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

December 20, 2017

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295 / Bin – 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS ADOPTING ALTERNATIVE SOURCE TERM, TSTF-448, REVISION 3, AND TSTF-312, REVISION 1 (CAC NOS. MF8861, MF8862, MF8916, MF8917, MF8918, AND MF8919; EPID NOS. L-2016-LLA-0017, L-2016-LLA-0018, AND L-2016-LLA-0019)

Dear Mr. Hutto:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 216 to Renewed Facility Operating License No. NPF-2 and Amendment No. 213 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The amendments are in response to your application dated November 22, 2016, as supplemented by letters dated May 23, June 8, September 7, November 21, and December 18, 2017.

Specifically, the amendments revise the licensing basis to support a full-scope implementation of the Alternative Source Term radiological analysis methodology and modifies Technical Specification (TS) 3.7.10, TS 3.9.3, and TS 5.5.18, consistent with Technical Specifications Task Force (TSTF) Travelers TSTF-312, Revision 1, “Administratively Control Containment Penetrations,” and TSTF-448, Revision 3, “Control Room Habitability.” The amendments also include a license condition associated with TSTF-448.

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

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 216 to NPF-2
2. Amendment No. 213 to NPF-8
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 216
Renewed License No. NPF-2

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 1, (the facility), Renewed Facility Operating License No. NPF-2, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated November 22, 2016, as supplemented by letters dated May 23, June 8, September 7, November 21, and December 18, 2017, 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 license 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, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(7) of Renewed Facility Operating License No. NPF-2 is hereby amended to read as follows:

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 216, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

2.C.(7) Upon implementation of Amendment No. 216 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

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

Michael T. Markley, Chief
Plant Licensing Branch II-1
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 20, 2017

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

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 213
Renewed License No. NPF-8

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 2, (the facility), Renewed Facility Operating License No. NPF-8, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated November 22, 2016, as supplemented by letters dated May 23, June 8, September 7, November 21, and December 18, 2017, 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 license 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, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(7) of Renewed Facility Operating License No. NPF-8 is hereby amended to read as follows:

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 213, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

2.C.(7) Upon implementation of Amendment No. 213 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

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

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
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 20, 2017

ATTACHMENT TO LICENSE AMENDMENT NOS. 216 AND 213

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. NPF-2 AND NPF-8

DOCKET NOS. 50-348 AND 50-364

Replace the following pages of the Renewed Facility Operating Licenses and Appendix "A" Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License	License
NPF-2, page 4	NPF-2, page 4
NPF-2, pages 8-12	NPF-2, pages 8-13
NPF-8, page 3	NPF-8, page 3
NPF-8, pages 6-11	NPF-8, pages 6-12
TSs	TSs
1.1-2	1.1-2
3.7.10-1	3.7.10-1
3.7.10-2	3.7.10-2
3.7.10-3	3.7.10-3
3.7.10-4	3.7.10-4
3.9.3-1	3.9.3-1
5.5-15	5.5-15
5.5-16	5.5-16

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 216, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission.

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

   Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

   1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

   2) Identification of the procedures used to quantify parameters that are critical to control points;

   3) Identification of process sampling points;

   4) A procedure for the recording and management of data;

   5) Procedures defining corrective actions for off control point chemistry conditions; and

(5) Updated Final Safety Analysis Report Supplement

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The Southern Nuclear Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. Southern Nuclear shall complete these activities no later than June 25, 2017, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(6) Reactor Vessel Material Surveillance Capsules

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

(7) Upon implementation of Amendment No. 216 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

D. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," and was submitted on May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 186, as supplemented by change approved by License Amendment No. 199.

E. This renewed license is subject to the following additional conditions for the protection of the environment:

(1) Southern Nuclear shall operate the facility within applicable Federal and State air and water quality standards and the Environmental Protection Plan (Appendix B).

(2) Before engaging in an operational activity not evaluated by the Commission, Southern Nuclear will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than evaluated in the Final Environmental Statement, Southern Nuclear shall provide a written evaluation of such activities and obtain prior approval of the Director, Office of Nuclear Reactor Regulation, for the activities.

F. Alabama Power Company shall meet the following antitrust conditions:

(1) Alabama Power Company shall recognize and accord to Alabama Electric Cooperative (AEC) the status of a competing electric utility in central and southern Alabama.

(2) Alabama Power Company shall offer to sell to AEC an undivided ownership interest in Units 1 and 2 of the Farley Nuclear Plant. The percentage of ownership interest to be so offered shall be an amount based on the relative sizes of the respective peak loads of AEC and the Alabama Power Company (excluding from the Alabama Power Company's peak load that amount imposed by members of AEC upon the electric system of Alabama Power Company) occurring in 1976. The price to be paid by AEC for its proportionate share of Units 1 and 2, determined in accordance with the foregoing formula, will be established by the parties through good faith negotiations. The price shall be sufficient to fairly reimburse Alabama Power Company for the proportionate share of its total costs related to the Units 1 and 2 including, but not limited to, all costs of construction, installation, ownership and licensing, as of a date, to be agreed to by the two parties, which fairly accommodates both their respective interests. The offer by Alabama Power Company to sell an undivided ownership interest in Units 1 and 2 may be conditioned, at Alabama Power Company's option, on the agreement by AEC to waive any right of partition of the Farley Plant and to avoid interference in the day-to-day operation of the plant.

(3) Alabama Power Company will provide, under contractual arrangements between Alabama Power Company and AEC, transmission services via its electric system (a) from AEC's electric system to AEC's off-system members; and (b) to AEC's electric system from electric systems other than Alabama Power Company's and from AEC's electric system to electric systems other than Alabama Power Company's. The contractual arrangements covering such transmission services shall embrace rates and charges reflecting conventional accounting and ratemaking concepts followed by the Federal Energy Regulatory Commission (or its successor in function) in testing the reasonableness of rates and charges for transmission services. Such contractual arrangements shall contain provisions protecting Alabama Power Company against economic detriment resulting from transmission line or transmission losses associated therewith.

(4) Alabama Power Company shall furnish such other bulk power supply services as are reasonably available from its system.

(5) Alabama Power Company shall enter into appropriate contractual arrangements amending the 1972 Interconnection Agreement as last amended to provide for a reserve sharing arrangement between Alabama Power Company and AEC under which Alabama Power Company will provide reserve generating capacity in accordance with practices applicable to its responsibility to the operating companies of the Southern Company System. AEC shall maintain a minimum level expressed as a percentage of coincident peak one-hour kilowatt load equal to the percent reserve level similarly expressed for Alabama Power Company as determined by the Southern Company System under its minimum reserve criterion then in effect. Alabama Power Company shall provide to AEC such data as needed from time to time to demonstrate the basis for the need for such minimum reserve level.

(6) Alabama Power Company shall refrain from taking any steps, including but not limited to, the adoption of restrictive provisions in rate filings or negotiated contracts for the sale of wholesale power, that serve to prevent any entity or group of entities engaged in the retail sale of firm electric power from fulfilling all or part of their bulk power requirements through self-generation or through purchases from some other source other than Alabama Power Company. Alabama Power Company shall further, upon request and subject to reasonable terms and conditions, sell partial requirements power to any such entity. Nothing in this paragraph shall be construed as preventing an applicant from taking reasonable steps, in accord with general practice in the industry, to ensure that the reliability of its system is not endangered by any action called for herein.

(7) Alabama Power Company shall engage in wheeling for and at the request of any municipally-owned distribution system:

a. of electric energy from delivery points of Alabama Power Company to said distribution system(s); and

b. of power generated by or available to a distribution system as a result of its ownership or entitlement[2] in generating facilities, to delivery points of Alabama Power Company designated by the distribution system.

Such wheeling services shall be available with respect to any unused capacity on the transmission lines of Alabama Power Company, the use of which will not jeopardize Alabama Power Company's system. The contractual arrangements covering such wheeling services shall be determined in accordance with the principles set forth in Condition (3) herein.

[2] "Entitlement" includes, but is not limited to, power made available to an entity pursuant to an exchange agreement.

Alabama Power Company shall make reasonable provisions for disclosed transmission requirements of any distribution system(s) in planning future transmission. "Disclosed" means the giving of reasonable advance notification of future requirements by said distribution system(s) utilizing wheeling services to be made available by Alabama Power Company.

(8) The foregoing conditions shall be implemented in a manner consistent with the provisions of the Federal Power Act and the Alabama Public Utility laws and regulations thereunder and all rates, charges, services or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.

Southern Nuclear shall not market or broker power or energy from Joseph M. Farley Nuclear Plant, Units 1 and 2. Alabama Power Company shall continue to be responsible for compliance with the obligations imposed on it by the antitrust conditions contained in this paragraph 2.F. of the renewed license. Alabama Power Company shall be responsible and accountable for the actions of its agent, Southern Nuclear, to the extent said agent's actions may, in any way, contravene the antitrust conditions of this paragraph 2.F.

G. <u>Mitigation Strategy License Condition</u>

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

H. In accordance with the requirement imposed by the October 8, 1976 order of the United States Court of Appeals for the District of Columbia Circuit in Natural Resources Defense Council vs. Nuclear Regulatory Commission, No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission "shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of such proceeding herein," this renewed license shall be subject to the outcome of such proceedings.

I. This renewed operating license is effective as of the date of issuance and shall expire at midnight on June 25, 2037.

FOR THE NUCLEAR REGULATORY COMMISSION

J.E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A - Technical Specifications
2. Preoperational Tests, Startup Tests and Other Items Which Must Be Completed Prior to Proceeding to Succeeding Operational Modes
3. Appendix B - Environmental Protection Plan
4. Appendix C - Additional conditions

Date of Issuance: May 12, 2005

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, 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

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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 or incorporated below:

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. <u>213</u> are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144

(4) Deleted per Amendment 149

(5) Deleted per Amendment 144

to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense- in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2 below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2 above.

2. The licensee shall implement the modifications to its facility, as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-15-2310, dated April 25, 2016, to complete the transition to full compliance with 10 CFR 50.48(c) before the conclusion of the 1R28 Spring 2018 Refueling Outage as provided in SNC letter dated August 11, 2017. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-14-1273, dated August 29, 2014, within 180 days after NRC approval, except for items 30 and 32. Items 30 and 32 shall be implemented by February 6, 2018.

(7) Upon implementation of Amendment No. 213 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

(8) Deleted per Amendment 144
(9) Deleted per Amendment 144
(10) Deleted per Amendment 144
(11) Deleted per Amendment 144
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(13) Deleted per Amendment 144
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(17) Deleted per Amendment 144
(18) Deleted per Amendment 144
(19) Deleted per Amendment 144
(20) Deleted per Amendment 144
(21) Deleted per Amendment 144

(22) <u>Additional Conditions</u>

The Additional conditions contained in Appendix C, as revised through Amendment No. 137, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

(23) <u>Updated Final Safety Analysis Report</u>

The Updated Final Safety Analysis Report supplement shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The Southern Nuclear Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. Southern Nuclear shall complete these activities no later than March 31, 2021, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(24) Reactor Vessel Material Surveillance Capsules

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

D. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," and was submitted on May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 181, as supplemented by a change approved by License Amendment No. 195.

E. Deleted per Amendment 144

F. Alabama Power Company shall meet the following antitrust conditions:

(1) Alabama Power Company shall recognize and accord to Alabama Electric Cooperative (AEC) the status of a competing electric utility in central and southern Alabama.

(2) Alabama Power Company shall offer to sell to AEC an undivided ownership interest in Units 1 and 2 of the Farley Nuclear Plant. The percentage of ownership interest to be so offered shall be an amount based on the relative sizes of the respective peak loads of AEC and Alabama Power Company (excluding from the Alabama Power Company's peak load that amount imposed by members of AEC upon the electric system of Alabama Power Company) occurring in 1976.

The price to be paid by AEC for its proportionate share of Units 1 and 2, determined in accordance with the foregoing formula, will be established by the parties through good faith negotiations. The price shall be sufficient to fairly reimburse Alabama Power Company for the proportionate share of its total costs related to the Units 1 and 2 including, but not limited to, all costs of construction, installation, ownership and licensing, as of a date, to be agreed to by the two parties, which fairly accommodates both their respective interests. The offer by Alabama Power Company to sell an undivided ownership interest in Units 1 and 2 may be conditioned, at Alabama Power Company's option, on the agreement by AEC to waive any right of partition of the Farley Plant and to avoid interference in the day-to-day operation of the plant.

(3) Alabama Power Company will provide, under contractual arrangements between Alabama Power Company and AEC, transmission services via its electric system (a) from AEC's electric system to AEC's off-system members; and (b) to AEC's electric system from electric systems other than Alabama Power Company's, and from AEC's electric system to electric systems other than Alabama Power Company's. The contractual arrangements covering such transmission services shall embrace rates and charges reflecting conventional accounting and ratemaking concepts followed by the Federal Energy Regulatory Commission (or its successor in function) in testing the reasonableness of rates and charges for transmission services. Such contractual arrangements shall contain provisions protecting Alabama Power Company against economic detriment resulting from transmission line or transmission losses associated therewith.

(4) Alabama Power Company shall furnish such other bulk power supply services as are reasonably available from its system.

(5) Alabama Power Company shall enter into appropriate contractual arrangements amending the 1972 Interconnection Agreement as last amended to provide for a reserve sharing arrangement between Alabama Power Company and AEC under which Alabama Power Company will provide reserve generating capacity in accordance with practices applicable to its responsibility to the operating companies of the Southern Company System. AEC shall maintain a minimum level expressed as a percentage of coincident peak one-hour kilowatt load equal to the percent reserve level similarly expressed for Alabama Power Company as determined by the Southern Company System under its minimum reserve criterion then in effect. Alabama Power Company shall provide to AEC such data as needed from time to time to demonstrate the basis for the need for such minimum reserve level.

(6) Alabama Power Company shall refrain from taking any steps, including but not limited, to the adoption of restrictive provisions in rate filings or negotiated contracts for the sale of wholesale power, that serve to prevent any entity or group of entities engaged in the retail sale of firm electric power from fulfilling all or part of their bulk power requirements through self-generation or through purchases from some other source other than Alabama Power Company.

Alabama Power Company shall further, upon request and subject to reasonable terms and conditions, sell partial requirements power to any such entity. Nothing in this paragraph shall be construed as preventing an applicant from taking reasonable steps, in accord with general practice in the industry, to ensure that the reliability of its system is not endangered by any action called for herein.

(7) Alabama Power Company shall engage in wheeling for and at the request of any municipally-owned distribution system:

a. of electric energy from delivery points of Alabama Power Company to said distribution system(s); and

b. of power generated by or available to a distribution system as a result of its ownership or entitlement[2] in generating facilities, to delivery points of Alabama Power Company designated by the distribution system.

Such wheeling services shall be available with respect to any unused capacity on the transmission lines of Alabama Power Company, the use of which will not jeopardize Alabama Power Company's system. The contractual arrangements covering such wheeling services shall be determined in accordance with the principles set forth in Condition (3) herein.

Alabama Power Company shall make reasonable provisions for disclosed transmission requirements of any distribution system(s) in planning future transmission. "Disclosed" means the giving of reasonable advance notification of future requirements by said distribution system(s) utilizing wheeling services to be made available by Alabama Power Company.

(8) The foregoing conditions shall be implemented in a manner consistent with the provisions of the Federal Power Act and the Alabama Public Utility laws and regulations thereunder and all rates, charges, services or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.

Southern Nuclear shall not market or broker power or energy from Joseph M. Farley Nuclear Plant, Units 1 and 2. Alabama Power Company shall continue to be responsible for compliance with the obligations imposed on it by the antitrust conditions contained in this paragraph 2.F. of the

[2] "Entitlement" includes, but is not limited to, power made available to an entity pursuant to an exchange agreement.

renewed license. Alabama Power Company shall be responsible and accountable for the actions of its agent, Southern Nuclear, to the extent said agent's actions may, in any way, contravene the antitrust conditions of this paragraph 2.F.

G. The facility requires relief from certain requirements of 10 CFR 50.55a(g) and exemptions from Appendices G, H and J to 10 CFR Part 50. The relief and exemptions are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 5. They are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, the relief and exemptions are hereby granted. With the granting of these relief and exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

H. Southern Nuclear shall immediately notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. <u>Mitigation Strategy License Condition</u>

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

J. Alabama Power Company shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

K. This renewed operating license is effective as of the date of issuance and shall expire at midnight on March 31, 2041.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachment:

1. Appendix A - Technical Specifications (NUREG-0697, as revised)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional conditions

Date of Issuance: May 12, 2005

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used to determine the DOSE EQUIVALENT I-131 shall be performed using Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
$\bar{E}$ — AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration/Pressurization System (CREFS)

LCO 3.7.10 Two CREFS trains shall be OPERABLE.

-------------------------------------- NOTE --
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREFS train inoperable for reasons other than Condition B.	A.1 Restore CREFS train to OPERABLE status.	7 days
B. One or more CREFS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions.	Immediately
	AND	
	B.2. Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND	
	B.3 Restore CRE boundary to OPERABLE status.	90 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------- Be in MODE 4.	6 hours 12 hours
D. Two CREFS trains inoperable in MODE 1, 2, 3, OR 4 for reasons other than Condition B.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1 Place OPERABLE CREFS train in emergency recirculation mode. OR E.2.1 Suspend CORE ALTERATIONS. AND E.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREFS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS. OR One or more CREFS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	F.1 Suspend CORE ALTERATIONS. AND F.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREFS Pressurization train with the heaters operating and each CREFS Recirculation and Filtration train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.10.3	----------------------------NOTE------------------------------- Not required to be performed in MODES 5 and 6. --- Verify each CREFS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

--NOTE--
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative control.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

5.5.17 Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.18 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

(continued)

5.5 Programs and Manuals

5.5.18 Control Room Envelope Habitability Program (continued)

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 216 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 213 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By application dated November 22, 2016 (Reference 1), as supplemented by letters dated May 23, 2017 (Reference 2); June 8, 2017 (Reference 3); September 7, 2017 (Reference 4); November 21, 2017 (Reference 44); and December 18, 2017 (Reference 45), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a request to change the Joseph M. Farley Nuclear Plant (FNP or Farley), Units 1 and 2, Technical Specifications (TSs).

Specifically, SNC requested to:

1. Revise the licensing basis to support a full scope implementation of the Alternative Source Term (AST) radiological analysis methodology as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term" (Reference 5), for the design basis accidents (DBA) discussed in Section 3.0 below, as described in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (Reference 6).

2. Incorporate Technical Specifications Task Force (TSTF) Traveler, TSTF-448, Revision 3, "Control Room Habitability" (Reference 7); and

3. Incorporate TSTF-312, Revision 1, "Administrative Control of Containment Penetrations" (Reference 8).

The supplemental letters dated May 23, June 8, September 7, November 21, and December 18, 2017, provide additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2017 (82 FR 160) (Reference 9).

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

The licensee proposed to revise FNP licensing basis to support a full-scope application of an AST methodology. This reanalysis involves several changes in selected analysis assumptions. As part of the implementation of the AST, the Total Effective Dose Equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11. This will also replace the whole body (and its equivalent to any part of the body) dose criteria of 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, "Control Room" (Reference 10).

In addition, the licensee proposed to revise TS 3.7.10, TS 3.9.3, and TS 5.5.18.

The proposed TS pages 3.7.10-1, 3.7.10-2, and 3.7.10-3 can be found in the June 8, 2017, supplement. The proposed TS pages 3.7.10-4 and 3.9.3-1 can be found in the November 22, 2016, application. The proposed TS pages 5.5-15 and 5.5-16 can be found in the November 21, 2017, supplement.

The proposed changes to TS 3.7.10 are highlighted in bold below. The deletions can be found in the markup pages provided by the licensee.

3.7.10 Control Room **Emergency Filtration/Pressurization System (CREFS)**

LCO 3.7.10 Two **CREFS trains shall be OPERABLE**.

-- NOTE --
The **control room envelope** (CRE) **boundary** may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREFS train inoperable **for reasons other than Condition B.**	A.1 Restore CREFS train to OPERABLE status	7 days

B. One or more CREFS trains inoperable due to inoperable CRE **boundary in MODE 1, 2, 3, or 4.**	B.1	Initiate **action to implement** mitigating actions	Immediately
	AND		
	B.2	**Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.**	24 hours
	AND		
	B.3	Restore CRE **boundary** to OPERABLE status.	**90** days
D. Two CREFS trains inoperable in MODE 1, 2, 3, OR 4 **for reasons other than Condition B.**	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours
F. Two CREFS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS **OR** **One or more CREFS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.**	F.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	F.2	Suspend movement of irradiated fuel assemblies.	Immediately

No changes were made to TS 3.7.10 Condition E.

The proposed changes to surveillance requirements (TS page 3.7.10-4) are in bold below:

SURVEILLANCE		FREQUENCY
SR 3.7.10.4	**Perform required CRE unfiltered air Inleakage testing in accordance with the Control Room Envelope Habitability Program.**	**In accordance with the Control Room Envelope Habitability Program**

In addition, TS pages 3.7.10 -1 thru 3.7.10-4, each page title changes from "Control Room 3.7.10" to "CREFS 3.7.10."

The proposed change to TS 3.9.3, "Containment Penetration" adds the following note:

> -- NOTE -----------------------------------
> Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative control.
> --

TS 5.5.18, "Control Room Integrity Program (CRIP)" is proposed to be revised in its entirety to state the following:

5.5.18 <u>Control Room Envelope Habitability Program</u>

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem [roentgen equivalent man] total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be

trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d.

In December 2017, the NRC staff identified the definition of "Dose Equivalent I-131" in TS 1.1 "Definitions" on TS page 1.1-2, was not consistent with the calculation performed in the submittal. As a result, on December 18, 2017, SNC submitted a supplement to revise the definition of "Dose Equivalent I-131," from, in part:

> The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977.

to, in part:

> The CEDE dose conversion factors used to determine the DOSE EQUIVALENT I-131 shall be performed using Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose conversion Factors for Inhalation, Submersion, and Ingestion."

The NRC staff finds the revised definition of Dose Equivalent I-131 acceptable because the dose conversion factors used in the determination of dose equivalent I-131 are consistent with the dose conversion factors used in the dose consequence analyses, submitted by the licensee in the application and are consistent with RG 1.183. The NRC staff concluded that the revised TS page 1.1-2 was a conforming change to what had already been submitted, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2017 (82 FR 160).

2.2 AST Regulatory Evaluation

The licensee's request was pursuant to 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs. FNP's current DBA radiological consequence analyses are based on the source term from U.S. Atomic Energy Commission Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (Reference 11).

The NRC staff evaluated the licensee's analysis of the radiological consequences of the affected DBAs for implementation of the AST methodology, and the associated changes to the TSs proposed by the licensee, against the radiological dose requirements specified in 10 CFR 50.67(b)(2), and dose limits specified in 10 CFR Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," Criterion 19, "Control Room," Section 50.67(b)(2) states:

> The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.
>
> [2] The use of 0.25 SV (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a Reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

The NRC staff's AST evaluation is based upon the following regulations, RGs, and standards:

- 10 CFR Part 50.67, "Accident Source Term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants": GDC 19, "Control room."
- NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
- NRC RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 1, January 2007 (Reference 13).
- NUREG-0800, SRP Section 6.4, "Control Room Habitability System," Revision 3, March 2007 (Reference 14):
  - Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (Reference 15), and
  - Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA's radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the accident radiation dose values in 10 CFR 50.67, and the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of Standard Review Plan (SRP) Section 15.0.1 (Reference 12). The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

2.3 TSTF-448 Regulatory Evaluation

The following FNP GDCs 1, 2, 3, 4, 5, and 19 apply to CRE habitability and TSTF-448. A summary of these FNP GDCs is provided below:

> FNP GDC 1, "*Quality standards and records*," requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
>
> FNP GDC 2, "*Design basis for protection against natural phenomena*," requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.
>
> FNP GDC 3, "*Fire protection*," requires SSCs important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.
>
> FNP GDC 4, "*Environmental and missile design bases,*" requires SSCs important to safety to be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs [Loss of Coolant Accident].
>
> FNP GDC 5, "*Sharing of structures, systems, and components*," requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
>
> FNP GDC 19, "*Control room*," requires that control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Prior to incorporation of TSTF-448, Revision 3, the STS requirements addressing CRE boundary operability resided only in NUREG-1431 (Reference 16), TS 3.7.10, "Control Room Emergency Filtration System (CREFS)."

In NUREG-1431, TS 3.7.10, the surveillance requirement associated with demonstrating the operability of the CRE boundary requires verifying that one CREFS train can maintain a positive pressure of 0.125 inches water gauge, relative to the adjacent building during the pressurization mode of operation at a makeup flow rate of 300 cfm (cubic feet per minute). Facilities that pressurize the CRE during the emergency mode of operation of the CREFS have similar surveillance requirements. Other facilities that do not pressurize the CRE have only a system flow rate criterion for the emergency mode of operation. Regardless, the results of ASTM E741 (Reference 17) tracer gas tests to measure CRE unfiltered inleakage at facilities indicated that the differential pressure surveillance (or the alternative surveillance at non-pressurization facilities) is not a reliable method for demonstrating CRE boundary operability. That is, licensees were able to obtain differential pressure and flow measurements satisfying the SR limits even though unfiltered inleakage was determined to exceed the value assumed in the safety analyses.

In addition to an inadequate surveillance requirement, the action requirements of this specification was ambiguous regarding CRE boundary operability in the event CRE unfiltered inleakage is found to exceed the analysis assumption. The ambiguity stemmed from the view that the CRE boundary may be considered operable but degraded in this condition, and that it would be deemed inoperable only if calculated radiological exposure limits for CRE occupants exceeded a licensing basis limit; e.g., as stated in GDC-19, even while crediting compensatory measures.

NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety" (AL 98-10) (Reference 18), states that "the discovery of an improper or inadequate TS value or required action is considered a degraded or nonconforming condition," which is defined in NRC Inspection Manual Chapter 0326; also see guidance in RIS 2005-20 (Reference 19). Imposing administrative controls in response to an improper or inadequate TS is considered an acceptable short-term corrective action. The [NRC] staff expects that, following the imposition of administrative controls, an amendment to the [inadequate] TS, with appropriate justification and schedule, will be submitted in a timely fashion."

Licensees that have found unfiltered inleakage in excess of the limit assumed in the safety analyses and have yet to either reduce the inleakage below the limit or establish a higher bounding limit through re-analysis, and have implemented compensatory actions to ensure the safety of CRE occupants, pending final resolution of the condition, consistent with RIS 2005-20. However, based on GL 2003-01 (Reference 20) and AL 98-10, the NRC staff expects each licensee to propose TS changes that include a surveillance to periodically measure CRE unfiltered inleakage in order to satisfy 10 CFR 50.36(c)(3), "Surveillance requirements" defined as "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and *that limiting conditions for operation will be met.*" (Emphasis added.)

The NRC staff also expects facilities to propose unambiguous remedial actions, consistent with 10 CFR 50.36(c)(2), for the condition of not meeting the limiting conditions for operation (LCO) due to an inoperable CRE boundary. The action requirements should specify a reasonable completion time to restore conformance to the LCO before requiring a facility to be shut down. This completion time should be based on the benefits of implementing mitigating actions to ensure CRE occupant safety and sufficient time to resolve most problems anticipated with the CRE boundary, while minimizing the chance that operators in the CRE will need to use mitigating actions during accident conditions.

2.4 TSTF-312 Regulatory Evaluation

The following FNP GDCs 16, 19, 54, and 56 apply to containment and TSTF-312. A summary of these FNP GDCs are provided below:

> FNP GDC 16, "*Containment design*," requires that the reactor containment and associated systems are provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.
>
> FNP GDC 19, "*Control room*" (see Section 2.3 above).
>
> FNP GDC 54, "*Piping systems penetrating containment*," requires that piping systems penetrating primary reactor containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems are designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.
>
> FNP GDC 56, "*Primary containment isolation*," requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis, namely: ...

The regulations in 10 CFR 50.67, "Accident source term," provide the limitations of radiological dose for an individual located at any point on the boundary of the exclusion area, for an individual located at any point on the outer boundary of the low population zone, and provide for the radiological protection of the personnel occupying the control room under accident conditions.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance on acceptable applications of alternative source terms. In Appendix B of this RG, guidance is provided on evaluating the radiological consequences of a fuel handling accident (FHA).

2.5 Atmospheric Dispersion Factors Regulatory Evaluation

The NRC staff's evaluation of the proposed atmospheric dispersion factors is based upon the following RGs, codes, and standards:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room"
- NRC NUREG–0800, Standard Review Plan (SRP) Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident" (Reference 21)

- NRC RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants" (Reference 22)

- NRC RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Reference 23)

- NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

- NRC RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Reference 24)

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of DBAs

The licensee has proposed a licensing basis change for its offsite and control room DBA dose consequence analysis for FNP. The proposed change will implement an AST methodology for determining DBAs offsite and control room doses. For full implementation of the AST DBAs analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and GDC 19.

As discussed in RG 1.183, Regulatory Position 1.2.1, full implementation is a modification of the facility design basis that addresses all characteristics of the AST, including composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in this application, but also to all future design basis dose consequence analyses at FNP. At a minimum for full implementation of the AST, the DBA LOCA must be reanalyzed. Since, upon issuance of this license amendment request (LAR), the AST and TEDE criteria will become part of the design basis for FNP, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, Tests, and Experiments," or unless the new application involved a change to a TS. However, a change from an approved AST to a different AST that is not approved for use at FNP would require a license amendment under 10 CFR 50.67.

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS reactor or secondary coolant specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the plant-specific proposed applications of an AST.

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. Also, the licensee's AST analyses were based on the pressurized-water reactor (PWR) DBAs identified in RG 1.183 that could potentially result in significant control room and offsite doses.

The licensee has performed its evaluation based on full implementation of the AST as defined in RG 1.183, with the exception of the equipment qualification (EQ). The licensee has determined that the current TID-14844 accident source term will remain the licensing basis for EQ. Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memorandum dated April 30, 2001, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" (Reference 25), and in NUREG-0933, Supplement 25, June 2001, "A Prioritization of Generic Safety Issues" (Reference 26). The conclusion of Generic Issue 187 states the following:

> The staff concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary.

Therefore, in consideration of the above-cited References, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at FNP. Guidance in RG 1.183, Regulatory Position 4.3, under Other Dose Consequences, states that:

> The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2) [Reference 27]. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE.

The NRC staff requested the licensee provide additional information describing how Regulatory Position 4.3 has been assessed. In SNC letter dated May 23, 2017, in response to RAI No. 38, the licensee stated:

> SNC's LAR seeks AST implementation for radiological consequences of major DBAs, specifically: Loss of Coolant Accident, Fuel Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Control Rod Ejection, and Locked Rotor Analysis. It also seeks to implement TSTF-312 and TSTF-448. There are no physical changes to the plant being proposed.
>
> RG 1.183, Regulatory Position 1.3.2, "Re-Analysis Guidance," states:
>
>> The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification

> changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

The only changes of assumptions or inputs are for the DBAs listed above, and those analyses' results have been submitted. All other existing analyses remain the same and, as mentioned above, there are no physical changes to the plant being proposed. If there are other changes *(e.g.,* procedural changes) after the approval of the LAR, then SNC will follow regulatory requirements *(e.g.,* 10 CFR 50.59).

The NRC staff reviewed the licensee's response above and finds that RG 1.183, Regulatory Position 4.3 has been assessed sufficiently for FNP.

A full implementation of the AST is proposed for FNP. Therefore, to support the licensing and plant operation changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

- LOCA
- FHA in the Containment
- FHA in the spent fuel pool area of the auxiliary building
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)

The DBA dose consequence analyses evaluated the integrated TEDE dose at the exclusion area boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) during the entire period of the passage of the radioactive cloud resulting from postulated release of fission products, and the integrated dose to a FNP control room (CR) operator were evaluated for the duration of the accident. The LOCA dose consequence analyses was performed by the licensee using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal and Dose Estimation" (RADTRAD) Version 3.10. The FHA dose consequence analyses was performed by the licensee using RADTRAD Version 3.03. The development of the RADTRAD radiological consequence computer code was sponsored by the NRC, as described in NUREG/CR-6604, and was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performs independent confirmatory dose evaluations using the RADTRAD computer code.

The MSLB, SGTR, CREA, and LRA dose consequence analyses were performed by the licensee using LocaDose. LocaDose is a proprietary Bechtel software that calculates radioactive isotope activities within regions, radioactive releases from regions, doses and dose rates within regions for humans and equipment, and inhalation and immersion doses to plant personnel. The NRC staff performed independent confirmatory dose evaluations of these accidents using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance guidelines from RG 1.183, are shown in Section 4.0, Table 1 of this SE.

Each DBA radiological source term used in the AST analyses was developed based on a core power level of 2831 megawatts thermal (MWt). The core power level represents the licensed

power of 2775 MWt with a 2 percent increase to account for measurement uncertainties. The use of 2831 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 2775 MWt and is, therefore, acceptable to the NRC staff for use in the full implementation of the AST at FNP. In addition, to account for potential cycle-to-cycle variations, the licensee applied margin factors to the core inventory. The margin factors for the various isotopes in each of the accidents analyzed vary from analysis to analysis. The margin factors add conservatism and margin for isotopes that are critical to the calculation of dose consequences for each DBA. Each margin factor adds extra margin above the reactor concentrations for an equilibrium core.

Guidance in RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the emergency core cooling system (ECCS) evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) [Reference 28] or ORIGEN-ARP (Ref. 18) [Reference 29].

In accordance with RG 1.183, the licensee developed the equilibrium core activity inventory using the ORIGEN 2. The determination of core inventory is dependent upon full power, core average conditions. The NRC staff finds this approach to be consistent with current regulatory guidance and therefore, acceptable.

The licensee used committed effective dose equivalent and effective dose equivalent dose conversion factors (DCFs) from Federal Guidance Reports (FGR) 11 and 12 to determine the TEDE dose in accordance with AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is, therefore, acceptable to the NRC staff.

3.2 LOCA

A DBA LOCA is a failure of the reactor coolant system (RCS) that results in the loss of reactor coolant which, if not mitigated, could result in fuel damage including core melt. Analyses are performed using a spectrum of RCS break sizes to evaluate fuel and emergency core cooling system (ECCS) performance. A large break LOCA is postulated as the failure of the largest pipe in the RCS. RG 1.183 establishes the large-break LOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective in preventing core damage.

The fission product release is assumed to occur in phases over a 2-hour period. When using the AST for the evaluation of a design-basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the

radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

In the evaluation of the LOCA design-basis radiological analysis the licensee considered dose contributions from the following potential activity release pathways:

- Containment leakage directly to the atmosphere.
- Release from the containment mini-purge.
- Engineered safety feature (ESF) systems leakage outside of containment.
- Refueling water storage tank (RWST) leakage to the atmosphere.

3.2.1 LOCA Source Term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position 3, regarding the fission product inventory, release fractions, timing of the release phases, radionuclide composition, and chemical form for the evaluation of the LOCA. For the DBA LOCA, the licensee uses the core average inventory, as discussed above, and assumes that all the fuel assemblies in the core are affected. The LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a result of these removal mechanisms a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gases and soluble fission products such as iodine and cesium, but not noble gases. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.

The containment sump pH analysis was previously reviewed and approved by the NRC staff in License Amendment Nos. 166 and 158 for FNP (Reference 30). The licensee's analysis concluded and the NRC staff confirmed that the pH of the containment sump is maintained equal to or greater than 7.0 after the onset of the spray recirculation mode and that the post-LOCA dose consequence analyses need not consider iodine re-evolution from the sump fluid in accordance with RG 1.183. The NRC staff verified that this LAR did not impact the containment sump pH analysis previously reviewed and approved and determined that it is applicable to the AST.

3.2.2 Assumptions on Transport in the Primary Containment

3.2.2.1 Containment Mixing, Natural Deposition and Leak Rate

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing as specified in RG 1.183 with the termination of the release into containment set at the end of the early in-vessel phase.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. RG 1.183 Appendix A position 3.2 states:

> Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2) [Reference 31). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3) [new Reference 32]. The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

In this LAR, Table B of Enclosure 5 states FNP's conformance with RG 1.183 Appendix A. Table B states that FNP's analysis for RG 1.183 regulatory position 3.2 is, "Conforms - An aerosol natural deposition rate of 0.1 h^{r-1} [hr^{-1}] is assumed based upon values presented in Section VI of NUREG/CR-6189." The NRC staff identified that the RADTRAD modeling used is not consistent with the inputs and assumptions in the LAR, and the LAR is not consistent between the various enclosures and tables. Therefore, the NRC staff requested that the licensee explain what assumptions are being used for natural deposition of elemental, organic and aerosol iodine in the unsprayed and sprayed portion of the containment and explain how the removal coefficient(s) were calculated to determine if the assumptions are consistent with RG 1.183. In addition, the NRC staff asked for enough detail to allow the NRC staff to confirm the methodology is consistent with NUREG-0800 SRP Chapter 6.5.2 and/or NUREG/CR-6189 as applicable.

In SNC letter dated May 23, 2017, SNC stated that Section III.4.C.i of NUREG-0800, Section 6.5.2 was used to calculate the wall deposition of elemental iodine. Using the equation for determining the removal coefficient of elemental iodine by wall deposition during injection in Section III.4.C.i of NUREG-0800 SRP Chapter 6.5.2 the licensee calculated the removal coefficient for natural deposition of elemental iodine to be 2.88 hr^{-1}. The licensee credits the wall deposition of elemental iodine throughout the event and credits aerosol deposition only in unsprayed regions of containment.

Additionally, the licensee credits aerosol deposition in the unsprayed region of containment at the start of the event and in the sprayed compartment of containment only after sprays have been terminated at 8 hours. The licensee stated, "An aerosol natural deposition rate of 0.1 hr^{-1} is applied based upon values presented in Section VI NUREG/CR-6189." However, SNC's response did not explain how this value is consistent with Section VI NUREG/CR-6189. The NRC staff reviewed NUREG/CR-6189 and determined that an aerosol natural deposition rate of 0.1 hr^{-1} does not seem to be consistent with NUREG/CR-6189 Section VI because it overestimates the aerosol natural deposition rate in four out of the five gap time intervals stated in Table 36 of NUREG/CR-6189. NUREG/CR-6189 provides a simplified model of aerosol removal by natural processes in reactor containments that applies to both the CREA in containment and LOCA. The simplified approach used in NUREG/CR-6189 does not vary with different DBAs. Therefore, the aerosol natural deposition rates calculated for the CREA also apply to the LOCA analysis. In response to a different request for information in the letter, SNC provided the calculated effective decontamination coefficient correlations, also known as the aerosol natural deposition rates, for the five gap time intervals stated in Table 36 of NUREG/CR-6189. These deposition rates were provide for the CREA analysis. The NRC staff reviewed SNC's response and the calculated aerosol natural deposition rates and determined

that they are consistent with NUREG/CR-6189. In the CREA analysis, SNC chose to use the lowest aerosol natural deposition rate, which is a conservative assumption because it removes the least amount of aerosols from the containment atmosphere. However, in the LOCA analysis, SNC is applying an aerosol natural deposition rate of 0.1 hr^{-1} for all five gap time intervals, which is not a conservative assumption. The calculated aerosol natural deposition rates for the LOCA analysis should reflect each of the NUREG/CR-6189 gap time intervals. It is non-conservative to apply a later time-period (13680 to 49680) aerosol natural deposition rate to the earlier and later time-periods (0-1800, 1800-6480, 6480-13680, and 49680-80000). The NRC staff requested that the licensee explain the technical safety basis for applying an aerosol natural deposition rate of 0.1 hr^{-1}. In SNC's letter dated September 7, 2017, SNC stated:

> In the evaluation of the dose consequences for the LOCA, a natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region when sprays are not operating. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," December 1983, documents results from Containment Systems Experiment testing. These tests show that settling of aerosols due to gravity is the dominant natural mechanism for fission product retention. This report finds that significant removal by sedimentation would be expected even at very low particulate concentrations. Figure 4-2 of IDCOR Program Technical Report 11.3 shows a ten-fold reduction in the airborne cesium concentration over a 7-hour period at relatively low concentrations. This represents an aerosol removal rate of 0.33 hr^{-1}. A more conservative value of 0.1 hr^{-1} is used in the analysis.
>
> Examples where this same technical basis has been applied for other approved LAR submittals include those for the St. Lucie Unit 2 License Amendment No. 152 in September 2008 (ADAMS Accession No. ML082060400) and for the Palisades Nuclear Plant License Amendment No. 226 in September 2007 (ADAMS Accession No. ML072470667).

The NRC staff reviewed the licensee's response including the cited past precedence, and determined that this methodology is less conservative than that used in NUREG-6189. Since the licensee did not credit wall deposition of organic iodine at any time in the LOCA analysis and the licensee's resultant dose is less than 1 percent lower than the calculated NUREG-6189 dose, the NRC staff finds the difference to be insignificant, and therefore, is acceptable

Guidance in RG 1.183, Regulatory Position 3.7 states that the primary containment should be assumed to leak at the peak pressure TS leak rate for the first 24 hours and that for pressurized water reactors, the leak rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. Accordingly, the licensee assumed a containment leak rate of 0.15 percent per day for the first 24 hours, after which the containment leak rate is reduced to 0.075 percent per day for the duration of the accident consistent with FNP's TSs. The licensee assumes the leakage is from both the sprayed and unsprayed regions of the containment to the environment.

3.2.2.2 Containment Spray Assumptions

Guidance in RG 1.183, Appendix A, Regulatory Position 3.3 states that:

> ....The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow

> exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

For FNP, the volume of the sprayed region is 1,668,660 cubic feet (ft^3) and the volume of the unsprayed region is 361,340 ft^3. A flow rate of 12,045 ft^3 per minute is used between the sprayed and unsprayed volume. In accordance with RG 1.183, Appendix A, section 3.3, the licensee used the mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building of two turnovers of the unsprayed regions per hour. Since the sprayed region is less than 90 percent of the total containment volume the licensee used a two-volume model to represent the sprayed and unsprayed regions of the containment. For FNP, the containment spray in the injection mode is initiated at 90 seconds after the LOCA initiation and terminates at 8 hours.

Using the guidance from NUREG-0800, SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System," and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays" (Reference 33), the licensee determined the aerosol removal rate from the effects of the containment spray system during injection mode is 5.45 per hour and during recirculation mode.

Section 3.3 of Appendix A of RG 1.183 states:

> The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays....

Using the guidance from SRP 6.5.2, the licensee determined the elemental iodine removal rate from the effects of the containment spray system in injection mode is 13.7 per hour. However, in accordance with the guidance in SRP 6.5.2, the licensee limited the removal rate constant for elemental iodine to zero when the elemental iodine DF reaches a value of 200. During injection mode, the licensee determined the aerosol removal rate is 5.45 per hour. During recirculation spray operation, the licensee determined the aerosol removal rate is 5.03 per hour. The aerosol removal coefficient is reduced by a factor of 10 when the aerosol DF reaches 50. The licensee applied the removal rates in the radiological dose analysis from the time of spray actuation until 8 hours post-LOCA. No credit is taken for organic iodine removal in the containment.

The NRC staff has reviewed the licensee's application of credit for iodine removal from the operation of the containment spray system and found that the analysis follows the applicable regulatory guidance, is conservative and is, therefore, acceptable.

3.2.3 Assumptions on ESF System Leakage

To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach as described in RG 1.183. This approach assumes with the exception of noble gases,

all the fission products released from the fuel to the containment instantaneously and homogeneously mix in the containment sump water at the time of release from the core. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. This source term assumption is conservative in that 100 percent of the radioiodine released from the fuel is assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections.

Guidance in RG 1.183, Appendix A, Regulatory Position 5.5, states that, if the temperature of the leakage is less than 212 degrees Fahrenheit (°F) or the calculated flash fraction is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid. The licensee has determined that the maximum temperature of the recirculation fluid is 265°F and that the flash fraction is 5.5 percent, which is less than 10 percent; therefore, per RG 1.183, the licensee assumes that 10 percent of the iodine activity associate with this leakage is airborne. In addition, in accordance with RG 1.183, for ESF leakage, the licensee assumes that the chemical form of the released iodine is 97 percent elemental and 3 percent organic.

For the LOCA analysis of ESF leakage, the licensee assumed ECCS leakage is 40,000 cubic centimeters per hour (cc/hr) for leakage of sump water outside of containment into the auxiliary building, which represents two times the maximum permitted leakage of 20,000 cc/hr, as specified in RG 1.183, Appendix A, Item 5.2. As stated above, actual ECCS leakage starts when the recirculation phase of the accident begins.

3.2.3.1 Assumptions on ESF System Back-Leakage to the Refuel Water Storage Tank

Although the RWST is isolated during recirculation, design leakage through ECCS valves provides a pathway for back leakage of the containment sump water to the RWST. The RWST is vented to the atmosphere. Since this release path represents a bypass of the containment, the radiological dose consequences are considered. The concentration of radionuclides in the containment sump water is as modeled above for ESF leakage. FNP assumes that containment sump water leaks into the RWST at a rate of 2 gallons per minute (gpm) representing two times the maximum permitted back leakage of 1 gpm, as specified in RG 1.183, Appendix A, Item 5.2. The back leakage to the RWST starts when recirculation occurs in the ECCS systems and continues for the 30 day duration of the event. The licensee assumed that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

The licensee used conservative assumptions to evaluate the RWST back leakage contribution to the LOCA dose, and therefore, the NRC staff finds this evaluation acceptable for the AST LOCA analysis.

3.2.4 Assumptions on Containment Purging

The licensee evaluated the radiological consequences of containment leakage via the mini-purge system, which are assumed to be open to the extent allowed by FNP TS at the initiation of the LOCA, and are terminated as part of the containment isolation. The assumed volumetric flow rate from the mini-purge system is 2,850 cubic feet per minute (cfm) and is released directly to the environment until terminated by the containment isolation at 30 seconds post-LOCA.

During this time period of 30 seconds following accident onset, the licensee assumes that fuel failure has not occurred. This assumption follows the guidance in Table 4 of RG 1.183, which indicates that the initial release of the RCS into containment for a PWR would occur within the first 30 seconds of the accident prior to the onset of fuel damage. Consistent with RG 1.183 RCS radionuclide concentrations for the AST analysis is based on the TS RCS equilibrium activity, which includes 1 percent fuel defects. Therefore, this conservative approach for the evaluation of the radiological dose consequence is acceptable to the NRC staff.

The licensee used conservative assumptions to evaluate the containment purge contribution to the LOCA dose, and therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.2.5 Control Room Habitability

The FNP control room is common to Units 1 and 2 and located in auxiliary building. The CREFS is designed to maintain the control room envelope at a positive pressure relative to the surrounding area, following postulated accidents with the exception of toxic gas/smoke releases. The CREFS is activated on a containment isolation actuation signal (CIAS) and/or high radiation in the normal outside air intakes. CREFS is designed to automatically isolate the normal air system and start both trains of the control room air conditioning system, pressurization system, and filtration system upon a CIAS. Detection of high radiation levels automatically isolates the normal air system, but the pressurization and filtration systems must be manually initiated. The design pressurization flow is 375 cfm and is drawn from one of the two emergency intakes. The control room unfiltered inleakage is 325 cfm and includes 10 cfm for control room ingress and egress.

The control room pressurization flow is routed through charcoal and high efficiency particulate air filters. Furthermore, during postulated accident conditions, part of the control room pressurization flow is recirculated and is filtered through charcoal and high efficiency particulate air filters (HEPA) at a flow rate of 2,700 cfm.

3.2.5.1 Control Room Ventilation Assumptions

The licensee's assumption of 325 cfm unfiltered inleakage is validated by inleakage testing conducted during the most recent tracer gas test on February 8, 2016. In preparation for the LAR, SNC re-performed the LOCA and FHA analyses and chose to use a higher unfiltered inleakage (325 cfm). In comparison to the most recent inleakage tests, performed in February of 2016, 300 or 325 cfm is very conservative. In the pressurization mode, the worst as-tested leakage was 54 cfm and for the isolation mode, the worst as-tested leakage was 41 cfm. Therefore, the assumed unfiltered inleakage is many times higher than the as-tested unfiltered inleakage.

The CREFS automatically transfers to the pressurization mode of operation after the CIAS. The licensee determined that the time to generate the CIAS will be 27 seconds following LOCA initiation. The licensee assumes the signal processing time for a CIAS from a safety injection (SI) signal is less than one second. From the time the controlling instrument (which can vary depending upon the accident) senses the condition that would initiate a SI or CIAS signal, less than one second passes before pressurization begins. The control room becomes pressurized in less than 1 minute from the accident.

The CREFS is designed to maintain the control room envelope at a positive pressure relative to the surrounding area, following a postulated LOCA. Upon CREFS initiation the normal outside air intakes for the control room are automatically isolated. Pressurization flow of 375 cfm is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, during postulated accident conditions, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 98.5 percent.

3.2.5.2 Direct Shine Dose Evaluations

Guidance in RG 1.183 regulatory position 4.2.1 states:

> The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:
>
> - Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
> - Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
> - Radiation shine from the external radioactive plume released from the facility,
> - Radiation shine from radioactive material in the reactor containment,
> - Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.

Table A of Enclosure 5 states FNP's conformance with RG 1.183 Section C. Table A states that FNP analysis for RG 1.183 regulatory position 4.2.1 is:

> Conforms - The analyses consider the applicable sources of contamination to the control room atmosphere for each event.
>
> With respect to external and containment shine sources and their impact on control room doses, the physical design of the control room envelope and the surrounding auxiliary building provide more than 18" of concrete shielding between the operators and shine sources in all directions around the control room.
>
> The Control Room Emergency Filtration System filters are located outside of and above the control room envelope. The control room ceiling is approximately 24"

thick. Accordingly, shielding from the walls and the filter unit casings prevents an appreciable dose to the operators during the accident.

The control room is surrounded by the Auxiliary Building (and so does not abut the containment), and is shielded from containment by more than 2 feet of concrete in all directions. The containment walls are 3'9" thick as well. Accordingly, the control room is adequately shielded from containment shine, as well as shine from containment leakage sources.

With respect to shine from the release plume, the exterior Auxiliary Building surrounds the control room and the exterior concrete walls are approximately 21" thick. The floors, walls, and ceilings of the control room add to the concrete shielding from the plume. Therefore, shine from the release plume to the control room occupants will not be significant.

For the Fuel Handling Accident scenario where the Personnel Airlock is open, the Auxiliary Building area around the control room could become contaminated. A small section of the control room envelope wall is only 1 foot thick inside the Auxiliary Building (between the control room and an interior hallway). Doses to the control room operators due to shine from the contaminated area through the 1 foot thick wall are included in the Fuel Handling Accident evaluation of control room doses and were found to be not significant.

SNC considered the potential impact of the shine sources mentioned above to the control room operator doses. However, due to the shielding, and the distance from the source, SNC made the determination that the shine would not contribute significantly to the dose. The control room at FNP is located within the auxiliary building of the plant. It is not adjacent to the containment, which consists of a concrete wall three feet nine inches thick. The floors and ceiling of the control room consist of two feet of concrete, and the exterior building walls are twenty one inches thick. Radiation releases in the auxiliary building during a LOCA are assumed to occur in the penetration room filtration system envelope, which is well away from the control room and does not communicate with it. The penetration room walls are two feet thick. Therefore, the radiation shine from the containment, from any releases in the auxiliary building, and from the radioactive plume is an insignificant contributor to operator dose.

The CREFS intake and recirculation filters are located in rooms above the control room and are separated by at least two feet of concrete from the occupied spaces in the control room. The radioactive sources are not in the control room envelope. As such, there is no significant contribution from these sources to operator dose.

3.2.5.3 Control Room Operator Dose during Ingress and Egress

Regulations in 10 CFR Section 50.67(b)(2) requires that the licensee's analyses demonstrate with reasonable assurance that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

The licensee evaluated the dose received by the control room operators during routine access to the control room for the 30 day period following the LOCA. The licensee assumed the following in their calculation:

- The operator travels to the site, parks in the parking lot, and walks from the parking lot to the auxiliary building entrance without any protection.
- The operator leaves using the same path.
- Meteorological data is the same as used for the LOCA dose consequences analysis previously discussed in the submittal.
- Containment leakage, quantity and timing, is the same as modeled in the LOCA dose consequences analysis.
- ESF Leakage outside containment, quantity and timing, is the same as modeled in the LOCA dose consequences analysis (40,000 cc/hr is modeled in the operator transit dose analysis).
- RWST leakage, quantity and timing, is the same as calculated in the LOCA dose consequences calculation. This iodine isotope leakage comes from the 2 gpm back leakage from RHR to the RWST modeled in the analysis.
- Containment elevation is 290 feet 3 inches (ground elevation is 154 feet 6 inches). Containment is 65 feet in diameter, and the wall is three feet nine inches thick concrete with a three inch steel plate liner.
- Walking distance from the parking lot to the entrance to the control building is conservatively assumed to be 1,501.5 feet.
- The operators will be confined to the control room for the first 24 hours (100 percent occupancy factor) following the accident. The first ingress/egress will begin 12 hours thereafter and be repeated every 12 hours for the length of the accident (30 days) (59 trips total).
- Operator walking speed is 3 miles per hour (mph).
- Operator respiration rate is twice the normal respiration rate for the control room (7.0E-04 cubic meters per second (m^3/sec)).
- For the ground shine dose, 100 percent of the non-noble gas released activity is deposited on the ground of the plant site.
- The dose point for the operator walking along the ingress/egress path is 5 feet off the ground.
- Shielding provided by intervening equipment, plant buildings, or other materials is ignored (except for the containment concrete and liner when determining what the containment shine does to the operator).
- The five points along the ingress/egress path below are used to determine dose rates.

Transit Segment	Starting Location	End Location	Segment Distance
1	Parking Lot	Dose Point 1	326.5 feet
2	Dose Point 1	Dose Point 2	106.1 feet
3	Dose Point 2	Dose Point 3	220.6 feet
4	Dose Point 3	Dose Point 4	436.4 feet
5	Dose Point 4	Dose Point 5	411.6 feet

The licensee determined the dose to the operator from ingress and egress is 0.2 rem. Contributions to the dose occur from containment leakage, ESF system leakage, RWST back leakage, ground shine from deposited radioactivity, and containment shine. The licensee used conservative assumptions to evaluate the control room operator dose during ingress and egress, and therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.2.6 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE and the assumptions found acceptable to the NRC staff are presented in Table 2. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and control room radiological doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and are, therefore, acceptable.

3.3 Fuel Handling Accident (FHA)

The FHA involves the drop of a fuel assembly during refueling operations. The mechanical part of the licensee's analysis remains unchanged from the current licensing basis and it assumes that the total number of failed fuel rods is 264, which is one fuel assembly out of the 157 fuel assemblies in the core. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3.7.13 and TS 3.9.6. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed AST amendment takes credit for the normal decay of irradiated fuel.

The analysis was performed assuming a decay period of 100 hours after shutdown. Two cases are analyzed for the FHA: (1) a FHA in containment and (2) a FHA in the spent fuel pool (SFP) area of the auxiliary building. A FHA in the SFP area of the auxiliary building would involve a release via the plant vent stack. A FHA in the containment would involve two release paths, one through the open equipment hatch directly to the environment, and one through the open personnel airlock (PAL) to the auxiliary building and then to the environment via the plant vent stack.

3.3.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. The licensee performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or SFP depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumes: (1) that the chemical form of radioiodine released from the fuel to the SFP consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, (2) the CsI released from the fuel completely dissociates in the pool water, and (3) because of the low pH of the pool water, the CsI re-evolves and releases elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The

licensee assumes that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

As corrected by item 8 of Regulatory Issue Summary 2006-04 (Reference 34), RG 1.183, Appendix B, Regulatory Position 2, should read as follows:

> If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species.

In accordance with RG 1.183, Appendix B, Regulatory Position 2, the licensee credits an overall iodine DF of 200 for a water cover depth of 23 feet. Consistent with RG 1.183, the licensee credits an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity and did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

The fraction of the core that is damaged is assumed to be one fuel assembly out of 157 fuel assemblies in the core, which consists of 264 fuel rods. A peaking factor of 1.7 is applied to the fission product inventory of the damaged rods.

The licensee analyzed the FHA based on the non-LOCA fractions of fission product inventory in the fuel rod gap presented in Table 3 of RG 1.183. The activity release fractions are 8 percent of the core I-131 inventory, 10 percent of the Kr-85 inventory, 5 percent of the remaining noble gas, 5 percent of the remaining halogen isotopes, and 12 percent of the core alkali metals. The licensee stated that the core does not contain any mixed oxide fuel, that each rod does not exceed a peak burnup of 62 GWD/MTU, and that for each rod burnup that exceeds 54 GWD/MTU that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot peak rod average power. The licensee used conservative assumptions to evaluate the FHA source term, and therefore, the NRC staff finds the source term to be acceptable for the AST FHA analysis.

3.3.2 Transport

3.3.2.1 FHA in SFP area of Auxiliary building

Releases from the FHA in SFP are via the plant vent stack with credit taken for filtration of the iodine from the penetration room filtration (PRF) system. Prior to the FHA in the SFP, the SFP air supply and exhaust systems maintain a slightly negative pressure above the SFP. The PRF SFP isolation dampers are normally open to the SFP area to provide an open path for air flow to the PRF system. There is a pair of radiation detectors in the SFP exhaust ductwork. On a high-high radiation alarm, the SFP air supply and exhaust dampers close and the PRF exhaust fans start. The air flow rate from the SFP area is limited by the PRF SFP isolation dampers. The normal ventilation system will be isolated automatically and the activity will be exhausted through the PRF system. Both of the 100 percent capacity PRF systems will receive an automatic start signal. This ensures the fuel handling accident in the SFP release is through the PRF. With normal air supply isolated, the PRF maintains the slightly negative pressure. The PRF system charcoal filter efficiency for elemental and organic iodine is 89.5 percent and the

HEPA filter efficiency for particulates is 89.5 percent. Consistent with RG 1.183 the FHA in SFP is released over a two-hour period.

3.3.2.2 FHA in Containment

Releases from the containment are through the containment equipment hatch and the PAL and no credit is taken to close these pathways. The release through the open equipment hatch is directly to the environment. The release through the open PAL credits mixing in a portion of the auxiliary building on the same level as the control room. The mixing in the auxiliary building volume is assumed to be instantaneous. However, the auxiliary building ventilation system does not meet regulatory position 5.1.2 in RG 1.183. Therefore, the NRC staff requested that the licensee demonstrate how mixing occurs without the auxiliary building ventilation system. In SNC letter dated May 23, 2017 the licensee stated:

> Without the operation of the Containment and Auxiliary Building ventilation systems, there would be no significant release of radioactivity following a fuel handling accident (FHA) inside Containment. The assumption of their operation is not to mitigate the event, but to provide a means of releasing the radioactivity to the environment. No credit is taken for their filtration capabilities. Mixing is assumed to occur instantaneously in the Auxiliary Building. No credit is taken for isotopic decay during mixing.
>
> The Auxiliary Building volume between the open personnel airlock (PAL) and the Main Control Room (MCR) entrance was not modeled to mitigate the effects of the FHA inside Containment. It was modeled to maximize the dose contribution from the 10 CFM unfiltered inleakage due to MCR ingress/egress.
>
> ...the Containment Operating Deck and PAL are at the same elevation as the MCR (EL 155'-0"). In the event of a FHA in Containment with the PAL open, the volume of the Auxiliary Building between the PAL and the MCR may contain unfiltered radionuclides.
>
> The assumed Auxiliary Building volume at this elevation was selected to provide the minimum volume between the open PAL and the MCR entrance. This provided a path for the radionuclides to the MCR, while minimizing dilution in the Auxiliary Building. The flow rate through the open PAL and this Auxiliary Building volume was selected, based on sensitivity studies, to maximize the MCR dose contribution from the 10 CFM ingress/egress unfiltered inleakage. It is conservative to assume holdup in the Auxiliary Building for this MCR dose contribution.

The NRC staff reviewed the licensee's response above and agrees that the above modeling will maximize the dose contribution to the control room. The NRC staff also requested that the licensee provide an evaluation that analyzes the three different configurations allowed by TS 3.9.3, shows consistency with RG 1.1.83, and meets the regulatory limits. The three configurations analyzed during the FHA in containment are:

- Open containment equipment hatch and open personnel airlock;
- Closed containment equipment hatch and open personnel airlock; and
- Open containment equipment hatch and closed personnel airlock.

Because of the inclusion of the two additional cases the licensee revised Enclosure 1, Table 3.6a and Enclosure 7 of the LAR submittal. For this analysis, if the containment equipment hatch or the personnel airlock is presumed to be open, the licensee took no credit to close it. The offsite doses calculated for the open containment equipment hatch and open personnel airlock configuration bound the other two possible configurations. The open personnel airlock could allow areas around the control room to become contaminated; therefore, the licensee's calculation accounts for dose impacts of ingress/egress through the control room doors. A small amount of the control room envelope wall internal to the auxiliary building is one foot thick; therefore, the licensee added the shine from the contaminated area through the wall to the control room operator dose. Consistent with RG 1.183 the FHA in containment is released over a two-hour period.

3.3.3 Control Room Habitability for the FHA

The licensee evaluated control room habitability for the FHA assuming that the activity is released directly to the CREFS normal intake from the plant vent using the plant vent to control room atmospheric dispersion factors for the release in the SFP and from the containment equipment hatch using the equipment hatch to control room atmospheric dispersion factors for the release in containment.

In the FHA analysis the licensee takes credit for the control room normal intake radiation monitors to isolate the control room normal unfiltered flow into the control room. The FHA does not cause an SI or CIAS signal. Radioactive material from the accident will reach the radiation monitors at the control room intakes, the radiation monitor signal will initiate the automatic isolation of the control room, and then the operators will manually initiate the emergency pressurization mode of CREFS operation. The FHA analysis evaluated the time from the event to the closure of the isolation dampers (from either FHA location) and found the time to be substantially less than 1 minute. The analysis conservatively assumes that the radiation monitor initiates isolation mode 1 minute after the event for either FHA location, and the operators take twenty minutes from that time to initiate the pressurization mode.

The CREFS is designed to maintain the control room envelope at a positive pressure relative to the surrounding area, following a postulated FHA. On detection of high radiation in the normal outside air intakes, the normal outside air intakes for the control room are automatically isolated and the normal flow rate of 2,340 cfm lowers to 600 cfm while CREFS is in isolation mode. Upon manual initiation of CREFS the pressurization flow will be 375 cfm and is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, during CREFS operation, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2,700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 94.5 percent.

3.3.4 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the radiological dose guidelines provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in

Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE, and the assumptions found acceptable to the NRC staff are presented in Table 3. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.4 Main Steam Line Break (MSLB) Accident

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes the main steam isolation valves (MSIVs) to close and, if the plant is operating at power when the event is initiated, causes a reactor scram. For the MSLB DBA radiological consequence analysis, a loss-of-offsite power occurs coincident with the reactor trip. Following a reactor trip and turbine trip, the radioactivity is released to the environment from the break point on the faulted steam generator. Because the loss-of-offsite power renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

The radiological consequences of a MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The affected steam generator (SG), hereafter referred to as the faulted SG, rapidly depressurizes and releases its initial contents to the environment. The MSLB accident is described in FNP UFSAR section 15.4.2. Guidance in RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a PWR MSLB.

As stated above, the steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly is stuck in its fully withdrawn position after the reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the SI system.

3.4.1 Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for a PWR MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TSs including the effects of pre-accident and concurrent iodine spiking. The licensee's evaluation indicates that no fuel damage would occur as a result of a MSLB accident.

Therefore, the licensee considered the two radioiodine spiking cases described in RG 1.183. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TSs for a spiking condition. For FNP, the maximum iodine concentration allowed by TS 3.4.16 as the result of an iodine spike is 30 micro curies per gram (μCi/gm) of dose equivalent I-131 (DEI).

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in FNP TSs. For FNP, the RCS TS 3.4.16 limit for equilibrium or normal operation is 0.5 µCi/gm DEI. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with RG 1.183.

For the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary-side activity using the equilibrium secondary-side specific activity found in FNP TS 3.7.16 of less than or equal to 0.1 µCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 20 percent of those in the reactor coolant system corresponding to 1 percent failed fuel. The feed water system flows into the SG are modeled as a source of radioiodine in this analysis. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic consistent with RG 1.183, Appendix E, Regulatory Position 4.

3.4.2 Release Transport

The licensee followed the guidance as described in RG 1.183, Appendix E, Regulatory Position 5, in all aspects of the transport analysis for the MSLB. For additional conservatism the licensee assumes a total primary-to-secondary leak rate equal to 1 gpm (1,440 gallons per day (gpd)), which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd through any one SG, which is a total of 450 gpd from all 3 steam generators. The licensee modeled the assumed primary-to-secondary leakage of 0.35 gpm into the faulted SG and 0.65 gpm into the two intact SGs.

Guidance in RG 1.183, Appendix E, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., [pounds of mass per hour] lbm/hr) should be consistent with the basis of the parameter being converted. The [alternate repair criteria] ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc [grams per cubic centimeter] (62.4 [pounds of mass per cubic foot] lbm/ft^3).

The licensee's assumes a leakage density of 62.4 lbm/ft^3. RG 1.183, Appendix E, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 [degrees Celsius] °C (212°[Fahrenheit] F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In accordance with RG 1.183, the licensee assumes that primary-to-secondary leakage continues until the RCS reaches 200°F, which is 24 hours after the MSLB, at which time

shutdown cooling is initiated. In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E Regulatory Positions 5.5.1, 5.5.2, 5.5.3, and 5.5.4 the licensee assumes that all of the primary-to-secondary leakage into the faulted steam generator will flow directly from the RCS to the environment with no partitioning. For the unaffected steam generators that are used for plant cooldown, the licensee assumes a partition factor of 100 is applied to the iodine nuclides. The iodine releases to the environment from the unaffected steam generators are assumed to be 97 percent elemental and 3 percent organic, which is consistent with Regulatory Position 4 in RG 1.183, Appendix E.

The total release from the faulted steam generator is 4.83E+05 lbm initially plus 0.35 gpm from the primary-to-secondary leakage for 24 hours. Twenty four hours after the accident, no further steam containing radionuclides is released from the faulted SG to the environment.

3.4.3 Control Room Habitability for the MSLB

The licensee evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the pressurization mode of operation after the initiation of CIAS signal generated in response to an SI signal. The licensee determined that the maximum time to generate the safety injection signal will conservatively be 27 seconds. The SI to CIAS signal generation time is less than one second, and pressurization occurs in less than a minute.

The CREFS is designed to maintain the control room envelope at a positive pressure relative to the surrounding area, following a postulated MSLB. Upon CREFS initiation the normal outside air intakes for the control room are automatically isolated. Pressurization flow will be 375 cfm and is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, during postulated accident conditions, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2,700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. The licensee assumes 310 cfm of unfiltered inleakage from the CREFS system, which includes 10 cfm for control room ingress and egress.

3.4.4 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE, and the assumptions found acceptable to the NRC staff are presented in Table 4. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and control room doses estimated by the licensee for the MSLB were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.5 SGTR Accident

The SGTR accident assumes an instantaneous and complete severance of a single steam generator tube. The postulated break allows primary coolant to leak to the secondary side of the ruptured steam generator. The radioactivity from the leaking steam generator tube mixes with the shell-side water in the affected steam generator. For the SGTR DBA radiological consequence analysis, mass transfer from the primary to the secondary continues until the break flow is terminated. Activity is released to the environment from the faulted SG until operator action is taken to isolate it. Break flow and release isolation is assumed to occur in thirty minutes for the faulted SG consistent with the FNP current licensing basis. Leakage into the intact SGs continues with activity released to the environment through steaming until the reactor coolant system is cooled to cold shutdown conditions after 8 hours. The SGTR assumes a concurrent loss-of-offsite power to maximize the release to the environment.

3.5.1 Source Term

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for a SGTR accident. Guidance in RG 1.183, Appendix F, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TSs and that two cases of iodine spiking should be assumed. The licensee's evaluation indicates that no fuel damage would occur as a result of a SGTR accident. Therefore, consistent with RG 1.183, the licensee performed the SGTR accident analyses for two radioiodine spiking cases. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TSs for a spiking condition. For FNP, the maximum iodine concentration allowed by TS 3.4.16 as a result of an iodine spike is 30 μCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For FNP, the RCS Specific Activity TS 3.4.16 limit for normal operation is 0.5 μCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate. The iodine release rate at equilibrium is equal to the rate at which iodine is RCS coolant system leakage. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours. The licensee assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system.

In accordance with RG 1.183 Appendix F, Regulatory Position 4, the licensee assumes the speciation for iodine release from the steam generators is 97 percent elemental and 3 percent organic. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of less than or equal to 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 20 percent of those in the RCS corresponding to 1 percent failed fuel. Although a loss-of-offsite power is assumed the licensee modeled feed water system flows into the SG as a source of radioiodine in this analysis for conservatism.

3.5.2 Release Transport

The licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5, in all aspects of the transport analysis for the SGTR. For additional conservatism the licensee assumes a total primary-to-secondary leak rate equal to 1 gpm (1,440 gpd), which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd through any one SG. The licensee modeled the assumed primary-to-secondary leakage of 0.35 gpm into the faulted SG and 0.65 gpm into the two intact SGs.

Guidance in RG 1.183, Appendix F, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

The licensee's SGTR leak rate of 1 gpm corresponds to a leakage density of 62.4 lbm/ft^3 and is into the three steam generators. Guidance in RG 1.183, Appendix F, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In accordance with RG 1.183, the licensee assumes that the release of radioactivity from the ruptured SG continues for 30 minutes and the unaffected SGs continues for 8 hours at which time shutdown cooling is initiated, and steam releases from the SGs have been terminated. The licensee evaluated the dose consequences from discharges of steam from the intact SGs for a period of 8 hours, until the primary system has cooled sufficiently to allow an alignment to shutdown cooling. At this point in the accident sequence, steaming is no longer required for cool down and releases from the intact SGs are terminated.

The licensee assumes that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam based upon the thermodynamic conditions in the RCS relative to the secondary system. The licensee assumes that the flashed portion of the break flow will ascend through the bulk water in the SG, enter the steam space of the affected SG, and be immediately available for release to the environment with no credit taken for scrubbing. Although RG 1.183 allows the use of the methodologies described in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" to determine the amount of scrubbing credit applied to the flashed portion of the break flow, the licensee did not credit scrubbing of the activity in the flashed break flow in the ruptured SG.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the SGs to the environment without reduction or

mitigation. In the ruptured SG, the licensee assumes the iodine in the flashed portion of the break flow is immediately available for release without reduction or mitigation. The break and leakage flow that does not flash mixes uniformly with the SG liquid mass and activity is released to the environment in direct proportion to the steaming rate and the partition coefficient, in accordance with RG 1.183 Appendix F regulatory position 5.6. A SG partition coefficient for iodine of 100 is assumed.

3.5.3 Control Room Habitability for the SGTR

The licensee evaluated control room habitability for the SGTR assuming that the CREFS automatically transfers to the pressurization mode of operation upon a CIAS signal generated in response to an SI signal. The licensee determined that the maximum time to generate the SI signal will be 27 seconds. The SI to CIAS signal generation time is less than one second, and pressurization occurs in less than a minute.

The CREFS is designed to maintain the control room envelope at a positive pressure relative to the surrounding area, following a postulated SGTR. Upon CREFS initiation the normal outside air intakes for the control room are automatically isolated. Pressurization flow is 375 cfm and is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, during postulated accident conditions, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. The licensee assumes 310 cfm of unfiltered inleakage from the CREFS system, which includes 10 cfm for control room ingress and egress.

3.5.4 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR and concluded that the radiological consequences at the EAB, LPZ, and control room within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE, and the assumptions found acceptable to the NRC staff are presented in Table 5. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and control room doses estimated by the licensee for the SGTR were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.6 Control Rod Ejection Accident (CREA)

FNP updated final safety analysis report section 15.4.6 (ADAMS Accession No. ML17117A373) describes the CREA as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Following the applicable guidance, the licensee evaluated two separate release scenarios for the CREA. In the first case, the failed fuel resulting from the CREA is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released to the environment at the containment TS leak rate for the first 24 hours and at half that value for the remaining 29 days.

For the second case, the radiological consequences from a CREA is evaluated assuming that the RCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case, fission products from the damaged fuel are assumed to be released to the primary coolant and transported to the secondary system through primary-to-secondary leakage in the SGs.

3.6.1 Source Term

The source term for the CREA is assumed to result in fuel damage consisting of localized damage to fuel cladding with some fuel melt occurring in the damaged rods. The source term for the CREA is described in RG 1.183, Appendix H, Regulatory Position 1, which states that:

> Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

The licensee assumed that as a result of the CREA, 10 percent of the fuel experiences departure from nucleate boiling resulting in cladding damage. A peaking factor of 1.7 was applied to the fission product inventory of the damaged rods. Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap and will be available for release in both the containment and the secondary-side release scenarios. The licensee assumes 0.25 percent of fuel rods experience fuel melting for both scenarios.

In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100 percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the ruptured control rod drive mechanism housing; and, 100 percent of the released activity is assumed to be released instantaneously and completely dissolved in the primary coolant and available for release to the secondary containment in the secondary-side release scenario. The licensee assumed that 100 percent of the noble gases and 25 percent of the iodine isotopes within the melting rods are available for release from the containment pathway and 100 percent of the noble gases and 50 percent of the iodine isotopes within the melting rods are available for release from the reactor coolant system through the secondary system pathway. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of less than or equal to 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 20 percent of those in the reactor coolant system corresponding to 1 percent failed fuel. For the secondary

side release, licensee modeled feed water system flows into the SG as a source of radioiodine in this analysis for conservatism.

3.6.2 Transport from Containment

The licensee used the minimum containment free air volume to conservatively maximize the radioactive concentration in containment. The licensee assumes that the activity released to the containment through the rupture in the reactor vessel head mixes instantaneously throughout the containment with no credit assumed for removal of iodine or noble gas in the containment due to containment sprays or for natural deposition of elemental iodine. The licensee is taking credit for natural deposition of aerosols in containment and a removal rate of 2.74E-02 per hour is used. The licensee assumes that all containment leakage is at the TS limit of 0.15 percent per day for the first 24 hours and 0.075 percent per day thereafter. The licensee assumes that the iodine released to the containment from the fuel consists of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic per RG 1.183, Appendix H, Regulatory Position 4. Because the licensee stated in the LAR that containment sprays will not necessarily be activated during a CREA, no credit was taken for pH being controlled at values of 7 or greater. The NRC staff asked the licensee to provide the plant specific evaluation that determined that the chemical form of radioiodine released to the containment atmosphere of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide is conservative at FNP and that the iodine does not re-evolve. In SNC's response dated May 23, 2017, the licensee stated:

> The statement "no credit is taken for pH being controlled at values of 7 or greater" was made in error.
>
> During the CRE accident, pressure in containment is not expected to reach the containment spray actuation set point. However, the Farley containment sump houses baskets of trisodium phosphate (TSP). SNC assumes the event introducing water into the sump causes dissolution of TSP, thereby maintaining the pH of the sump water at 7 or greater. Given that the pH condition of RG 1.183 is met, it may be assumed that iodine in the containment atmosphere is 95% particulate, 4.85% elemental, and 0.15% organic.
>
> Furthermore, a decontamination factor (DF) of 50 is applied to the deposition rate to limit the fraction of particulates airborne in containment that are removed due to deposition.

The NRC staff agrees with the licensee that if pH is controlled at 7 or greater than consistent with RG 1.183 Appendix H regulatory position 4 the chemical form of radioiodine released to the containment atmosphere may be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

3.6.3 Transport from Secondary System

In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the reactor coolant system to the steam generators secondary side assuming a total primary-to-secondary leak rate equal to 1 gpm from all three steam generators, which is higher than the TS 3.4.13d total allowable leak rate of 150 gallons per day through any one steam generator, to account for any accident induced leakage. In accordance with FNP current licensing basis the licensee assumes that this leak rate persists for a period of

2,500 seconds and the secondary system mass release to the environment lasts for 98 seconds. However, this is not consistent with RG 1.183, Appendix H, Regulatory Position 7.1, which states that a leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the TSs should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated. The NRC staff requested the licensee explain for the CREA release from the secondary system accident analysis and accident response, how it is consistent with RG 1.183 Appendix H position 7.1. In SNC's response in letter dated May 23, 2017, RAI No. 27, the licensee stated:

> In the CREA analysis, it is conservatively assumed that all accident-induced leakage from primary to secondary system is directly to the environment. This leakage continues for 2500 sec, the time required for the pressures within the primary and secondary systems to equalize.
>
> Additionally, the pre-existing activity in the secondary system is also assumed to be released directly to the environment. As indicated on Page E10-3 of Enclosure 10, the total mass of the secondary system fluid released during the accident is 468,600 lbm. Based on an analysis, it has been determined that the minimum possible time required to release this mass through the main steam safety valves is 98 sec. In the dose analysis, the flow from the steam generators to the environment is set to a rate that effectively exhausts this mass in 98 sec.
>
> Hence, the analysis does reflect a condition where primary-to-secondary release continues until shutdown cooling is in operation and releases from the steam generators have been terminated, consistent with RG 1.183, Appendix H, regulatory position 7.1.

The NRC staff finds that because the assumption that the leak rate persists for a period of 2,500 seconds and the secondary system mass release to the environment lasts for 98 seconds does reflect that primary-to-secondary release continues until shutdown cooling is in operation and releases from the steam generators have been terminated, that it is consistent with RG 1.183, Appendix H, Regulatory Position 7.1, and is acceptable.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Position 5.5, the licensee assumes that all of the primary-to-secondary leakage in the steam generators mix with the secondary water without flashing. For iodine, because the SG tubes remain covered for the duration of the CREA, the partition coefficient of 100 was taken directly from RG 1.183. The retention of particulate radionuclides in the SG is limited by the moisture carryover from the SG, which is 0.1 percent at FNP. The licensee assumed for the secondary side release that the chemical form of iodine released would be the same as that from the containment 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic, this assumption is a deviation from the guidance in RG 1.183, Appendix H, Regulatory Position 5, which states that the iodine release from the SGs to the environment should be 97 percent elemental and 3 percent organic.

The NRC staff requested the licensee provide a plant specific evaluation that determined that the chemical form of radioiodine released from the SGs of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic iodide is conservative and to show that the iodine does not re-evolve. In SNC letter dated September 7, 2017, the licensee stated that they revised the CREA analysis to better address the RG 1.183 regulatory position and that the revised analysis

uses a speciation of 97 percent elemental iodine and 3 percent organic iodine for the secondary release pathway. The NRC staff finds the revised CREA analysis to be consistent with RG 1.183, Appendix H, Regulatory Position 5, and is acceptable.

3.6.4 Control Room Habitability for the CREA

The licensee evaluated control room habitability for the CRE assuming that the CREFS automatically transfers to the pressurization mode of operation upon a CIAS signal generated in response to an SI signal. The licensee determined that the maximum time to generate the SI signal will be 27 seconds. The SI to CIAS signal generation time is less than one second, and pressurization occurs in less than a minute.

The CREFS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated CREA. Upon CREFS initiation the normal outside air intakes for the control room are automatically isolated. Pressurization flow will be 375 cfm and is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, during postulated accident conditions, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2,700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. The licensee assumes 325 cfm of unfiltered inleakage from the CREFS system, which includes 10 cfm for control room ingress and egress.

3.6.5 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE, and the assumptions found acceptable to the NRC staff are presented in Table 6. The NRC staff performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and control room doses estimated by the licensee for the CREA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.7 Locked Rotor Accident (LRA)

The LRA considers the instantaneous seizure of a reactor coolant pump (RCP) rotor, which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow causes a reactor trip. The licensee's evaluation indicates that fuel cladding damage will occur as a result of this accident. Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage. Radioactivity is released to the outside atmosphere from the secondary coolant system via steaming until cold shutdown conditions are established in the RCS. Following reactor trip and based on a coincident assumption of loss-of-offsite power, the condenser is unavailable and reactor cooldown is achieved using steam releases from the SGs until initiation of shutdown cooling. For conservatism, the licensee assumes a total primary-to-secondary leak rate equal to 1 gpm

from all three SGs, which is higher than the TS total allowable leak rate of 150 gallons per day through any one SG to account for any accident induced leakage.

3.7.1 Source Term

The licensee assumed that the instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel clad damage. The radiological dose analysis for this event conservatively assumes 20 percent fuel clad damage with no fuel melt predicted. Therefore, the source term available for release is associated with this fraction of damaged fuel cladding and the fraction of core activity existing in the gap. A radial peaking factor of 1.7 was applied to the fission product inventory of the damaged rods. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the RCS. Consistent with RG 1.183 Table 3, the licensee analyzed the LRA based on the non-LOCA fraction of fission product inventory in the fuel rod gap of 8 percent of the core I-131 inventory, 10 percent of the Kr-85 inventory, 5 percent of the remaining noble gas, 5 percent of the remaining halogen isotopes, and 12 percent of the core alkali metals. The licensee asserts that at FNP the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. Following the guidance in RG 1.183, Appendix G, Regulatory Position 4, the licensee assumes that the chemical form of radioiodine released from the fuel to the reactor coolant consists of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide, and that the iodine releases from the SGs to the environment is 97 percent elemental iodine and 3 percent organic iodine.

In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 20 percent of those in the RCS corresponding to 1 percent failed fuel. The licensee assumes a loss-of-offsite power to maximize the release to the environment; however, continued feedwater flow is modeled into the SG as a source of radioiodine in this analysis for conservatism.

3.7.2 Release Transport

The activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. The licensee assumes a conservative value for the design-basis leak rate of 1 gpm from all three SGs. This equates to a total of 1,440 gpd, which is greater than the maximum allowable operational leakage of 150 gpd for any one SG imposed in TS 3.4.13d. A loss-of-offsite power is assumed to occur concurrently with the reactor trip, which results in releases to the environment associated with the secondary coolant steaming from the SGs.

Because of the release dynamic of the activity from the SGs, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, because the SG tubes remain covered for the duration of the LRA, the partition coefficient of 100 was taken directly from the suggested guidance. The retention of particulate radionuclides in the SG is limited by the moisture carryover from the SG, which is 0.1 percent at FNP. Because of their volatility, 100 percent of the noble gases are assumed to be released. The licensee assumes that the steaming release from SG, and primary-to-secondary coolant leakage end after 8 hours, at which time the shutdown cooling is initiated. The total release from the SG to the environment is 1.481E+06 lbm for 8 hours.

The licensee's analysis is consistent with Appendix G of RG 1.183, which identifies acceptable radiological analysis assumptions for an LRA. The licensee assumes that the control room isolates and enters the emergency ventilation mode at the onset of the LRA and that for conservatism, an assessment is being performed for a delayed manual CREFS initiation. However, the licensee did not discuss the timing associated with the control room isolation nor the manual isolation time. Therefore, the NRC staff requested that the licensee further explain the operation of the CREFS during LRA. In the May 23, 2017, RAI response No. 32 the licensee stated, in part:

> It is possible that a Locked Rotor Accident will cause an SI signal to be generated. As discussed in the response to RAI [request for additional information] no. 7, the maximum time for an SI signal to be generated is 27 seconds. The signal processing time from an SI signal to a CIAS signal is less than one second, and the pressurization mode would be initiated in less than one minute. The dose contribution of that one minute to enter the pressurization mode is insignificant.
>
> However, for the Locked Rotor Accident, it is possible that the accident will not automatically initiate an SI signal or a CIAS signal. Therefore, the AST LRA analysis assumes that the CR HVAC supplies a normal intake flow of 2340 cfm to the CR, unfiltered until manual action is taken to initiate pressurization mode. The analysis assumes a 20 minute time for operators to achieve pressurization. After pressurization is achieved the analysis assumes a filtered make-up flow of 375 CFM from the CR HVAC system. SNC updated the analysis to assume 325 CFM of unfiltered inleakage from the HVAC system, taken from the contaminated air adjacent to the emergency HVAC intakes. The unfiltered inleakage includes 10 CFM for CR ingress and egress. The analysis assumes this control room ventilation scheme for the course of the accident.
>
> With respect to operator performance times for the manual action to achieve CREFS pressurization mode for Plant Farley, testing has shown a maximum time of 9 minutes, 55 seconds to achieve the action.

Upon CREFS manual initiation the pressurization flow will be 375 cfm and is drawn from one of the two emergency intakes. The control room pressurization flow is routed through charcoal and HEPA filters. The control room pressurization charcoal filter efficiency for elemental and organic iodine is 98.5 percent and the HEPA filter efficiency for particulates is 98.5 percent. Additionally, part of the control room flow is recirculated and filtered through charcoal and HEPA filters at a flow rate of 2,700 cfm. The control room recirculation charcoal filter efficiency for elemental and organic iodine is 94.5 percent and the HEPA filter efficiency for particulates is 98.5 percent.

3.7.3 NRC Staff Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the radiological dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The licensee's calculated dose results are given in Section 4.0, Table 1 of this SE, and the assumptions found acceptable to the NRC staff are presented in Table 7. The NRC staff

performed independent confirmatory dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and control room doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.8 AST Human Factors Review

The NRC staff reviewed the application with respect to human factors. The NRC staff issued RAI No. 42 requesting the licensee to describe if FNP will be updating any emergency operating procedures and, if so, describe any operator training associated with those updates. In SNC's May 23, 2017, response, the licensee stated:

> SNC is not intending to update any human factor reviews relating to emergency operating procedures (EOPs) as part of this LAR. If, in the future, SNC does change EOPs, it will do so in accordance with regulatory requirements.

The NRC staff finds the licensee's response acceptable.

3.9 AST NRC Staff Conclusion

The NRC staff has reviewed the AST implementation proposed by the licensee for FNP. In performing this review, the NRC staff relied upon information placed on the docket by the licensee and NRC staff confirmatory calculations.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed AST. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, with differences discussed and accepted earlier in this SE. The NRC staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds with reasonable assurance that the licensee's estimates of the TEDE due to design basis accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The NRC staff finds reasonable assurance that FNP will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The NRC staff concludes that the proposed AST implementation is acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the FNP design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in FNP's design basis.

3.10 TSTF-448

The availability of this TS improvement was published in the Federal Register on January 17, 2007, (72 FR 2022), as part of the Consolidated Line Item Improvement Process. The SE that follows is reproduced and modified as applicable from the January 17, 2007, *Federal Register* Notice.

3.10.1 Introduction

On August 8, 2006, the commercial nuclear electrical power generation industry owners group Technical Specifications Task Force submitted a proposed change, TSTF-448, Revision 3, to the improved Standard Technical Specifications (STS) (NUREGs 1430-1434) (Reference 35, Reference 16, Reference 36, Reference 37, and Reference 38) on behalf of the industry. TSTF-448, Revision 3, is a proposal to establish more effective and appropriate action, surveillance, and administrative STS requirements related to ensuring the habitability of the CRE.

In Generic Letter 2003-01, licensees were alerted to findings at facilities that existing TS surveillance requirements for the Control Room Emergency Filtration/Pressurization System (CREFS) may not be adequate. Specifically, the results of ASTM E741tracer gas tests to measure CRE unfiltered inleakage at facilities indicated that the differential pressure surveillance is not a reliable method for demonstrating CRE boundary operability. Licensees were requested to address existing TSs as follows:

> Provide confirmation that your technical specifications verify the integrity [i.e., operability] of the CRE [boundary], and the assumed [unfiltered] inleakage rates of potentially contaminated air. If you currently have a differential pressure surveillance requirement to demonstrate CRE [boundary] integrity, provide the basis for your conclusion that it remains adequate to demonstrate CRE integrity in light of the ASTM E741 testing results. If you conclude that your differential pressure surveillance requirement is no longer adequate, provide a schedule for: (1) revising the surveillance requirement in your technical specification to Reference an acceptable surveillance methodology (e.g., ASTM E741), and (2) making any necessary modifications to your CRE [boundary] so that compliance with your new surveillance requirement can be demonstrated.
>
> If your facility does not currently have a technical specification surveillance requirement for your CRE integrity, explain how and at what frequency you confirm your CRE integrity and why this is adequate to demonstrate CRE integrity.

To promote standardization and to minimize the resources that would be needed to create and process plant-specific amendment applications in response to the concerns described in the generic letter, the industry and the NRC proposed revisions to CRE habitability system requirements contained in the STS, using the STS change traveler process. This effort culminated in Revision 3 to traveler TSTF-448, "Control Room Habitability," which the NRC staff approved on January 17, 2007.

Consistent with the traveler as incorporated into NUREG-1431, the licensee proposed revising action and surveillance requirements in Specification 3.7.10, "Control Room," and revising the

administrative controls program, Specification 5.5.18, "Control Room Envelope Habitability Program." The purpose of the changes is to ensure that CRE boundary operability is maintained and verified through effective surveillance and programmatic requirements, and that appropriate remedial actions are taken in the event of an inoperable CRE boundary.

3.10.2 Control Room and Control Room Envelope

NRC RG 1.196 uses the term "control room envelope (CRE)" in addition to the term "control room" and defines each term as follows:

> Control Room: The plant area, defined in the facility licensing basis, in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. It encompasses the instrumentation and controls necessary for a safe shutdown of the plant and typically includes the critical document Reference file, computer room (if used as an integral part of the emergency response plan), shift supervisor's office, operator wash room and kitchen, and other critical areas to which frequent personnel access or continuous occupancy may be necessary in the event of an accident.
>
> Control Room Envelope: The plant area, defined in the facility licensing basis that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.

NRC RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003 (Reference 39), also contains these definitions, but uses the term CRE to mean both. This is because the protected environment provided for operators varies with the nuclear power facility. At some facilities this environment is limited to the control room; at others, it is the CRE. In this SE, consistent with the proposed changes to the STS, the CRE will be used to designate both. For consistency, facilities should use the term CRE with an appropriate facility-specific definition derived from the above CRE definition.

3.10.3 CREFS

The CREFS provides a protected environment from which operators can control the unit, during airborne challenges from radioactivity, hazardous chemicals, and fire byproducts, such as fire suppression agents and smoke, during both normal and accident conditions. The CREFS is designed to maintain a habitable environment in the control room envelope for 30 days of continuous occupancy after a DBA without exceeding a 5 rem TEDE. The CREFS consists of two redundant trains, each capable of maintaining the habitability of the CRE. The CREFS is considered operable when the individual components necessary to limit operator exposure are operable in both trains. A CREFS train is considered operable when the associated:

- Fan is operable;
- HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

- Heater, demister, ductwork, valves, and dampers are operable, and air circulation can be maintained; and
- CRE boundary is operable (the single boundary supports both trains).

The CRE boundary is considered operable when the measured unfiltered air inleakage is less than or equal to the inleakage value assumed by the licensing basis analyses of DBA consequences to CRE occupants.

3.10.4 Adoption of TSTF-448, Revision 3, by FNP

Adoption of TSTF-448, Revision 3, will assure that the facility's TS LCO for the CREFS is met by demonstrating unfiltered leakage into the CRE is within limits; i.e., the operability of the CRE boundary. In support of this surveillance, which specifies a test interval (frequency) of 6 years, TSTF-448 also adds TS administrative controls to assure the habitability of the CRE between performances of the ASTM E741 test. In addition, adoption of TSTF-448 will establish clearly stated and reasonable required actions in the event CRE unfiltered inleakage is found to exceed the analysis assumption.

The changes made by TSTF-448 to the STS requirements for the CREFS and the CRE boundary conform to 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3). Their adoption will better assure that FNP's CRE will remain habitable during normal operation and DBA conditions.

3.10.5 Proposed Changes

The proposed amendments would strengthen CRE habitability TS requirements by changing TS 3.7.10, "Control Room," and revises the TS administrative controls program on CRE habitability. Accompanying the proposed TS changes are appropriate conforming technical changes to the TS Bases. The proposed revision to the TS Bases also includes editorial and administrative changes to reflect applicable changes to the corresponding STS Bases, which were made to improve clarity, conform to the latest information and References, correct factual errors, and achieve more consistency among the STS NUREGs. These changes are consistent with STS as revised by TSTF-448, Revision 3.

The NRC staff compared the proposed TS changes to the STS and the STS markups and evaluations in TSTF-448. The NRC staff also reviewed the proposed changes to the TS Bases for consistency with the STS Bases and the plant-specific design and licensing bases, although approval of the TS Bases is not a condition for accepting the proposed amendments. However, TS 5.5.14, "Technical Specifications (TS) Bases Control Program," provides assurance that the licensee has established and will maintain the adequacy of the Bases. The proposed Bases for TS 3.7.10 refer to specific guidance in NEI 99-03, "Control Room Habitability Assessment Guidance," Revision 0, dated June 2001 (Reference 40), which the NRC staff has formally endorsed, with exceptions, through RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated May 2003.

In addition, the proposed change would add the following conditions to the Operating Licenses of FNP, Unit 1 and Unit 2:

2.C.(7) Upon implementation of Amendment No. [216 for Unit 1, 213 for Unit 2] adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by

SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

3.10.6 Editorial Changes

The licensee proposed editorial changes to TS 3.7.10, "Control Room," to establish standard terminology, such as "control room envelope (CRE)" in place of "CRE," and "CREFS" in place of "Control Room," with the exception of the plant-specific name for the CREFS. The licensee also proposed to revise the CREFS LCO note from, "The CRE may be opened intermittently under administrative controls," to, "The control room envelope (CRE) boundary may be opened intermittently under administrative control." These changes improve the usability and quality of the presentation of the TSs, have no impact on safety, and therefore, are acceptable.

3.10.7 TS 3.7.10 "CREFS"

The licensee proposed to revise the action requirements of TS 3.7.10, "CREFS," to acknowledge that an inoperable CRE boundary, depending upon the location of the associated degradation, could cause just one, instead of both CREFS trains to be inoperable. This is accomplished by revising Condition A to exclude Condition B, and revising Condition B to address one or more CREFS trains, as follows:

- Condition A - One CREFS train inoperable for reasons other than Condition B.
- Condition B - One or more CREFS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.

This change clarifies how to apply the action requirements in the event just one CREFS train is unable to ensure CRE occupant safety within licensing basis limits because of an inoperable CRE boundary. It enhances the usability of Conditions A and B with a presentation that is more consistent with the intent of the existing requirements. This change is an administrative change because it neither reduces nor increases the existing action requirements, and therefore, is acceptable.

The licensee proposed to revise existing Required Action B.1, "Initiate mitigating actions," which has an immediate Completion Time, to Required Action B.1, to immediately initiate action to implement mitigating actions; replace existing Required Actions B.2.1, "Restore CRE to OPERABLE status," and B.2.2.1, "General Design Criteria (GDC) 19 met using mitigating actions in B.1," which have a 24-hour Completion Time, with Required Action B.2 to verify, within 24 hours, that in the event of a DBA, CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke; and revise existing Required Action B.2.2.2, "Restore CRE to operable status," which has a completion time of 30 days, to Required Action B.3, "Restore CRE boundary to OPERABLE status," within 90 days.

The 24-hour Completion Time of new Required Action B.2 is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions as directed by Required Action B.1. The 90-day Completion Time of revised Required Action B.3 is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. The 90-day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most anticipated problems with the CRE boundary. Therefore, proposed Action B is acceptable.

To distinguish new Condition B from the existing condition for one CREFS train inoperable, Condition A is revised to state, "One CREFS train inoperable for reasons other than Condition B." To distinguish new Condition B from the existing condition for two CREFS trains inoperable, Condition D is revised to state, "Two CREFS trains inoperable during MODE 1, 2, 3, or 4 for reasons other than Condition B." The changes to existing Conditions A and D are less restrictive because these Conditions will no longer apply in the event one or two CREFS trains are inoperable due to an inoperable CRE boundary during unit operation in Mode 1, 2, 3, or 4. This is acceptable because the new Action B establishes adequate remedial measures in this condition.

The proposed CRE inleakage measurement SR states, "Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program." The CRE Habitability Program TS, proposed TS 5.5.18, requires that the program include requirements for determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of RG 1.197. This guidance References ASTM E741 as an acceptable method for ascertaining the unfiltered leakage into the CRE. The licensee has proposed to follow this method. The NRC staff finds that the proposed alternative method satisfies the criteria of RG 1.197. Therefore, the proposed CRE inleakage measurement SR is acceptable.

The FNP TS 3.7.10 LCO states:

> Two Control Room Emergency Filtration/Pressurization System (CREFS) Trains and the Control Room Envelope (CRE) shall be Operable.

The licensee proposed to revise TS 3.7.10 LCO to state:

> Two CREFS trains shall be OPERABLE.

This change along with the revision to SR 3.7.10.4 and TS 5.5.18, "Control Room Envelope Habitability Program," ensures the requirements for the CRE remain in the TSs and are relocated from the LCO. This change has no impact on safety, and therefore, is acceptable.

3.10.8 TS 5.5.18, CRE Habitability Program

The proposed administrative controls program TS is consistent with the model program TS in TSTF-448, Revision 3. In combination with SR 3.7.10.4, this program is intended to ensure the operability of the CRE boundary, which as part of an operable CREFS will ensure that CRE habitability is maintained such that CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under DBA conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

A TS CRE Habitability Program acceptable to the NRC staff requires the program to contain the following elements:

Definitions of CRE and CRE boundary. This element is intended to ensure that these definitions accurately describe the plant areas that are within the CRE, and also the interfaces that form the CRE boundary, and are consistent with the general definitions discussed in Section 3.10.2 of this SE. Establishing what is meant by the CRE and the CRE boundary will preclude ambiguity in the implementation of the program.

Configuration control and preventive maintenance of the CRE boundary. This element is intended to ensure the CRE boundary is maintained in its design condition. Guidance for implementing this element is contained in RG 1.196, which endorsed, with exceptions, NEI 99-03. Maintaining the CRE boundary in its design condition provides assurance that its leak-tightness will not significantly degrade between CRE inleakage determinations.

Assessment of CRE habitability at the frequencies stated in Sections C.1 and C.2 of RG 1.197 and measurement of unfiltered air leakage into the CRE in accordance with the testing methods and at the frequencies stated in Sections C.1 and C.2 of RG 1.197. This element is intended to ensure that the plant assesses CRE habitability consistent with Sections C.1 and C.2 of RG 1.197. Assessing CRE habitability at the NRC accepted frequencies provides assurance that significant degradation of the CRE boundary will not go undetected between CRE inleakage determinations. Determination of CRE inleakage using test methods acceptable to the NRC staff assures that test results are reliable for ascertaining CRE boundary operability. Determination of CRE inleakage at the NRC accepted frequencies provides assurance that significant degradation of the CRE boundary will not occur between CRE inleakage determinations.

Measurement of CRE pressure with respect to all areas adjacent to the CRE boundary at designated locations for use in assessing the CRE boundary is performed at a frequency of 24 months on a staggered test basis (with respect to the CREFS trains). This element is intended to ensure that CRE differential pressure is regularly measured to identify changes in pressure warranting evaluation of the condition of the CRE boundary. Obtaining and trending pressure data provides additional assurance that significant degradation of the CRE boundary will not go undetected between CRE inleakage determinations.

Quantitative limits on unfiltered inleakage. This element is intended to establish the CRE inleakage limit as the CRE unfiltered infiltration rate assumed in the CRE occupant radiological consequence analyses of design basis accidents. Having an unambiguous criterion for the CRE boundary to be considered operable in order to meet LCO 3.7.10, will ensure that associated action requirements will be consistently applied in the event of CRE degradation resulting in inleakage exceeding the limit.

Consistent with TSTF-448, Revision 3, the program states that the provisions of SR 3.0.2 are applicable to the program frequencies for performing the activities required by program paragraph number c, parts (i) and (ii) (assessment of CRE habitability and measurement of CRE inleakage), and paragraph number d (measurement of CRE differential pressure). This statement is needed to avoid confusion. SR 3.0.2 is applicable to the surveillance that References the testing in the CRE Habitability Program. However, SR 3.0.2 is not applicable to Administrative Controls unless specifically invoked. Providing this statement in the program eliminates any confusion regarding whether SR 3.0.2 is applicable, and is acceptable.

Consistent with TSTF-448, Revision 3, proposed TS 5.5.18 states that (1) a CRE Habitability Program shall be established and implemented, (2) the program shall include all of the NRC-staff required elements, as described above, and (3) the provisions of SR 3.0.2 shall apply to program frequencies. Therefore, TS 5.5.18, which is consistent with the model program TS approved by the NRC staff in TSTF-448, Revision 3, is acceptable.

3.10.9 <u>NRC Staff Conclusion</u>

Based on the above the NRC staff finds the licensee's proposed changes to the TSs and the Facility Operating Licenses are consistent with TSTF-448, Revision 3, and are, therefore, acceptable.

3.11 <u>TSTF-312</u>

3.11.1 <u>Background</u>

TS 3.9.3, "Containment Penetrations," requires, during core alterations and during movement of irradiated fuel assemblies within containment, that:

> The containment penetrations shall be in the following status:
>
> a. The equipment hatch is capable of being closed and held in place by four bolts;
> b. One door in each air lock is capable of being closed; and
> c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, bling flange, or equivalent, or
2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

Current TS LCO 3.9.3.c does not permit opening certain containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere, during fuel movement inside containment. The proposed change would add a note to LCO allowing, "Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative control."

Approval of applications of this nature are acceptable to the NRC staff if (1) dose consequences indicate acceptable radiological consequences without credit for isolation of the containment and each individual penetration flow path, and (2) the licensee has committed to implement administrative controls that ensure that the open personnel airlock and any open penetrations can and will be promptly closed, following containment evacuation, in the event of an FHA.

3.11.2 NRC Staff Evaluation

The FHA analysis is discussed in detail in Section 3.3 above. The FHA analysis considered a dropped fuel assembly inside the containment with the equipment hatch and personnel airlock open and a dropped fuel assembly in the spent fuel pool. The results of the analysis show that the radiological consequences resulting from the postulated FHA at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67.

The licensee will implement administrative controls that ensure that the containment will be evacuated and the open personnel airlock will be closed within 30 minutes of detection of a FHA. Although SNC has committed to establish administrative controls to ensure awareness of open containment penetration flow paths, and personnel will be designated and readily available to isolate any open containment penetration flow path, they did not specifically commit to isolate the containment penetration flow paths immediately upon a detection of a FHA. Therefore, the NRC staff requested that the licensee explain why they did not commit to isolate any open containment penetration flow paths immediately upon a detection of a FHA. In SNC's letter dated May 23, 2017, response to RAI No. 17, SNC stated:

> Item #2, above, [Regulatory Commitment: Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.] is the provision for the isolation of the flow path, which is consistent with TSTF-312.
>
> Under SNC's LLRT [local leakage rate test] procedures, personnel are stationed at the containment penetration being tested. Therefore, if an FHA were to occur at the same time an LLRT is being conducted, the LLRT personnel would be immediately available to isolate the penetration.
>
> The proposed change also includes the addition of text to the LCO discussion in Bases 3.9.3 stipulating that the administrative controls that are put in place when penetrations flow path(s) are unisolated ensure that: 1) appropriate personnel are aware of the open status of the penetration flow path during core alterations

> or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of an FHA.

In the licensee's application, it states that individuals are "designated and readily available to isolate," in the event of an FHA. TSTF-312 states that the open penetration "can and will be promptly closed." Therefore, the NRC staff requested a supplement RAI No. 17 requesting that the licensee clarify if the open penetration flow path(s) will be promptly closed in the event of an FHA. In SNC's letter dated September 7, 2017, the licensee stated:

> The intention of the LAR submittal and previous RAI response has been to fully comply with the requirement outlined in TSTF-312. Specifically, SNC is committing to implement acceptable administrative procedures that ensure in the event of a refueling accident that the open airlock can and will be promptly closed following containment evacuation and that the open penetration(s) can and will be promptly closed.

Based on the licensee's response above, the NRC staff finds the licensee will implement administrative controls that ensure that the open personnel airlock and any open penetrations can and will be promptly closed, following containment evacuation, in the event of an FHA.

3.11.3 NRC Staff Conclusion

The NRC staff reviewed the FHA analysis as described in Section 3.3 of this SE used by the licensee to assess the radiological impacts of adding a Note to the TS LCO 3.9.3, allowing penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control. The staff concludes that (1) dose consequences indicate acceptable radiological consequences without credit for isolation of the containment and each individual penetration flow path, and (2) the licensee will implement administrative controls that ensure that the open personnel airlock and any open penetrations can and will be promptly closed, following containment evacuation, in the event of an FHA.

Based on the above the NRC staff finds the licensee's proposed Note is consistent with TSTF-312, Revision 1, and is, therefore, acceptable.

3.12 Meteorology Staff's Evaluation

The AST LAR uses atmospheric dispersion factors (χ/Q values) for the EAB, the outer boundary of the LPZ, and the control room receptors in the calculations made for the radiological consequences assessments. NRC staff reviewed the licensee's atmospheric dispersion analyses as described below.

3.12.1 Meteorological Data

The licensee provided two sets of meteorological data for use in its atmospheric dispersion analyses. The first was a continuous four year period of hourly averaged data measured at the FNP meteorological tower from January 1, 2000, through December 31, 2003. The second data set included the same 2000 through 2003 data, but also contained an additional six months of derived data from 1999, resulting in a collection span of 4.5 years. This second data set was created by the licensee in response to previous NRC staff concerns that the 2000 through 2003 dataset contained an apparent lack of southerly winds. The licensee stated that the four year

data set generally provided more conservative results. Both sets of data were previously reviewed by the NRC and approved for use in atmospheric dispersion calculations in License Amendment Nos. 165 and 157 (Reference 41). The data were provided in electronic form and formatted for input into the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes" (Reference 42). The NRC staff found the meteorological data suitable for use in making calculations for the Atmospheric Dispersion Factors used to support this LAR.

3.12.2 Control Room Atmospheric Dispersion Factors

In order to assess the control room post-accident atmospheric dispersion conditions, the licensee generated χ/Q values using the ARCON96 computer code and guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." Guidance in RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in design basis accident radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there were no unusual siting, building arrangements, release characterization, release-receptor configuration, meteorological regimes, or terrain conditions that precluded the use of this model in support of the current LAR. The licensee considered various release-receptor combinations for the onsite control room atmospheric dispersion factors in order to determine the limiting release-receptor combination for the events. The existing χ/Q values of record were used for the containment hatch, reactor, and plant vent release points (Reference 41 and Reference 30). New χ/Q values were developed for the containment mini-purge and RWST release points for the LOCA based on the 4-year data set.

The NRC staff reviewed the licensee's assessments of control room post-accident dispersion conditions for the containment mini-purge and RWST release points, which were generated from the licensee's 4-year meteorological data and atmospheric dispersion modeling. This included a review of the inputs and assumptions, which the NRC staff found generally consistent with site configuration drawings and the NRC staff practice. In addition, the NRC staff generated sample comparative χ/Q value estimates for the containment mini-purge and RWST release points and found the resultant χ/Q values to be similar to those calculated by the licensee for the cases considered. Therefore, on the basis of this review, the NRC staff finds that the χ/Q values used by the licensee for the containment mini-purge and RWST release points are acceptable for use in making the control room radiological consequences assessments associated with this LAR.

3.12.3 Offsite Atmospheric Dispersion Factors

The licensee used the current licensing basis EAB and LPZ χ/Q values for the postulated design basis accident analyses. The process for calculating those values is described in Section 2.3.4.2.1 of Chapter 2 to the FNP, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR) (Reference 43). The values are also presented in Table 2.3-12 of the UFSAR. Per UFSAR Sections 2.1.2 and 2.1.3, the EAB and LPZ are bounded by circles centered on the reactor containment centerlines. The new release pathway, the RWST, is in close proximity to the reactor containment buildings as compared to the distances to EAB and LPZ. Therefore, the current licensing basis EAB and LPZ χ/Q values are acceptable for modeling releases from the RWST.

Section 5.3 of RG 1.183 states that χ/Q values for the EAB and LPZ approved by the staff during initial facility licensing (or in subsequent licensing proceedings) may be used in

performing the AST radiological analyses. Consequently, the NRC staff concluded that the current licensing basis EAB and LPZ χ/Q values are acceptable for use by the licensee in making the offsite radiological consequences assessments associated with this LAR.

3.12.4 NRC Staff Conclusion

The NRC staff reviewed the guidance, assumptions, and methodology used by the licensee to assess the χ/Q values associated with postulated releases from the potential release points. The staff found that the licensee used methods consistent with regulatory guidance identified in Section 2.0 of this SE. The licensee used meteorological data that complied with the guidance of RG 1.23. The inputs and assumptions used to calculate the control room χ/Q values were also consistent with the guidance of RG 1.194. The licensee used current licensing basis EAB and LPZ χ/Q values for the design basis dose analyses. Therefore, on the basis of this review of the atmospheric dispersion analysis, NRC staff finds that the licensee's proposed χ/Q values acceptable for use in calculating the radiological consequences assessments associated with this LAR.

4.0 TABLES

Table 1
Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem)

Accident	EAB[1]	LPZ[2]	SRP 15.0.1 and RG 1.183 Limit	Control Room	10 CFR 50.67 and GDC 19 Limit
Loss of Coolant Accident	13.2	6.0	25	4.9[3]	5
Fuel Handling Accident in Spent Fuel Pool	0.5	0.2	6.3	0.1	5
Fuel Handling Accident in Containment	2.4	0.9	6.3	2.3	5
Main Steam Line Break					
Pre-incident Iodine Spike	0.94	0.37	25	0.23	5
Accident initiated Iodine Spike	0.95	0.45	2.5	0.45	5
Steam Generator Tube Rupture					
Pre-incident Iodine Spike	2.4	0.92	25	0.48	5
Accident initiated Iodine Spike	0.82	0.34	2.5	0.17	5
Control Rod Ejection Accident					
Containment release	2.5	1.9	6.3	2.7	5
Secondary release	0.5	0.2	6.3	<0.1	5

Locked Rotor Accident Instantaneous CREFS initiation	1.2	0.83	2.5	0.36	5
Delayed CREFS initiation	1.2	0.83	2.5	0.39	5

(1) Exclusion Area Boundary
(2) Low Population Zone
(3) Dose due to occupancy is 4.7 rem and dose due to ingress/egress to control room is 0.2 rem

Table 2
Loss of Coolant Accident Assumptions

Parameter	Value
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Fuel release fractions	Per RG 1.183
Fuel release timing	Per RG 1.183
Margin Factors added to the core inventory	Kr-85: 1.15 Xe-133: 1.05 Cs-134: 1.35 Cs-136: 1.25 Cs-137: 1.20 Halogens: 1.03 Other Noble gases: 1.03 Other Isotopes: 1.03
Reactor Coolant System Mass	440,900 pounds mass
RCS concentration	Based on 1% failed fuel and 0.5 micro curies per gram (DEI) per TS 3.4.16
Containment Mini-Purge Parameters	
Minimum Containment Free Volume	2.03E+06 cubic feet
Chemical form of Iodine released	4.85% Elemental 95% Particulate 0.15% Organic
Containment Purge Filtration	none
Containment Purge Flow rate	2,850 cubic feet per minute
Containment Purge Isolation	30 seconds or less
Removal by wall deposition or containment sprays	none
Containment Leakage Parameters	
Containment volume	2.03E+06 cubic feet Sprayed: 1,668,660 cubic feet Unsprayed: 361,340 cubic feet
Chemical form of Iodine released	4.85% Elemental 95% Particulate 0.15% Organic
Containment spray removal coefficient	Element iodine: 13.7 per hour Aerosol: 5.45 per hour during injection mode and 5.13 per hour during recirculation mode Organic: None
Natural deposition of aerosols	0.1 per hour after sprays are terminated

Table 2 Loss of Coolant Accident Assumptions	
Parameter	**Value**
Containment spray	Initiation time: 90 seconds Termination time: 8 hours
Containment Spray Flow rate	2,480 gallons per minute in injection mode 2,290 gallons per minute in recirculation mode
Long term sump water pH	≥ 7.0
Maximum allowable DF for fission product removal	Elemental Iodine: 200
Containment Leak Rate	0 to 24 hours: 0.15% weight per day 1 to 30 days: 0.075% weight per day
Engineered Safety Features System Leakage Parameters	
Sump volume	49,200 cubic feet
Minimum time after LOCA when recirculation is initiated	20 minutes
Leakage duration	30 days
Maximum ECCS fluid temperature after initiation of recirculation	265°F
ECCS leak rate	40,000 cubic centimeters per hour
ECCS leakage iodine flashing fraction	10 percent
Chemical form of Iodine released	97% Elemental 3% Organic
Refueling Water Storage Tank (RWST) Back Leakage Parameters	
Minimum time after LOCA when recirculation is initiated	20 minutes
RWST volume at transfer to recirculation mode	29,002 gallons
RWST Capacity	505,562 gallons
RWST leakage inflow rate	2 gallons per minute
RWST leakage iodine flashing fraction	Per Tables in Supplement dated May 23, 2017
Control Room Emergency Ventilation	
Initiation time	Safety Injection signal generated: 27 seconds Pressurization: 1 minute

Table 3 FHA Assumptions	
Parameter	**Value**
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Number of damaged fuel assemblies	1
Total number of damaged fuel rods in the assembly	264
Number of fuel assemblies in core	157

Table 3
FHA Assumptions

Parameter	**Value**
Decay time prior to fuel movement	100 hours
Radial Peaking Factor	1.7
Fraction of Core Inventory in gap	I-131: 8% Kr-85: 10% Other Noble Gases: 5% Other Halides: 5% Alkali Metals: 12%
Margin Factors added to the core inventory	Kr-85: 1.15 Xe-133: 1.05 Other Noble gases: 1.03 Other Iodine Isotopes: 1.03
Iodine chemical form of gap release	99.85% Elemental 0.15% Organic
Iodine chemical forms released from pool	57% Elemental 43% Organic
Overlaying water depth	23 feet
Water Decontamination Factors	Iodine: 200 Noble Gas: 1 Particulates: infinite
Iodine chemical forms released from pool	Elemental 57% Organic 43%
Release rate to the environment	2 hours or less
FHA in Spent Fuel Pool	
FHA in spent fuel pool in auxiliary building release flow rate	Penetration room filtration: 5,000 cubic feet per minute
Fuel Handling volume	72,150 cubic feet
Penetration Room Filtration System filter efficiencies	Elemental and organic Iodine: 89.5% Aerosols: 89.5%
FHA in Containment	
Containment volume	2.03E+06 cubic feet
Mixing Volume in Containment	1.0E-06 cubic feet
Offsite Dose Model (open equipment hatch (EH) and open personnel airlock (PAL)) Containment Hatch Flow rate Personnel Airlock Flow Rate[1]	 55,000 cubic feet per minute 25,000 cubic feet per minute
Control Room Ingress/Egress Dose Model (open EH and open PAL) Containment Hatch Flow rate Auxiliary Building Mixing Volume Personnel Airlock Flow Rate[1]	 43,500 cubic feet per minute 100,650 cubic feet 12,300 cubic feet per minute
Control Room Containment Hatch Dose Model (open EH and open PAL) Containment Hatch Flow rate Personnel Airlock Flow Rate	 55,000 cubic feet per minute 12,300 cubic feet per minute

Table 3 FHA Assumptions	
Parameter	**Value**
Control Room Personnel Airlock Dose Model (open EH and open PAL) Containment Hatch Flow rate Personnel Airlock Flow Rate[1]	 43,500 cubic feet per minute 25,000 cubic feet per minute
Control Room and Offsite Dose Model (open EH and closed PAL) Containment Hatch Flow rate	 55,000 cubic feet per minute
Control Room Ingress/Egress Dose Model (closed EH and open PAL) Auxiliary Building Mixing Volume Personnel Airlock Flow Rate[1]	 100,650 cubic feet 12,300 cubic feet per minute
Control Room Personnel Airlock Dose Model (closed EH and open PAL) Personnel Airlock Flow Rate[1]	 25,000 cubic feet per minute
Control Room Emergency Ventilation	
Isolation time Pressurization time	Radiation monitor response time: 60 seconds Manual: 21 minutes

[1] Note about Personnel Airlock flow rates: For the three model elements where release to the environment should be conservatively maximized, the flow rate through the Personnel Airlock is assumed to be 25,000 cfm and passes directly to the environment through the Plant Vent Stack as drawn by the normal auxiliary building HVAC system. This flow rate assumption conservatively increases the release to the environment and back into the CR through the CREFS intakes. For the model elements where the ingress/egress through the CR doors is modeled, the flow out the airlock into the auxiliary building hall is 12,300 cfm, which is based on sensitivity studies and the capacity of the normal HVAC system. The studies show that this flow rate maximizes the auxiliary building radioactivity concentrations, and causes the highest dose from ingress/egress through the CR doors. The normal HVAC system enables the release to the auxiliary building from containment. The exhaust through the plant vent stack for this model element is 12,290 cfm, as 10 cfm goes into the CR through the doors.

Table 4 Main Steam Line Break Assumptions	
Parameter	**Value**
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Reactor coolant System (RCS) mass	440,900 pounds mass 2.00E+08 grams
RCS Volume	1.02E+04 cubic feet
RCS Leak rate to faulted SG	0.35 gallons per minute and leakage density 62.4 pounds mass per cubic feet
RCS Leak rate to intact SGs	0.65 gallons per minute
Failed fuel percentage	10%
RCS concentration	Based on 1% failed fuel and 0.5 micro curies per gram (DEI) per TS 3.4.16

Table 4 Main Steam Line Break Assumptions	
Parameter	**Value**
RCS Iodine Appearance Rates	I-131: 9.6 curies per hour I-132: 12 curies per hour I-133: 19 curies per hour I-134: 17 curies per hour I-135: 15 curies per hour
Pre-accident iodine spike concentration	30.0 micro curies per gram (DEI) per TS 3.4.16
Accident initiated iodine spike appearance rate	500 times equilibrium appearance rate
Duration of accident initiated iodine spike	8 hours
Secondary System Release Parameters	
Initial secondary coolant iodine concentration	0.1 micro curies per gram (DEI) per TS 3.7.16
Intact SG Mass	1.68E+05 pounds mass each, full
SG Volume	2,690 cubic feet each
Steam releases from intact SGs	0 to 2 hours: 3.48E+05 pounds mass 2 to 8 hours: 7.74E+05 pounds mass 8 to 24 hours: 1.04E+6 pounds mass
Feed water to Intact SG	0 to 2 hours: 4.81E+05 pounds mass 2 to 8 hours: 7.83E+05 pounds mass 8 to 24 hours: 1.04E+06 pounds mass
Faulted SG to Environment	0 to 24 hours: 4.83E+05 pounds mass
Alkali Metal concentration in secondary system	20% of those assumed in RCS corresponding to 1% failed fuel
Feed water Appearance Rate 0 to 2 hours	I-131: 8.3 curies per hour I-132: 3.0 curies per hour I-133: 13 curies per hour I-134: 2.0 curies per hour I-135: 7.3 curies per hour
Feed water Appearance Rate 2 to 8 hours	I-131: 4.5 curies per hour I-132: 1.6 curies per hour I-133: 7.2 curies per hour I-134: 1.1 curies per hour I-135: 3.9 curies per hour
Feed water Appearance Rate 8 to 24 hours	I-131: 2.2 curies per hour I-132: 0.8 curies per hour I-133: 3.6 curies per hour I-134: 0.54 curies per hour I-135: 2.0 curies per hour
Iodine Partition Coefficient in intact SGs	100
Alkali Metal concentration in secondary system	20% of those assumed in RCS corresponding to 1% failed fuel
Feed water concentration	Corresponding to TS 3.7.16
Termination of release: when cold shutdown is reached	24 hours

Table 4
Main Steam Line Break Assumptions

Parameter	Value
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated: 27 seconds Pressurization: 1 minute

Table 5
Steam Generator Tube Rupture Assumptions

Parameter	Value
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
RCS mass	440,900 pounds mass 2.00E+08 grams
RCS Volume	1.02E4 cubic feet
Time of reactor trip	324 seconds
Faulted SG isolated	30minutes
Break flow Terminated	30 minutes
Atmospheric Relief Valve release from faulted SG ended	30 minutes
RCS cooled to cold shutdown	8 hours
Tube leakage rate to SGs	Intact SG :0.65 gallons per minute Ruptured SG 0.35 gallons per minute leakage density 62.4 pounds mass per cubic feet
RCS iodine concentration	0.5 micro curies per gram (DEI) per TS 3.4.16
RCS equilibrium iodine appearance rates	I-131: 9.6 curies per hour I-132: 12 curies per hour I-133: 19 curies per hour I-134: 17 curies per hour I-135: 15 curies per hour
Pre-accident iodine spike concentration	30.0 micro curies per gram (DEI) per TS 3.4.16
Accident initiated iodine spike appearance rate	335 times equilibrium appearance rate
Duration of accident initiated iodine spike	8 hours
RCS to Environment	0 to 0.09 hours: 3.12E+04 pounds mass 0.09 to 0.5 hours: 1.28E+05 pounds mass
RCS flashing fraction	0 to 0.09 hours: 21% 0.09 to 0.5 hours: 15%
Iodine Partition Coefficient (flashed portion)	1.0 (released without hold up)
Iodine Partition Coefficient (non-flashed portion)	100
Secondary System Release Parameters	
Intact SG mass	1.05E+05 pounds mass each, full
SG Volume	1,685 cubic feet each

Table 5 Steam Generator Tube Rupture Assumptions	
Parameter	**Value**
Iodine species released to environment	97% Elemental 3% Organic
Initial secondary coolant iodine concentration TS 3.7.16	0.1 micro curies per gram (DEI)
Faulted SG to environment	0 to 0.09 hours: 3.67E+05 pounds mass 0.09 to 0.5 hours: 7.90E+04 pounds mass
Intact SG to environment	0 to 0.09 hours: 7.34E+05 pounds mass 0.09 to 2 hours: 4.22E+05 pounds mass 2 to 8 hours: 9.34E+05 pounds mass
Feed water to intact SG	0.09 to 2 hours: 3.27E+05 pounds mass 2 to 8 hours: 9.81E+05 pounds mass
Iodine Partition Coefficient	100
Alkali Metal concentration in secondary system	20% of those assumed in RCS corresponding to 1% failed fuel
Feed water Appearance Rate 0.09 to 2 hours	I-131: 5.9 curies per hour I-132: 2.1 curies per hour I-133: 9.4 curies per hour I-134: 1.4 curies per hour I-135: 5.2 curies per hour
Feed water Appearance Rate 2 to 8 hours	I-131: 5.6 curies per hour I-132: 2.0 curies per hour I-133: 9.0 curies per hour I-134: 1.3 curies per hour I-135: 4.9 curies per hour
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated: 27 seconds Pressurization: 1

Table 6 Control Rod Ejection Accident Assumptions	
Parameter	**Value**
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Failed fuel percentage	10%
Percentage of core inventory in fuel gap	10% noble gases and halogens 5% other halogens 12% alkali metals
Melted Fuel percentage	0.25%
RCS Mass	440,900 pounds mass 2.00E+08 grams
RCS Volume	1.02E+04 cubic feet
Radial Peaking Factor	1.7

Table 6
Control Rod Ejection Accident Assumptions

Parameter	**Value**
Margin Factors added to the core inventory	Kr-85: 1.15 Xe-133: 1.05 Cs-134: 1.35 Cs-136: 1.25 Cs-137: 1.20 Iodine isotopes: 1.02 Other Noble gases: 1.02 Other Isotopes: 1.03
RCS concentration	Based on 1% failed fuel
Containment Leakage Pathway	
Containment volume	2.03E+06 cubic feet
Containment leak rate	0 to 24 hours: 0.15% volume fraction per day 1 to 30 days: 0.075% volume fraction per day
Percentage of nobles gases and iodine available for containment release from melted fuel	Noble Gases: 100% Iodine: 25%
Chemical form of iodine in failed fuel	95% Particulate 4.85% Elemental 0.15% Organic
Natural Deposition Aerosol Removal Rate	2.74E-02 per hour
Termination of containment release	30 days
Secondary Side Pathway	
SG mass	1.68E+05 pounds mass per SG, full
SG Volume	2,693 cubic feet each
Primary to Secondary leak rate	1 gallon per minute (total for all SGs) and leakage density 62.4 pounds mass per cubic feet
Primary to Secondary leakage duration	2,500 seconds
Secondary System Mass release	468,600 pounds mass
Secondary System Mass release duration	98 seconds
Initial secondary coolant iodine concentration TS 3.7.16	0.1 micro curies per gram (DEI)
Percentage of nobles gases and iodine available for secondary release from melted fuel	Noble Gases: 100% Iodine: 50%
Iodine species released to environment	97% Elemental 3% Organic
Iodine Partition Coefficient in SGs	100
Alkali Metal Partition Coefficient	1000
Alkali Metal concentration in secondary system	20% of those assumed in RCS corresponding to 1% failed fuel

Table 6 Control Rod Ejection Accident Assumptions	
Parameter	**Value**
Feedwater Appearance Rate	I-131: 590 curies per hour I-132: 210 curies per hour I-133: 940 curies per hour I-134: 140 curies per hour I-135: 520 curies per hour
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated: 27 seconds Pressurization: 1 minute

Table 7 Locked Rotor Accident Assumptions	
Parameter	**Value**
Core Power Level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Reactor Coolant System Mass	440,900 pounds mass 2.00E+08 grams
Reactor Coolant System Volume	1.02E+04 cubic feet
Primary to Secondary SG tube leakage	1 gallons per minute (total for all SGs) and leakage density 62.4 pounds mass per cubic feet
Failed fuel percentage	20%
Radial Peaking Factor	1.7
Fraction of Core Inventory in fuel gap	I-131: 8% Kr-85: 10% Other Noble Gases: 5% Other Halogens: 5% Alkali Metals: 12%
Margin Factors added to the core inventory	Kr-85: 1.15 Xe-133: 1.05 Cs-134: 1.35 Cs-136: 1.25 Cs-137: 1.20 Iodine isotopes: 1.02 Other Noble gases: 1.02 Other Isotopes: 1.03
RCS concentration	Based on 1% failed fuel
Iodine form of gap release	95% Particulate 4.85% Elemental 0.15% Organic
Secondary Side Parameters	
SG mass	1.68E+05 pounds mass per SG, full
SG Volume	2,690 cubic feet each
Iodine species released to environment	97% Elemental 3% Organic

Table 7
Locked Rotor Accident Assumptions

Parameter	**Value**
SG to environment	0 to 2 hours: 5.64E+05 pounds mass 2 to 8 hours: 9.17E+05 pounds mass
Feed water to SG	0 to 2 hours: 7.63E+05 pounds mass 2 to 8 hours: 9.29E+05 pounds mass
Initial secondary coolant iodine concentration TS 3.7.16	0.1 micro curies per gram (DEI)
Alkali Metal concentration in secondary system	20% of those assumed in RCS corresponding to 1% failed fuel
Feed water Appearance Rate for 0 to 2 hours	I-131: 13 curies per hour I-132: 4.7 curies per hour I-133: 21 curies per hour I-134: 3.1 curies per hour I-135: 12 curies per hour
Feed water Appearance Rate for 2 to 8 hours	I-131: 5.3 curies per hour I-132: 1.9 curies per hour I-133: 8.5 curies per hour I-134: 1.3 curies per hour I-135: 4.7 curies per hour
Iodine Partition Coefficient	100
Alkali Metal Partition Coefficient	1000
Fraction of Noble Gas released	1.0 (released without hold up)
Termination of release from SGs	8 hours
Control Room Emergency Ventilation	
Automatic Initiation time Manual Initiation time	Safety Injection Signal generated: 27 seconds Manual: 20 minutes

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments on October 16, 2017. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on January 3, 2017 (82 FR 160). Accordingly, the amendments meet 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 amendments.

7.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS ADOPTING ALTERNATIVE SOURCE TERM, TSTF-448, REVISION 3, AND TSTF-312, REVISION 1 (CAC NOS. MF8861, MF8862, MF8916, MF8917, MF8918, AND MF8919; EPID NOS. L-2016-LLA-0017, L-2016-LLA-0018, AND L-2016-LLA-0019) DATED DECEMBER 20, 2017

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