is limited to the evaluation of a control rod ejection accident in a PWR.	The St. Lucie, Unit No. 2, is a pressurized water reactor. This methodology is used to analyze CEA ejection. Therefore, this L&C is met.
2	The AREA methodology consists of coupled AREVA codes and methods. The application of the AREA methodology is limited to the conditions and limitations on the SEs of the approved codes that the AREA methodology uses in its analysis of a control rod ejection accident.	The L&Cs for each of the topical reports used in the St. Lucie Unit 2 CEA ejection analysis are addressed in this section. Therefore, this L&C is met.
3	The AREA methodology is limited to only the GALILEO derived thermal-mechanical properties of fuel pins. The use of another NRC approved code for the thermal-mechanical properties must be noted and the computed differences quantified and justified.	Only GALILEO derived thermal-mechanical properties are used for the CEA ejection analysis for St. Lucie, Unit No. 2. Therefore, this L&C is met.

The NRC staff has reviewed the three Limitations and Conditions above and the licensee responses and determined that the licensee responses adequately address the Limitations and Conditions for ANP-10338P-A (Reference 6) for use in the CEA analysis.

The Limitations and Conditions related to the ARCADIA methodology (Reference 2 and 3) were addressed in Section 3.1.

The Limitations and Conditions related to the COBRA-FLX methodology (Reference 7) are shown below in Table 3.4.3.

Table 3.4.3 - Limitations and Conditions for ANP-10311P-A, Revision1, “COBRA-FLX: A Core Thermal-Hydraulic Analysis Code”, October 2017

Limitation on ANP-10311-P-A, Revision 1		Licensee Response
1	The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer will not be used for safety-related analysis and are specifically excluded from this review. Additional limitations are specified for the empirical correlations that will be used in licensing calculations. These empirical correlations are listed in Table 1-2 and Appendix A of the TR (Reference 11) and are summarized as the following: a) water properties (IAPWS-IF97) b) friction factor correlation constants i. Lehman friction factor (with or without Szablewski correction) ii. wall viscosity correction option c) two-phase friction multiplier - homogeneous model only d) bulk void correlation - Chexal-Lellouche (using the full curve fit routine or tables with interpolation) e) subcooled void correlation - Saha-Zuber f) subcooled boiling profile fit correlation - Zuber-Staub g) nucleate boiling forced convection heat transfer correlation - Chen h) post-departure from nucleate boiling (DNB) forced convection heat transfer correlation -Groeneveld 5. 7 i) single-phase convection heat transfer correlations i. Sieder-Tate for normal flow conditions ii. McAdams natural convection correlation for very low flow conditions	The COBRA-FLX models used to support the St. Lucie, Unit No. 2, CEA ejection accident analysis are consistent with models and correlations that are specified in this L&C. The fuel rod model and rewetting model for post-CHF heat transfer are not used in this analysis.

	Limitation on ANP-10311-P-A, Revision 1	Licensee Response
2	*This review examined only the specific models and correlations requested by the applicant, as summarized in Section 2.0 of [the SE approving: ANP-10311 P-A, Revision 1, "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code." These are the only models and correlations that may be used in licensing calculations with the COBRA-FLX subchannel code. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer shall not be used for safety related analysis, and are specifically excluded from this review.*	The COBRA-FLX models used to support the St. Lucie, Unit No. 2, CEA ejection analysis are consistent with models and correlations that were approved in Reference 7. The fuel rod model and rewetting model for post-CHF heat transfer are not used in this analysis.

The NRC staff has reviewed the licensee response to the Limitations and Conditions listed above and find that ANP-10311-P-A (Reference 7) is appropriate for use in the CEA analysis and the L&Cs are met.

The Limitations and Conditions related to the ORFEO-HMP methodology (Reference 8) are shown below in Table 3.4.4.

Table 3.4.4 - Limitations and Conditions for ANP-10311, Revision 1, Supplement 1P-A, Revision 0, COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation, March 2023

	Limitation on ANP-10311, Revision 1, Supplement 1P-A, Revision 0	Licensee Response
1	*The ORFEO-HMP correlation is limited to the application domain as defined in Tables 2-2, 2-3 and 2-4 of the TR (*Reference 8*). Use of the correlation outside of this domain is not considered in this SE.*	The COBRA-FLX models used in the St. Lucie, Unit No. 2, CEA ejection analysis were created with design data that fall within the range of applicability of the topical report. Therefore, this L&C is met.
2	*The ORFEO-HMP correlation is approved for use in the COBRA-FLX computer code with a design limit of* ***[[ ]]*** *Use of the correlation in another computer code or with a lower design limit is not considered in this SE.*	The CEA ejection MDNBR analysis uses COBRA-FLX with the modeling options consistent with Reference 7. A design limit of ***[[ ]]*** is used for ORFEO-HMP correlation over the application domain. Therefore, this L&C is met.
3	*The ORFEO-HMP correlation is approved for use consistent with the method of its use in performing the validation analysis.*	COBRA-FLX employs the ORFEO-HMP correlation in the CEA ejection MDNBR analysis. Table 5-1 in Section 5.0 of the Reference 8, identifies important models and code options which were used in performing the validation of the ORFEO-HMP correlation. The ORFEO-HMP correlation is applied consistent with those models and code options in the CEA ejection analysis. Therefore, this L&C is met.

The NRC staff has reviewed the licensee response to the Limitations and Conditions listed above and find that ANP-10311, Revision 1, Supplement 1P-A, Revision 0 (Reference 8) is appropriate for use in the CEA analysis and the L&Cs are met.

The Limitations and Conditions related to the GALILEO methodology (Reference 5) are shown below in Table 3.4.5.

Table 3.4.5 - Limitations and Conditions for ANP-10323P-A, Revision 1, GALILEO Fuel Rod Thermal Mechanical Methodology for Pressurized Water Reactors, November 2020

Limitation on ANP-10323P-A, Revision 1		Licensee Response
1	*The application of GALILEO should assume fuel failure when the predicted fuel temperatures exceed the fuel melting temperatures as calculated by GALILEO due to the lack of properties for molten fuel in GALILEO and other properties such as thermal conductivity and fission gas release.*	Fuel failure is assumed when fuel temperatures exceed fuel melting temperatures calculated by GALILEO. Therefore, this L&C is met.
2	*The ability to make changes to both the mean and standard deviation of model parameter uncertainty values without NRC review and approval is not approved. Because of the complex interaction between parameters in fuel performance codes, the NRC staff does not approve the ability to make changes to the model parameters without NRC approval.*	No changes to the mean and standard deviation of model parameter uncertainty values used in GALILEO are made in AREA analyses. Therefore, this L&C is met.
3	*The peak axial node burnup for the fuel rod is limited to* ***[[ ]]*** *GWd/MTU.*	In designs where the rod burnup is limited to **[[ ]]** GWd/MTU, the peak nodal burnup is not expected to exceed **[[ ]]** GWd/MTU. Therefore, the **[[ ]]** GWd/MTU limitation will not be challenged. Therefore, this L&C is met.
4	*No methodology has been approved for providing initial data or conditions for ECCS analysis.*	This document does not include an ECCS analysis. Therefore, this L&C is not applicable.

The NRC staff has reviewed the licensee response to the Limitations and Conditions listed above and find that ANP-10323P-A, Revision 1 (Reference 5) is appropriate for use in the CEA analysis and the L&Cs are met.

The Limitations and Conditions related to the rod bow methodology (Reference 9 and updated in Reference 10) are shown below in Table 3.4.6 and Table 3.4.7.

Table 3.4.6 - Limitations and Conditions for XN-75-32(P)(A) and Supplements 1, 2, 3, & 4, Computational Procedure for Evaluating Fuel Rod Bowing

Limitation on XN-75-32(P)(A) and Supplements 1, 2, 3, & 4		Licensee Response
1	*The acceptance is not applicable to fuel designs which exhibit a greater propensity for bowing than that given in data from which the models reviewed were developed.*	This L&C refers to the original gap closure correlation from Reference 9. Since this analysis is not using the gap closure correlation from Reference 9 and is instead using the upper design limit gap closure ratio (UDL GCR) from Reference 10 (Equation 11-4), this L&C is not applicable to the St. Lucie, Unit No. 2, AREA analysis. No further action is required; thus, the L&C is met.
2	*If the residual DNBR penalties due to fuel rod bowing are partially or totally offset by using generic or plant-specific DNBR margin, the margin used to offset these penalties must be documented in the bases to the technical specifications and any remnant penalties must be accommodated into the technical specifications.*	Generic and/or plant-specific margins are not used to offset the application of residual Departure from Nucleate Boiling Ratio (DNBR) rod bow penalties in the St. Lucie, Unit No. 2, AREA analysis. Since this method discussed in the condition was not performed, there is no requirement to document this in the TS bases; thus, the L&C is met.
3	*If the inter-assembly gap distance increases by more than 50 mils, the NRC requires a more detailed analysis.*	It was noted in the SER that assembly bow effects were not considered for FQ (linear heat generation rate (LHGR)) or DNBR analyses. The NRC stated that, due to a number of conservatisms, this was acceptable if the 95/95 inter-assembly gap increased by less than 50 mils. For analyses using Reference 9, it is conservatively assumed that the change in inter-assembly gap distance exceeds 50 mils. As such, the impact of assembly bow on LHGR is accounted for by using the same penalties developed for the ARITA topical report (Reference 11, Section 9.1.3.5). This methodology is conservative for the treatment of assembly bow. Therefore, this L&C is met.

Limitation on XN-75-32(P)(A) and Supplements 1, 2, 3, & 4		Licensee Response
4	*The statistical method used by Exxon in determining the DNBR penalty is considered incomplete in that it does not properly account for the bowing of all eight rods surrounding the hot rod in the core.* *While this is a significant deficiency in the Exxon rod bowing DNBR statistical methodology, we agree with Exxon that there are several conservatisms in the treatment that provide sufficient margin to offset the deficiency.***	The conservatisms given in the Reference 9 SER are utilized in the St. Lucie, Unit No. 2, CEA ejection accident analysis. Therefore, Framatome's current rod bow methodology is considered conservative and compliant. The updated gap closure correlation in Reference 10, Section 11.4, does not affect the applicability of the conservatisms provided in the Reference 9 SER. Therefore, this L&C is met.

***In Reference 9, pages 14-15 of the SER, a list of conservatisms used in the methodology is presented by the NRC. The NRC deemed the DNBR statistical methodology as acceptable with the use of those conservatisms.*

Table 3.4.7 - Limitations and Conditions for BAW-10227P-A, Revision 2, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel

Limitation on BAW-10227P-A, Revision 2		Licensee Response
1	*When applying the methodology described in BAW-10227P, Rev. 2,* **[[** **]]** *licensees shall ensure that changes to expected fatigue cycles are appropriately captured in the fatigue evaluation.*	This L&C is not applicable to the calculation and application of a rod bow penalty. No further action is required. Therefore, this L&C is met.

The NRC staff has reviewed the licensee response to the Limitations and Conditions listed above and find that XN-75-32(P)(A) and Supplements 1, 2, 3, & 4 (Reference 9) and BAW-10227P-A, Revision 2 (Reference 10) are appropriate for use in the CEA analysis and the L&Cs are met.

3.5 Technical Conclusion

The NRC staff reviewed the LAR and its supplement to evaluate the acceptability of the proposed changes to TS to update the listing of NRC-approved analytical methods used to determine the core operating limits. Specifically, changes to the fuel thermal-mechanics, core thermal-hydraulics, emergency core cooling, nuclear design, and select design basis event analyses are proposed using NRC-approved advanced codes and methods.

Based on its review, the NRC staff has determined that the licensee provided an adequate technical basis to support the proposed changes. Specifically, the NRC staff has determined that the licensee has demonstrated that (1) Framatome codes and methods are applicable for St. Lucie, Unit No. 2, (2) the licensee complies with the staff limitations and conditions imposed for application of TRs used in the execution of the LAR, (3) the safety analysis results submitted to the NRC staff demonstrate compliance with applicable regulatory requirements, and (4) the proposed TS changes are acceptable. Therefore, the NRC staff finds that the requirements of 10 CFR 50.36(c)(5) will continue to be met because the licensee proposed changes to TS administrative controls will continue to ensure the operation of the facility in a safe manner.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Florida State official was notified of the proposed issuance of the amendment on August 25, 2025. The State official submitted no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR, Part 20. 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 published in the *Federal Register* on October 29, 2024 (89 FR 85994). Accordingly, the 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 the 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

1. ANP-4091P, Revision 1, St. Lucie Unit 2 Control Element Assembly Ejection Accident Analysis Topical Report, August 2024.

2. ANP-10297P-A, Revision 0, The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results, February 2013.

3. ANP-10297P-A, Revision 0, Supplement 1P-A, Revision1, Ther ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results, December 2020.

4. ANP-10323P, Revision 0, Fuel Rod Thermal-Mechanical Methodology for Boiling Water Reactors and Pressurized Water Reactors, July 2013.

5. ANP-10323P-A, Revision 1, GALILEO Fuel Rod Thermal Mechanical methodology for Pressurized Water Reactors, November 2020.

6. ANP-10338P-A, Revision 0, AREA – ARCADIA Rod Ejection Accident, December 2017.

7. ANP-10311P-A, Revision 1, COBRA-FLX: A Core Thermal-Hydraulic Analysis Code, October 2017.

8. ANP-10311, Revision 1, Supplement 1P-A, Revision 0, COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation, March 2023.

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

10. BAW-10227P-A, Revision 2, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, January 2023.

11. ANP-10339P-A, Revision 0, ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology, October 2023.

12. EMF-92-116(P)(A), Revision 0, Supplement 1, Revision 0(P)(A) "Generic Mechanical Design Criteria for PWR Fuel Designs."

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

14. ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018.

15. Section 4.2, "Fuel System Design," Revision 3, March 2007, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

16. BAW-10227P-A, Revision 2, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," January 2023.

17. ANP-10323P-A, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," November 2020.

18. BAW-10084P-A, Revision 3, "Program To Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995.

19. ANP-10339P-A, Revision 0, "ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology," October 2023.

20. ANP-10297P-A, Revision 0, Supplement 1P-A, Revision 1, "The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," December 2020.

21. ANP-10338P-A, Revision 0, "AREA™ - ARCADIA® Rod Ejection Accident," December 2017.

22. ANP-10297P-A, Revision 0, The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results, February 2013.

23. NRC-IC-23-017, Request for U.S. Nuclear Regulatory Commission Confirmation that a Change to the ARCADIA Code System is within the Scope of Approved Methods in Framatome, Inc. Topical Report, ANP-10297, Revision 0, Supplement 1P-A, Revision 1, “The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results” (EPID L-2023-TOP-0019), September 11, 2023.

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

25. ANP-10311P-A, Revision 1, Supplement 1P-A, Revision 0 “COBRA-FLX: ORFEO-HMP Critical Heat Flux Correlation”, March 2023.

26. ANP-4090P Revision 1, “St. Lucie Unit 2 24-Month Fuel Analysis License Amendment Request”, August 2024.

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Date of Issuance: December 12, 2025

SUBJECT: ST. LUCIE PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 213 TO REVISE TECHNICAL SPECIFICATION 5.6.3, "CORE OPERATING LIMITS REPORT," BY UPDATING THE LISTING OF NRC-APPROVED ANALYTICAL METHODS USED TO DETERMINE THE CORE OPERATING LIMITS (EPID L-2024-LLA-0124) DATED DECEMBER 12, 2025

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