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

April 14, 2011

Mr. Larry Meyer
Site Vice President
NextEra Energy
Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT (PBNP), UNITS 1 AND 2 - ISSUANCE OF LICENSE AMENDMENTS REGARDING USE OF ALTERNATE SOURCE TERM (TAC NOS. ME0219 AND ME0220)

Dear Mr. Meyer:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment Nos. 240 and 244 to Renewed Facility Operating License Nos. DPR-24 and DPR-27, for PBNP, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 8, 2008, as supplemented by letters dated January 16, January 27, February 20, April 17 (two letters), May 8, May 15, June 1, July 24, August 20, September 4 (two letters), September 10, October 2, November 20, November 25, and December 17 of 2009; January 14, February 4 (two letters), March 5, April 20, July 8, July 29, August 12, September 3, October 12, and November 16 of 2010; January 27, February 10, March 11, and April 6 of 2011.

The amendments modify the requirements of TS 3.4.16, "RCS [reactor coolant system] Specific Activity," and TS 3.7.13, "Secondary Specific Activity," as related to the use of an alternate source term (AST) associated with accident offsite and control room dose consequences.

Implementation of the AST supports adoption of the control room envelope habitability controls in accordance with NRC-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-448, Revision 3, "Control Room Habitability." To support this change, the amendment modifies the following: 1) TS 1.1, "Definitions"; 2) TS 3.7.9, "Control Room Emergency Filtration System (CREFS)," Limiting Condition for Operation 3.7.9, including Surveillance Requirements 3.7.9.2, 3.7.9.3 and 3.7.9.6; 3) TS 5.5.15, "Containment Leakage Rate Testing Program"; and 4) the addition of TS 5.5.18, "Control Room Envelope Habitability."

Finally, TS 5.6.4, Core Operating Limits Report (COLR) includes the addition of an approved analytical methodology as described in WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Analyses."

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

Sincerely,

/RA/ for

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 240 to DPR-24
2. Amendment No. 244 to DPR-27
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NOS. 50-266

POINT BEACH NUCLEAR PLANT, UNITS 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment Nos. 240
Renewed License No. DPR-24

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated December 8, 2008, as supplemented by letters dated January 16, January 27, February 20, April 17 (two letters), May 8, May 15, June 1, July 24, August 20, September 4 (two letters), September 10, October 2, November 20, November 25, and December 17 of 2009; January 14, February 4 (two letters), March 5, April 20, July 8, July 29, August 12, September 3, October 12, and November 16 of 2010; January 27, February 10, March 11, and April 6 of 2011, 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 Paragraph 4.B of Renewed Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 240, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

3. Accordingly, the license is amended by the following license conditions to be added to Appendix C, Additional Conditions, with wording as follows:

   I. Upon implementation of Amendment Nos. 240/244 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.6, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.1 8.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 3.7.9.6, in accordance with Specification 5.5.18.c.(i), shall be within 18 months of implementation of this amendment.

   b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within three (3) years of completion of the testing prescribed in item a. above.

   c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 18 months of implementation of this amendment.

   II. NextEra Energy Point Beach, LLC shall modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained.

   III. NextEra Energy Point Beach, LLC shall revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation.

   IV. NextEra Energy Point Beach, LLC shall modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications shall include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from an emergency diesel generator supplied source for the CREFS fans required for the new system alignment.

   V. NextEra Energy Point Beach, LLC shall modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components required to direct radioactive releases in the PAB to the vent stack shall be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. NextEra Energy Point Beach, LLC shall revise PBNP EOPs to address starting the VNPAB fans.

VI. NextEra Energy Point Beach, LLC shall perform Train B Emergency Diesel Generator load testing over a range of 2877 to 2950 kW at rated power factor. This license condition will remain in effect until implementation of LAR 261 for Unit 2.

VII. NextEra Energy Point Beach, LLC shall install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) shall be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.

VIII. NextEra Energy Point Beach, LLC shall procure the CREFS mitigating filtration unit with electrical power requirements equivalent to the CREFS filter fan motors (i.e., equivalent horse power, efficiency, power factor, and voltage requirements).

License condition 3.I, above, shall be implemented immediately upon implementation of this license amendment.

License conditions 3.II through 3.VIII, above, shall be implemented no later than the Unit 2 refueling outage in the spring of 2011.

4. The license amendment is effective as of its date of issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

R. Pascarelli

Robert J. Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-24,
Appendix C, and Technical Specifications

Date of Issuance: April 14, 2011

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed License No. DPR-27

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated December 8, 2008, as supplemented by letters dated January 16, January 27, February 20, April 17 (two letters), May 8, May 15, June 1, July 24, August 20, September 4 (two letters), September 10, October 2, November 20, November 25, and December 17 of 2009; January 14, February 4 (two letters), March 5, April 20, July 8, July 29, August 12, September 3, October 12, and November 16 of 2010; January 27, February 10, March 11, and April 6 of 2011, 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.

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 4.B of Renewed Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 244, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

3. Accordingly, the license is amended by the following license conditions to be added to Appendix C, Additional Conditions, with wording as follows:

   I. Upon implementation of Amendment Nos. 240/244 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.6, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.1 8.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 3.7.9.6, in accordance with Specification 5.5.18.c.(i), shall be within 18 months of implementation of this amendment.

      b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within three (3) years of completion of the testing prescribed in item a. above.

      c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 18 months of implementation of this amendment.

   II. NextEra Energy Point Beach, LLC shall modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained.

   III. NextEra Energy Point Beach, LLC shall revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation.

   IV. NextEra Energy Point Beach, LLC shall modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications shall include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from an emergency diesel generator supplied source for the CREFS fans required for the new system alignment.

   V. NextEra Energy Point Beach, LLC shall modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components required to direct radioactive releases in the PAB to the vent stack shall be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. NextEra Energy Point Beach, LLC shall revise PBNP EOPs to address starting the VNPAB fans.

VI. NextEra Energy Point Beach, LLC shall perform Train B Emergency Diesel Generator load testing over a range of 2877 to 2950 kW at rated power factor. This license condition will remain in effect until implementation of LAR 261 for Unit 2.

VII. NextEra Energy Point Beach, LLC shall install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) shall be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.

VIII. NextEra Energy Point Beach, LLC shall procure the CREFS mitigating filtration unit with electrical power requirements equivalent to the CREFS filter fan motors (i.e., equivalent horse power, efficiency, power factor, and voltage requirements).

License condition 3.I, above, shall be implemented immediately upon implementation of this license amendment.

License conditions 3.II through 3.VIII, above, shall be implemented no later than the Unit 2 refueling outage in the spring of 2011.

4. The license amendment is effective as of its date of issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-27,
Appendix C, and Technical Specifications

Date of Issuance: April 14, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 240

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 244

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of the Renewed Facility Operating License Nos. DPR-24 and DPR-27, Appendix C, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Operating License

REMOVE	INSERT
Unit 1 License Pages 3 and 6	Unit 1 License Pages 3 and 6
Unit 2 License Pages 3 and 6	Unit 2 License Pages 3 and 6

Appendix C

REMOVE	INSERT
---	Unit 1 Pages C-3 and C-4
---	Unit 2 Pages C-3 and C-4

Technical Specifications

REMOVE	INSERT
1.1-3	1.1-3
3.4.16-2	3.4.16-2
3.7.9-1	3.7.9-1
3.7.9-2	3.7.9-2
---	3.7.9-3
---	3.7.9-4
3.7.13-1	3.7.13-1
---	3.7.14-1
5.5-16	5.5-16
---	5.5-18
---	5.5-19
5.6-4	5.6-4

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

E. Pursuant to the Act and 10 CFR Parts 30 and 70, NextEra Energy Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

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

A. Maximum Power Levels

NextEra Energy Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 240, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

2. Operations to mitigate fuel damage considering the following:

    a. Protection and use of personnel assets
    b. Communications
    c. Minimizing fire spread
    d. Procedures for implementing integrated fire response strategy
    e. Identification of readily-available pre-staged equipment
    f. Training on integrated fire response strategy
    g. Spent fuel pool mitigation measures

3. Actions to minimize release to include consideration of:

    a. Water spray scrubbing
    b. Dose to onsite responders

M. <u>Additional Conditions</u>

The additional conditions contained in Appendix C, as revised through Amendment No. 240, are hereby incorporated into this license. NextEra Energy Point Beach shall operate the facility in accordance with the additional conditions.

5. The issuance of this renewed operating license is without prejudice to subsequent licensing action which may be taken by the Commission with regard to the ongoing rulemaking hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems (Docket No. RM 50-1).

6. This renewed operating license is effective as of the date of issuance, and shall expire at midnight on October 5, 2030.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

R. W. Borchardt, Deputy Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A - Technical Specifications
2. Appendix B - Environmental Technical Specifications
3. Appendix C - Additional Conditions

Date of Issuance: December 22, 2005

APPENDIX C
ADDITIONAL CONDITIONS
OPERATING LICENSE DPR-24

NextEra Energy Point Beach, LLC shall comply with the following conditions and the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
240	Upon implementation of Amendment Nos. 240/244 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.6, 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 3.7.9.6, in accordance with Specification 5.5.18.c.(i), shall be within 18 months of implementation of this amendment. b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within three (3) years of completion of the testing prescribed in item a. above. c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 18 months of implementation of this amendment.	Immediately
240	NextEra Energy Point Beach, LLC shall modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained.	No later than the Unit 2 (2011) refueling outage
240	NextEra Energy Point Beach, LLC shall revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation.	No later than the Unit 2 (2011) refueling outage
240	NextEra Energy Point Beach, LLC shall modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications shall include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from an emergency diesel generator supplied source for the CREFS fans required for the new system alignment.	No later than the Unit 2 (2011) refueling outage
240	NextEra Energy Point Beach, LLC shall modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components required to direct radioactive releases in the PAB to the vent stack shall be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. NextEra Energy Point Beach, LLC shall revise PBNP EOPs to address starting the VNPAB fans.	No later than the Unit 2 (2011) refueling outage

APPENDIX C
ADDITIONAL CONDITIONS
OPERATING LICENSE DPR-24

NextEra Energy Point Beach, LLC shall comply with the following conditions and the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
240	NextEra Energy Point Beach, LLC shall perform Train B Emergency Diesel Generator load testing over a range of 2877 to 2950 kW at rated power factor. This license condition will remain in effect until implementation of LAR 261 for Unit 2.	No later than the Unit 2 (2011) refueling outage
240	NextEra Energy Point Beach, LLC shall install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) shall be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute {EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.	No later than the Unit 2 (2011) refueling outage
240	NextEra Energy Point Beach, LLC shall procure mitigating filtration unit motors equivalent to W-14A/B (equivalent HP, efficiency, power factor, and voltage requirements).	No later than the Unit 2 (2011) refueling outage

C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NextEra Energy Point Beach to receive, possess and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed source for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

E. Pursuant to the Act and 10 CFR Parts 30 and 70, NextEra Energy Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

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

A. Maximum Power Levels

NextEra Energy Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 244, are hereby incorporated in the renewed operating license. NextEra Energy Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

e. Identification of readily-available pre-staged equipment
f. Training on integrated fire response strategy
g. Spent fuel pool mitigation measures

3. Actions to minimize release to include consideration of:

a. Water spray scrubbing
b. Dose to onsite responders

L. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 244, are hereby incorporated into this license. NextEra Energy Point Beach shall operate the facility in accordance with the additional conditions.

5. The issuance of this renewed operating license is without prejudice to subsequent licensing action which may be taken by the Commission with regard to the ongoing rulemaking hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems (Docket No. RM 50-1).

6. This renewed operating license is effective as of the date of issuance, and shall expire at midnight on March 8, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

R. W. Borchardt, Deputy Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A -Technical Specifications
2. Appendix B - Environmental Technical Specifications
3. Appendix C - Additional Conditions

Date of Issuance: December 22, 2005

APPENDIX C
ADDITIONAL CONDITIONS
OPERATING LICENSE DPR-27

NextEra Energy Point Beach, LLC shall comply with the following conditions and the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
244	Upon implementation of Amendment Nos. 240/244 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.6, 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 3.7.9.6, in accordance with Specification 5.5.18.c.(i), shall be within 18 months of implementation of this amendment. b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within three (3) years of completion of the testing prescribed in item a. above. c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 18 months of implementation of this amendment.	Immediately
244	NextEra Energy Point Beach, LLC shall modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained.	No later than the Unit 2 (2011) refueling outage
244	NextEra Energy Point Beach, LLC shall revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation.	No later than the Unit 2 (2011) refueling outage
244	NextEra Energy Point Beach, LLC shall modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications shall include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from an emergency diesel generator supplied source for the CREFS fans required for the new system alignment.	No later than the Unit 2 (2011) refueling outage
244	NextEra Energy Point Beach, LLC shall modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components required to direct radioactive releases in the PAB to the vent stack shall be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. NextEra Energy Point Beach, LLC shall revise PBNP EOPs to address starting the VNPAB fans.	No later than the Unit 2 (2011) refueling outage

APPENDIX C
ADDITIONAL CONDITIONS
OPERATING LICENSE DPR-27

NextEra Energy Point Beach, LLC shall comply with the following conditions and the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
244	NextEra Energy Point Beach, LLC shall perform Train B Emergency Diesel Generator load testing over a range of 2877 to 2950 kW at rated power factor. This license condition shall remain in effect until implementation of LAR 261 for Unit 2.	No later than the Unit 2 (2011) refueling outage
244	NextEra Energy Point Beach, LLC shall install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) shall be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.	No later than the Unit 2 (2011) refueling outage
244	NextEra Energy Point Beach, LLC shall procure mitigating filtration unit motors equivalent to W-14A/B (equivalent HP, efficiency, power factor, and voltage requirements).	No later than the Unit 2 (2011) refueling outage

1.1 Definitions

L_a	The maximum allowable primary containment leakage rate, L_a, shall be 0.2% of primary containment air weight per day at the peak design containment pressure (P_a).
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 >50 μCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	-------------------------NOTE------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT Xe-133 Specific Activity ≤ 520 μCi/gm.	7 days
SR 3.4.16.2	-------------------------NOTE------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.5 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Filtration System (CREFS)

LCO 3.7.9 CREFS shall be OPERABLE with:

a. Two control room recirculation fans,

b. Two control room emergency fans,

c. One filter train,

d. Two control room emergency fan control dampers, and

e. Two isolation dampers in the kitchen area exhaust duct.

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

APPLICABILITY: MODES 1, 2, 3, 4,
During movement of irradiated fuel assemblies

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE----------- Separate Condition entry is allowed for each component. ------------------------------ One control room recirculation fan inoperable. OR One control room emergency fan inoperable. OR One control room emergency fan control damper inoperable.	A.1 Restore inoperable fan or damper to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One isolation damper in the kitchen area exhaust duct inoperable.	B.1	Restore isolation damper to OPERABLE status.	7 days
	OR		
	B.2	Place and maintain the other isolation damper in the same duct in the closed position.	7 days
C. -----------NOTE----------- Separate Condition entry is allowed for each component. ---------------------------------	C.1	Initiate actions to implement mitigating actions.	Immediately
	AND		
Two control room recirculation fans inoperable.	C.2	Suspend movement of irradiated fuel assemblies.	Immediately
OR	AND		
Two control room emergency fans inoperable.	C.3	Verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits.	24 hours
OR	AND		
Two control room emergency fan control dampers inoperable.	C.4	Restore inoperable fans, dampers or filter train to OPERABLE status.	7 days
OR			
Filter train inoperable for reasons other than Condition D.			

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. -----------NOTE----------- Separate Condition entry is allowed for each component. ---------------------------------	D.1	Initiate actions to implement mitigating actions.	Immediately
	AND		
Filter train inoperable due to an inoperable CRE boundary	D.2	Suspend movement of irradiated fuel assemblies.	Immediately
OR	AND		
Two isolation dampers in the kitchen exhaust duct inoperable.	D.3	Verify mitigating actions ensure CRE occupant radiological and chemical exposures will not exceed limits, and CRE occupants are protected from smoke hazards.	24 hours
	AND		
	D.4	Restore CRE boundary to OPERABLE status.	90 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, 3, or 4 or not met during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	E.2	Be in MODE 3.	6 hours
	AND		
	E.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.9.1	Operate the CREFS for ≥ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify each CREFS emergency and recirculation fan actuates on an actual or simulated actuation signal.	18 months
SR 3.7.9.4	Verify each CREFS automatic damper in the emergency mode flow path actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.9.5	Verify CREFS manual start capability and alignment.	18 months
SR 3.7.9.6	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

3.7 PLANT SYSTEMS

3.7.13 Secondary Specific Activity

LCO 3.7.13 The specific activity of the secondary coolant shall be ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the specific activity of the secondary coolant is ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.14 Primary Auxiliary Building Ventilation (VNPAB)

LCO 3.7.14 VNPAB shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. VNPAB inoperable.	A.1 Restore VNPAB to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Operate the VNPAB filter and stack fans for ≥ 15 minutes. Verify the associated low flow lights for filter fans and for stack fans are not lit.	31 days
SR 3.7.14.2	Verify the VNPAB system can maintain a PAB pressure less than atmospheric pressure and less than turbine building pressure.	18 months
SR 3.7.14.3	Verify VNPAB manual start capability and alignment.	18 months

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

b. The peak design containment internal accident pressure, P_a, is 60 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be 0.2% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

3. Air lock testing acceptance criteria are:

   i. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   ii. For each door seal, leakage rate is equivalent to $\leq 0.02\ L_a$ at $\geq P_a$ when tested at a differential pressure of $\geq$ to 10 inches of Hg.

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

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

5.5 Programs and Manuals

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. 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. Additionally, separate from the CREFS, the program shall ensure CRE occupants can maintain the reactor in a safe condition following a hazardous chemical release or smoke challenge. 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 technical specification emergency mode of operation by the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 18 months. The results shall be trended at a frequency of 18 months and used as part of the periodic 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.

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 Programs and Manuals

5.5.18 Control Room Envelope Habitability Program (continued)

g. An adequate supply of self contained breathing apparatus (SCBA) units in the CRE to protect CRE occupants from a hazardous chemical release.

h. Portable smoke ejection equipment per the Fire Protection Evaluation Report and Safe Shutdown Analysis Report to address a potential smoke challenge.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(4) WCAP-14787-P, Rev. 2, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header), October, 2002 (approved by NRC Safety Evaluation, November 29, 2002).

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

(11) Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

(12) Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System," Revision 0, May 2000.

(13) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005.

(14) WCAP-16259 P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis."

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 240 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

AND

AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-27

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application dated December 8, 2008[1], as supplemented by additional letters[2], NextEra Energy Point Beach, LLC (formerly Florida Power & Light Energy Point Beach, LLC)(the licensee) requested changes to the Technical Specifications (TSs) for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 12, 2010 (75 FR 62602).

The proposed changes would revise the TSs to fully implement an alternative source term (AST) methodology at PBNP, Units 1 and 2. The application provides the TS changes and evaluations of the radiological consequences of design-basis accidents (DBAs) for

[1] Agencywide Documents Access and Management System (ADAMS) Accession Number (AN) ML083450683.

[2] January 16, 2009 (AN ML090160571), January 27, 2009 (AN ML090280348), February 20, 2009 (AN ML090540860), April 17, 2009 (2 letters – AN ML091100215 and AN ML091100182), May 8, 2009 (AN ML091320437), May 15, 2009 (AN ML091380113), June 1, 2009 (AN ML091560413), July 24, 2009 (AN ML092080441), August 20, 2009 (AN ML092330180), September 4, 2009 (2 letters – AN ML092520547 and AN ML092510118), September 10, 2009 (AN ML092540144), October 2, 2009 (AN ML092750348), November 20, 2009 (AN ML093310308), November 25, 2009 (AN ML093290322), December 17, 2009 (AN ML093560112), January 14, 2010 (AN ML100190066), February 4, 2010 (2 letters – AN ML100360065 and AN ML100360077), March 5, 2010 (AN ML100670043), April 20, 2010 (AN ML101100605), July 8, 2010 (AN ML101890783), July 29, 2010 (AN ML102110122), August 12, 2010 (AN ML102250367), September 3, 2010 (AN ML102460115), October 12, 2010 (AN ML102860121), November 16, 2010 (AN ML103210186), January 27, 2011 (AN ML110270085), February 10, 2011 (AN ML110420103), March 11, 2011 (AN ML110730295), and April 4, 2011 (AN ML110970363).

implementation of the AST in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67 (10 CFR 50.67) and by using the methodology described in Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.

In support of the AST implementation, the licensee also requested NRC approval of the following items: (1) modification of the control room emergency filtration system (CREFS) to create a new alignment for the accident mode; (2) approval to continue containment spray, while on sump recirculation, on certain conditions; and (3) use of the Westinghouse RAVE methodology to determine the percentage of fuel rods in departure from nucleate boiling (DNB) for the analysis of the reactor coolant pump (RCP) locked rotor (LR) event. TS 5.6.4, "Core Operating Limits Report (COLR)," will be revised to include the new analytical methodology described in Westinghouse Commercial Atomic Power (WCAP)-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA [Loss of Coolant Accident] Analyses."

The proposed changes also revise the surveillance requirements of TS 3.4.16, "RCS [reactor coolant system] Specific Activity," and TS 3.7.13, "Secondary Specific Activity," to implement the AST accident offsite and control room dose consequences.

Implementation of the AST supports adoption of the control room envelope habitability controls in accordance with NRC-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-448, Revision 3, "Control Room Habitability." On August 8, 2006, the commercial nuclear electrical power generation industry owners group submitted a proposed change, TSTF-448, Revision 3, to the improved standard technical specifications (STS) (NUREGs 1430-1434) on behalf of the industry (TSTF-448, Revisions 0, 1, and 2 were prior draft iterations). 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 control room envelope (CRE).

In NRC Generic Letter (GL) 2003-01, "Control Room Habitability," dated June 12, 2003[3] (ADAMS Accession No. ML031620248), licensees were alerted to findings at facilities indicating that existing TS surveillance requirements for the Control Room Envelope Emergency Ventilation System (CREEVS) may be inadequate. Specifically, the results of ASTM [American Society for Testing and Materials] E741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," issued in 2000, utilize tracer 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*

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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 GL 2003-01, 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." The Notice of Availability for adopting TSTF-448, Revision 3, was published in the *Federal Register* on January 17, 2007 (72 FR 2022).

Consistent with the traveler as incorporated into NUREG-1431, the licensee proposed revising action and surveillance requirements in Specification 3.7.9, "Control Room Emergency Filtration System (CREFS)," and adding a new administrative controls program, Specification 5.5.18, "CRE 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. Some editorial and plant specific changes were incorporated into this safety evaluation (SE) resulting in minor deviations from the model SE text in TSTF-448, Revision 3.

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ATTACHMENT 1

ATTACHMENT 2

ATTACHMENT 3

2.0 EVALUATION

2.1 Radiological Consequences Analyses

2.1.1 Regulatory Evaluation

The NRC staff reviewed the licensee's evaluation of the radiological consequences of affected DBAs for implementation of the AST methodology, and the associated changes to the TS proposed by the licensee, against the requirements specified in 10 CFR 50.67(b)(2). Section 50.67(b)(2) of 10 CFR requires that the licensee's analyses demonstrate with reasonable assurance that:

- 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 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE).

- 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 25 rem TEDE.

- Adequate radiation protection is provided to permit access to and occupancy of the control room (CR) under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

This SE addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the reference 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. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulations, regulatory guides, and standards:

- 10 CFR Part 50.67, "Accident source term";

- 10 CFR Part 50, Appendix A, "General Design Criterion (GDC) for Nuclear Power Plants," GDC 19, "Control room";

- RG 1.23, "Onsite Meteorological Programs," Revision 0, issued February 1972;

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, issued March 2007;

- RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," Revision 0, issued March 1972;

- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 3, issued June 2001;

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued November 1982;

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, issued July 2000;

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, issued June 2003;

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 0, issued May 2003;

- NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," published May 1985;

- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Revision 3, published March 2007;

- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Revision 3, published March 2007;

- NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, published March 2007;

- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, published July 2000;

- NUREG-0800, "Standard Review Plan," Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, published July 1981;

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," published February 1995; and

- NUREG/CR-5950, "Iodine Evolution and pH Control," published December 1992.

The NRC staff also considered relevant information in the PBNP, Units 1 and 2, Safety Analysis Report (SAR) and TSs.

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) and the integrated dose to a PBNP, Units 1 and 2 CR operators were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, computer code. The NRC sponsored the development of the RADTRAD radiological consequence computer code, as described in NUREG/CR-6604. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff uses the RADTRAD computer code to perform independent confirmatory dose evaluations as needed to ensure a thorough understanding of the licensee's methods. Although the NRC staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses.

2.1.2 Technical Evaluation

2.1.2.1 Atmospheric Dispersion Estimates

Meteorological Data

The Point Beach meteorological measurement program was upgraded from December 1999 to December 2005. To support this license amendment request (LAR), the licensee used five years of hourly onsite meteorological data collected between September 2000 and September 2005 to generate new CR air intake atmospheric dispersion factors (χ/Q values). Wind speed and wind direction were measured at the 45 and 10 meter levels and the atmospheric stability categorization was based on temperature difference measurements between these two levels. The measurements were primarily from the primary tower located about 40 meters inland of the Lake Michigan shoreline. A backup tower located about 300 meters from the Lake Michigan shoreline is instrumented at the 10 meter level to provide data when measurements from the primary tower are not available. A third tower is located inland, about 8 miles from the Point Beach site, to provide additional information on effects of the land-lake interface on the local meteorology. Instrument calibrations are performed on a semi-annual basis, as well as after major equipment malfunctions, equipment modifications and replacements. A visual inspection is performed at each tower site at least once per month to check the physical integrity of the site and the appearance of the sensors for any obvious signs of damage or faulty operation and to verify that the signal conditioning equipment is operating properly. The towers are sited to minimize the effects of potential obstructions such as trees and facility structures. Instruments are placed to reduce tower interference, and sensors used to measure temperature difference are shielded from direct sunlight and precipitation.

The September 2000 to September 2005 data were provided for NRC staff review in the form of hourly meteorological data files for input into the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake"). The resultant χ/Q values were used to estimate CR dose consequences from the postulated DBAs associated with the current LAR. The NRC staff also generated a joint wind speed, wind direction, and atmospheric stability frequency distribution using these data to calculate EAB and low population zone (LPZ) χ/Q values for comparison with χ/Q values previously calculated by the licensee.

The NRC staff performed a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," published in July 1982. Further review was performed using computer spreadsheets. Examination of the data files revealed that while stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions during the day, as expected, there was a higher reported occurrence of unstable and stable conditions than at many other nuclear power plant sites. In addition, there was a somewhat higher occurrence of unstable conditions at night and stable conditions during the day. As part of a previous Point Beach LAR[3], the licensee provided a discussion of historic measurements and noted that the difference in stability measurements appeared to be related to the primary tower's immediate proximity to a large body of water. In addition, NRC staff notes that temperature differences measured between the 45 to 10 meter levels may skew atmospheric stability categorization to more frequently report unstable and stable categories than when temperature difference measurements are made between the 60 and 10 meter levels as recommended in RG 1.23, Revision 1. If this occurred and had an effect on the calculations, then the NRC staff judges that it likely would result in a slight over-estimate of the χ/Q values and resultant dose estimates generated in support of this LAR. Wind speed and wind direction frequency distributions for each measurement channel were reasonably similar from year to year. The combined data recovery of the wind speed, wind direction, and stability data was in the mid-to-upper 90 percentiles at both levels during 2000 and 2002 through September 2005. The combined data recovery in 2001 was about 91 percent. This meets the data recovery recommendation of RG 1.23.

In summary, the NRC staff reviewed available information relative to the onsite meteorological measurements program, the 2000 through 2005 meteorological data measured at the PBNP site, and the ARCON96 meteorological data input files provided by the licensee. Based on this review, the NRC staff concludes that the data provides an acceptable basis for making estimates of atmospheric dispersion for the DBA CR dose assessments associated with the current LAR.

<u>Control Room Atmospheric Dispersion Factors</u>

In the December 8, 2008, submittal letter, the licensee postulated releases from the following locations to the control room air intake:

- Unit 1 and Unit 2 Containment Wall
- Unit 1 and Unit 2 Containment Façade
- Auxiliary Building Vent
- Drumming Area Vent
- Unit 1 "A" and Unit 1 "B" Main Steam Safety Valves (MSSVs)
- Unit 2 "A" and Unit 2 "B" MSSVs
- Spent Fuel Pool (SFP)

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- Units 1 and 2 Purge Stacks
- Units 1 and 2 Refueling Water Storage Tanks (RWSTs)

To assess the CR post-accident atmospheric dispersion conditions, the licensee generated χ/Q values using the ARCON96 computer code and guidance provided in RG 1.194. The RG 1.194 indicates that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there is no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for Point Beach. All sources were modeled as ground level releases based upon guidance provided in RG 1.194. The shortest horizontal distance between each release location and the control room intake was input as the distance between the release and receptor locations. Releases from the Containment Wall were modeled as diffuse sources with initial sigma-y and sigma-z input values of 5.7 and 6.6 meters, respectively. All other releases were modeled as point sources.

The licensee stated that all potential release scenarios for the DBAs associated with the current LAR were considered, including those due to single failures. The licensee compared the resultant χ/Q values and found the χ/Q values associated with releases from the following locations to be limiting for each of the indicated DBAs:

- Loss of Coolant Accident – Unit 2 Containment Wall, Auxiliary Building Vent, Unit 2 RWST
- Steam Generator Tube Rupture – Unit 2 "A" MSSVs
- Locked Rotor – Unit 2 "A" MSSVs
- Main Steam Line Break – Unit 2 "A" MSSVs and Unit 2 Containment Façade
- Control Rod Ejection – Unit 2 Containment Wall and Unit 2 "A" MSSVs
- Fuel Handling Accident – Unit 2 Purge Stack
- Reactor Vessel Head Drop – Auxiliary Building Vent, Unit 2 RWST

Subsequently, by letter dated September 3, 2010, the licensee provided a summary of the loss-of-coolant accident (LOCA) control room does analysis performed without credit for the Primary Auxiliary Building Ventilation System (VNPAB). This included a revision to the CR χ/Q values for the assessment considering Emergency Core Cooling System (ECCS) leakage to the Primary Auxiliary Building (PAB), by taking no credit for the VNPAB. The licensee provided data, figures, and a discussion to support use of the χ/Q values which were calculated for a postulated release from the PBNP Unit 2 façade roof vent 2-V7 in enclosures to an October 12, 2010, supplemental letter to the NRC.

The NRC staff reviewed the licensee's assessments of CR post-accident atmospheric dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The NRC staff qualitatively reviewed inputs to the ARCON96 computer runs for the CR χ/Q value assessment and found them generally consistent with site configuration drawings and NRC staff practice. In addition, NRC staff performed a check of the licensee's atmospheric dispersion estimates by running the ARCON96 computer code with application of the RG 1.194 criteria and obtained similar results. On the basis of this review, the

NRC staff has concluded that the licensee's CR χ/Q values listed in Table 3.1-1 are acceptable for use in the DBA CR dose assessments.

Offsite Atmospheric Dispersion Factors

The licensee used the current Point Beach licensing basis EAB and LPZ χ/Q values listed in Table 3.1-2 to assess the radiological consequences of the DBAs postulated in this LAR. These χ/Q values are presented in the PBNP Final Safety Analysis Report (FSAR) in Table 14.3.5-2 which lists parameters applicable to the loss of coolant accident. The χ/Q values were generated based upon guidance in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," using onsite data collected from 1991 through 1993. The licensee assessed the χ/Q values in the FSAR against χ/Q values calculated using the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radiological Materials from Nuclear Power Stations") and the onsite meteorological data collected between September 2000 and September 2005 and determined that the current licensing basis EAB and LPZ χ/Q values are conservative. The NRC staff also made comparison estimates using PAVAN and the September 2000 to September 2005 data and agrees with the licensee's conclusion. On the basis of this review, the NRC staff concludes that the χ/Q values listed in Table 3.1-2 are acceptable for use in the dose assessments associated with the current LAR.

2.1.2.2 Radiological Consequences of Design Basis Accidents

The licensee has proposed a licensing basis change for its offsite and control room DBA dose consequence analysis for PBNP Units 1 and 2. The proposed change will implement an AST methodology for determining DBA offsite and CR dose. For full implementation of the AST DBA 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 and GDC 19. To incorporate a full implementation of the AST, RG 1.183, Regulatory Position 1.2.1, specifies that the DBA LOCA must be reanalyzed.

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 specific proposed applications of an AST.

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences of the following DBAs:

- Loss-of-Coolant Accident
- Steam Generator Tube Rupture
- Locked Rotor Accident
- Main Steam Line Break
- Control Rod Drive Ejection Accident

- Fuel Handling Accident
- Reactor Vessel Head Drop Accident

Although the current licensed maximum reactor core power level for PBNP is 1540 megawatts thermal (MWt), the above analyses assume a maximum core power of 1800 MWt with a 0.6 percent uncertainty (analyzed core power of 1811 MWt). This power level was chosen to support a future extended power uprate (EPU) license amendment request and is conservative with respect to the current licensed power level.

The licensee has proposed to modify CREFS to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications will include redundancy for all CREFS active components and auto-start capability on loss of offsite power from a diesel generator supplied source for the CREFS fans required for the new system alignment. This mode is referred to as control room ventilation system (VNCR) accident mode to avoid confusion with plant operating MODES in the TS. The licensee is requesting NRC approval of this new operational mode. The acceptability of this proposed accident mode was reviewed by the NRC staff and discussed in Section 2.5 of this SE.

2.1.2.2.1 Loss of Coolant Accident

The LOCA event is assumed to be caused by an abrupt failure of a main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation. The analysis considers the release of activity from the containment via containment leakage. In addition, once the recirculation mode of the ECCS is established, activity in the sump solution may be released to the environment by ECCS leakage into the PAB and into the RWST. The licensee does not take credit for auxiliary building vent stack filtration.

The core inventory release fractions and release timing for the gap and early in-vessel release phases of the DBA LOCA were taken from RG 1.183, Tables 2 and 4, respectively. Also consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation released from failed fuel is 95 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioactive iodine special released from the SGs is 97 percent elemental and 3 percent organic.

The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Containment leakage directly to atmosphere
- Leakage from ECCS outside containment

Containment Leakage

The PBNP containment is projected to leak at its design leakage of 0.2 percent of its contents by weight per day for the first 24 hours and then at 0.1 percent for the remainder of the 30-day accident duration. For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment.

The licensee uses a two-region containment transport model in assessing the containment leakage pathway. This model is comprised of a region that is sprayed by the containment spray system and an unsprayed region. The sprayed region envelopes 58.2 percent of the total free volume of the containment. One train of the containment spray (CS) system is assumed to operate in the injection mode following the LOCA. When the RWST drains to a predetermined level, the operators switch to recirculation of the sump liquid to provide a source to the sprays. The minimum injection spray duration is 60 minutes. The switchover is assumed to take 20 minutes. During this time, the analysis does not credit any spray removal in the containment. The residual heat removal (RHR) system injection flow will be adjusted in post-modification testing to maintain greater than 500 gallons per minute (gpm) during the alignment and following initiation of containment spray on recirculation. CS flow on recirculation is greater than 900 gpm. During spray operation, credit is taken for sedimentation removal of particulates in the unsprayed region. After spray termination and during the 20-minute switchover from injection to recirculation, credit for sedimentation is taken in both the sprayed and unsprayed regions.

Credit is taken for reduction of airborne radioactivity in the containment by natural deposition. The elemental spray coefficient is initially assumed to be 20 hr^{-1} (limit per SRP 6.5.2), but is conservatively further reduced to 9.2 hr^{-1} at the start of sump recirculation. The licensee limits the elemental iodine decontamination factor (DF) to a maximum value of 200, which is determined to occur at 2.73 hours. Similarly, the licensee decreases that aerosol removal coefficient by a factor of ten when the aerosol DF reaches 50, which was determined to occur at 3.3 hours. The natural deposition removal coefficient for particulates was determined to be 0.1 hr^{-1}. It is assumed that sedimentation removal does not continue beyond a DF of 1000, which is reached at 31.6 hours.

The NRC staff reviewed the licensee's proposed changes to the PBNP AST LOCA dose analysis submitted in a January 27, 2011, supplemental letter. The licensee assumed two hours of CS system operation during post-LOCA recirculation, instead of three hours, to accommodate additional margin for the boron precipitation analysis necessary to support EPU conditions. The NRC staff evaluated the impact of the reduced CS time on the calculated dose, and determined that the change would result in a minor increase to the CR dose because most of the iodine scrubbing will occur within the proposed two hours of recirculation spray time (1.33 hours to 3.33 hours).

The licensee states that new operator actions to align core injection and CS flow to preset throttled positions will be introduced as a result of the new LOCA radiological analysis. The review of the acceptability of these proposed manual actions was reviewed by the NRC staff and is discussed in Section 2.5 of this SE.

Emergency Core Cooling System (ECCS) Leakage

For the ECCS leakage analysis, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS. When ECCS recirculation is established following a LOCA, leakage is assumed to occur from ECCS equipment outside of containment. Recirculation is conservatively initiated at 0 minutes. The assumption of the ECCS leakage beginning at 0 minutes is not consistent with the assumption of injection spray termination in the containment leakage portion of the analysis. However, beginning the ECCS leakage at 0 minutes adds conservatism to the dose consequences. The leakage continues for the 30-day period following the accident considered in the analysis.

Activity enters the sump and flows out of containment in the ECCS recirculation flow and is released to the environment through leakage from the ECCS. Only iodine is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the core.

Generally, the NRC staff expects that the amount of iodine assumed to flash and become airborne should be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and ventilation rate. The term pH (from potential of Hydrogen) is the logarithm of the reciprocal of hydrogen-ion concentration in gram atoms per liter and provides a measure on a scale from 0 to 14 of the acidity or alkalinity of a solution (where a pH of 7 is neutral, a pH greater than 7 is more basic, and a pH less than 7 is more acidic). In the September 3, 2010, letter, the licensee proposed to use a calculated time-dependent iodine flashing fraction that would range from 7 percent at the start of the containment sump water recirculation and gradually decrease to 2 percent. The flashing fraction would remain at 2 percent for the duration of the 30-day accident. For this estimate, the licensee used actual PBNP specific sump water temperature and pressure circulating outside the containment following a LOCA. The licensee then calculated the time-dependent flashing fraction using the constant enthalpy equation as discussed in RG 1.183. The PBNP sump water pH history, as stated in the PBNP FSAR, ranges from 7.0 to 10.5, and therefore is consistently basic. The NRC staff independently confirmed the licensee's calculation. Base on the licensee's use of constant enthalpy equation and actual sump coolant data (i.e., basic sump history, temperature and pressure), the NRC staff concludes that the calculated flashing fraction range of 7 percent to 2 percent for the amount of iodine that becomes airborne is acceptable.

The VNPAB exhaust system ensures that the PAB vent stack is the source of the release associated with the ECCS equipment leakage during the containment sump recirculation phase of a large-break LOCA, by providing exhaust flow from areas which have possible contamination. The VNPAB system operates during normal unit operation. The VNPAB is classified as non-safety related. The VNPAB exhaust system consists of two filter fans, two stack fans, and the associated ductwork and backdraft dampers necessary to ensure the required exhaust flow path can be maintained. The VNPAB fans are powered by safety-related power supplies with diesel generator backup. According to the licensee, the VNPAB system will be modified to provide redundancy for all active components needed to operate the PAB exhaust system. Although the licensee is not taking credit for the use of VNPAB in its AST analysis, the licensee has proposed a TS for the VNPAB, TS 3.7.1.4, "Primary Auxiliary Building Ventilation (VNPAB)," which is described in an April 17, 2009, letter.

The licensee states that new operator actions to align core injection and CS flow to preset throttled positions will be introduced as a result of the new LOCA radiological analysis. The acceptability of this proposed accident mode was reviewed by the NRC staff and is discussed in Section 2.5 of this SE.

Control Room Dose

The licensee analyzed the gamma shine dose to the CR operators for the bounding LOCA accident. The dose contribution in the CR due to direct shine from the external cloud and from contained sources is analyzed. The external cloud contribution includes containment leakage, ECCS leakage, and RWST back-leakage. The contained sources include shine from the containment structure and the CR heating, ventilation, and air conditioning (HVAC) filter. The 30-day deep dose equivalent (DDE) to a CR operator due to the airborne source in containment, the passing plume source and the CR filter source is calculated. The analysis takes credit for shielding modifications to the CRE, including the CR walls and ceiling.

The proposed CR VNCR accident mode consists of filtered outside air combined-with filtered recirculation. The control room HVAC filter shine dose is calculated based on accumulation of a) the particulate fission products and elemental/organic/particulate iodines resulting from containment leakage and b) elemental/organic iodines resulting from ECCS leakage and RWST back-leakage. The elemental and organic iodines are assumed to be accumulated on the charcoal filter. The particulate iodines and the other particulate fission products are assumed to be accumulated on the high-efficiency particulate air (HEPA) filter. All non-noble gas radioactive materials that enter the control room, either via the control room intake or via control room inleakage, are accumulated on the filter with 100 percent efficiency. This maximizes the loading on the filter for the duration of the accident.

Computer code SW-QADCGGP is used to calculate the direct shine dose to an operator in the control room from the airborne source inside containment, external plume source, and the control room charcoal/HEPA filter sources. SW-QADCGGP is a Shaw S&W version of the industry standard point-kernel radiation shielding computer code QAD-CGGP. The geometry utilized in the model does not have any significant unaccounted for scattering paths from the source to the receptor. Multiple receptors are placed inside the control room to ensure that the maximum dose is calculated. The source-shield-receptor geometry is such that the dose due to oblique angle scattering is not significant. The most conservative buildup factor is used if the gamma rays traverse through a multiplicity of materials and the last material constitutes less than 3 mean-free-paths. The dose due to direct shine from the external cloud and contained source is 0.28 rem.

In the event of a large break LOCA, the safety injection (SI) setpoint will be reached shortly after event initiation. The SI/containment isolation signal causes the VNCR system to switch from the normal operation mode to the VNCR accident mode of operation. The SI setpoint is assumed to be reached immediately at the start of the event and a conservative 60-second delay time for switching from normal to the VNCR accident mode (filtered recirculation with filtered fresh air intake) is modeled. Following isolation, there will be no unfiltered outside air makeup and the filtered makeup flow of 2500 cfm and recirculation flow of 1955 cubic feet per minute (cfm) is initiated. The licensee assumes an unfiltered inleakage rate of 200 cfm and recirculation filter efficiencies of 95 percent elemental, 95 percent organic, and 99 percent particulate.

Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, by confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The LOCA analysis assumption and parameters can be found in Table 3.2-2 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.2 Steam Generator Tube Rupture

A double-ended rupture of a single SG tube is assumed to occur. At the start of the accident, radionuclides from the primary coolant enter the SG, via the ruptured tube and primary-to-secondary leakage, and are released to the atmosphere through the condenser air ejector exhaust via the auxiliary building vent stack prior to reactor trip. The primary-to-secondary break flow results in depressurization of the RCS. Reactor trip and SI are assumed to be automatically initiated simultaneously on low pressurizer pressure. For calculating dose rates, a loss of offsite power is assumed concurrent with the reactor trip; therefore, use of the condenser is lost and the steam is released via the SG safety or atmospheric dump valves (ADVs).

In accordance with RG 1.183 guidance, two iodine spiking cases were considered: (1) a pre-accident iodine spike case and (2) an accident initiated iodine spike case. The pre-accident iodine spike case assumes that a reactor transient occurs prior to the SG tube rupture (SGTR) and has raised the primary coolant iodine concentration to a conservative value of 60 microcuries per gram (μCi/gm) Dose Equivalent Iodine-131 (DEI-131). The accident initiated iodine spike case assumes the primary coolant activity was initially at the proposed TS limit of 0.5 μCi/gm DEI-131 when the event occurs. The accident is assumed to cause the iodine concentration to spike by addition of iodine activity by a factor of 335 times the equilibrium iodine appearance rate for 8 hours.

The primary-to-secondary coolant leakage of 500 gallons per day (gpd) per SG goes to the intact SG. The activity in the coolant is available for release to the environment through secondary coolant steaming through the SG ADVs. The licensee assumes an iodine partitioning factor of 0.01 in the intact SG in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released. Primary coolant is assumed to pass through the ruptured SG tubes and be available for release to the outside environment by steaming through the ruptured SG. A portion of the rupture flow flashes directly to steam and the remainder mixes with secondary coolant for subsequent steaming through the ADV. The flashing fraction and partitioning fractions for the ruptured SG are in accordance with RG 1.183 guidance.

The CR VNCR system begins in normal mode. Actuation of the VNCR accident mode is conservatively assumed to occur when the SI/containment isolation actuation setpoint is reached at 220 seconds. Based on the release during the first moments of the accident, the source term is large enough that the radiation monitor alarm setpoint would have been reached within one second post accident. In addition, a delay of 60 seconds is assumed to account for HVAC configuration alignment, e.g., damper position changes.

Summary

The licensee evaluated the radiological consequences resulting from the postulated SGTR using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, a confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The SGTR analysis assumptions and parameters can be found in Table 3.2-3 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.3 Locked Rotor Accident

An instantaneous seizure of a RCP rotor is assumed to occur. This rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage is predicted to occur due to departure from nucleate boiling (DNB) as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the ADVs or MSSVs. In addition, iodine and alkali metal activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the SGs following the accident.

The licensee used Westinghouse Commercial Atomic Power (WCAP) -16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," to estimate the percentage of failed fuel (rods in DNB) assumed in the locked rotor (LR) event radiological analysis. The licensee is requesting NRC review and approval of the plant-specific application of WCAP-16259-P-A and associated TS changes to support the LR radiological analysis input assumption that 30 percent of the fuel rods in the core experience DNB. The review of the acceptability of utilizing WCAP-16259-P-A is discussed in Section 2.6 of this SE.

A radial peaking factor of 1.7 was applied. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage of 500 gpd for each SG. The licensee assumed that this leakage mixes with the bulk water of the SGs and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The tubes in the SGs would remain covered by the bulk water. The licensee assumed that the radionuclide concentration in the SG is partitioned such that 1 percent of the radionuclides in the bulk water of the SGs enters the vapor space and is released to the environment. The steam release from the SGs continues until the RHR system can be used to complete the cooldown at approximately 30 hours.

The CR HVAC is switched to the VNCR accident mode after receiving a high radiation ventilation system line monitor signal. This signal is reached within one minute, however a conservative time of two minutes is assumed to switch the CR to the VNCR accident mode. Following isolation, there will be no unfiltered outside air makeup and the filtered makeup flow of 2500 cfm and recirculation flow of 1955 cfm is initiated. The licensee assumes an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 95 percent elemental, 95 percent organic, and 99 percent particulate.

Summary

The licensee evaluated the radiological consequences resulting from the postulated LR using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, by confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The LR analysis assumption and parameters can be found in Table 3.2-4 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.4 Main Steam Line Break

The accident considered is the complete severance of a main steam line outside the containment but upstream of the main steam isolation valves (MSIVs). Upon a main steam line break (MSLB), the affected SG rapidly depressurizes. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the unaffected SG. The radiological consequences of a break outside containment will bound the results from a break inside containment. Because of this MSLB, the secondary water in the affected SG completely flashes to steam within two minutes.

In accordance with RG 1.183 guidance, two iodine spiking cases were considered: (1) a pre-accident iodine spike case and (2) an accident initiated iodine spike case. The pre-accident iodine spike case assumes that a reactor transient occurs prior to the MSLB and has raised the primary coolant iodine concentration to a conservative value of 60 μCi/gm DEI-131. The accident-initiated iodine spike case assumes the primary coolant activity was initially at the proposed TS limit of 0.5 μCi/gm DEI-131 when the event occurs. The accident is assumed to cause the iodine concentration to spike by addition of iodine activity by a factor of 500 times the equilibrium iodine appearance rate for 4 hours.

Leakage from the RCS to the SGs is assumed to be the maximum value permitted by TSs. Primary-to-secondary leakage is assumed to be 500 gpd each to the faulted and intact SGs. The leakage to the affected SG is assumed to immediately flash to steam and be released to the environment without holdup or dilution. The leakage in the unaffected SG mixes with bulk water and is released at the assumed steaming rate. Within 60 hours after the accident, the reactor coolant system has been cooled to below 212°F, and there are no further steam releases to the atmosphere from the faulted SG. The licensee assumed an iodine partitioning factor of 0.01 in the unaffected SG in accordance with RG 1.183 guidance. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

In the event of a MSLB, the low steam line pressure SI setpoint will be reached shortly after event initiation. The SI/containment isolation signal or a radiation monitor signal cause the VNCR system to switch from the normal operation mode to the VNCR accident mode of operation. The analysis conservatively did not credit the SI signal but relied on the ventilation system line radiation monitor signal for CR isolation. It was confirmed that the radiation monitor setpoint is reached within 15 seconds. The VNCR system switches from normal operation to VNCR accident mode of operation at 75 seconds (15 seconds for radiation signal plus 60 second delay time).

Summary

The licensee evaluated the radiological consequences resulting from the postulated MSLB using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, a confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The MSLB analysis assumption and parameters can be found in Table 3.2-5 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.5 Control Rod Drive Ejection Accident

The accident considered is the mechanical failure of a control rod drive mechanism pressure housing, resulting in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected due to the reactivity spike. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip occurs, safety injection actuates, and a loss of offsite power (LOOP) occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage) and
- Release of RCS inventory via primary-to-secondary leakage through SGs.

While the actual doses from a control rod drive ejection (CRDE) accident would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would show that the composite dose would also be acceptable.

The licensee assumed that 10 percent of the fuel rods fail, releasing the radionuclide inventory in the fuel rod gap. The licensee further assumed that 10 percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are in the fuel gap, consistent with the guidance provided in RG 1.183. A radial peaking factor of 1.7 was applied. In addition, localized heating is assumed to cause 0.25 percent of the fuel to melt, releasing 100 percent of the noble gases and 50 percent of the iodine and alkali metals contained in the melted fuel to the containment. For the secondary release case, 100 percent of the noble gases and 50 percent of the iodine and alkali metals contained in the melted fuel are released to the secondary.

For the containment leakage case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. The licensee assumed that the containment leaks at the TS value of 0.2 percent volume per day for the first 24 hours and 0.1 percent volume per day for the remainder of the 30 day period following the accident. This is consistent with the RG 1.183 guidance. No credit is taken for plateout onto containment surfaces or for containment spray operation, which would remove airborne particulates and elemental iodine. However, sedimentation of particulates is credited at a removal rate of 0.1 per hour.

For the secondary release case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage at 500 gpd for each SG for 2000 seconds. This time is used because the licensee's analyses of the small break LOCA pressure transient have shown that the primary pressure is less than the secondary pressure before this time. The licensee assumed that this leakage mixes with the bulk water of the SGs and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The licensee conservatively assumed that the chemical form of the radioiodine released to the environment would be 97 percent elemental and 3 percent organic consistent with the guideline provided in RG 1.183. The licensee assumed that the aerosol and iodine radionuclide concentration in the SG is partitioned such that 1 percent of the radionuclides entering the unaffected SG from the RCS will enter the vapor space and be released to the environment. This assumption is also consistent with the guideline provided in RG 1.183. Steam releases from the SG are conservatively assumed to continue for 30 hours.

In the event of a CRDE, the low pressurizer pressure SI setpoint will be reached at approximately 76 seconds, which is rounded up to 90 seconds, after event initiation. The SI/containment isolation signal causes the CR HVAC [heating, ventilation, and air conditioning] to switch from the normal operation mode to the post-accident mode of operation. It is assumed that the VNCR system switches from normal operation to VNCR accident mode of operation at 150 seconds. This time accounts for the 90 seconds for SI/containment isolation signal plus a 60-second delay time. Following isolation, there will be no unfiltered outside air makeup and the filtered makeup flow of 2500 cfm and recirculation flow of 1955 cfm is initiated. The licensee assumes an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 95 percent elemental, 95 percent organic, and 99 percent particulate.

Summary

The licensee evaluated the radiological consequences resulting from the postulated CRDE using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, by confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The CRDE analysis assumption and parameters can be found in Table 3.2-6 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.6 Fuel Handling Accident

The fuel handling accident (FHA) analysis postulates that a spent fuel assembly is dropped during refueling. The kinetic energy developed in this drop is conservatively assumed to be dissipated to the cladding on all fuel rods in the dropped assembly. The fission product inventory in the core is largely contained in the fuel pellets that are enclosed in the fuel rod clad. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod clad. The licensee assumed that the core inventory of fission products, which has decayed for 65 hours, is equally distributed in all fuel assemblies in the core. To account for differences in core power distribution across the core, the averaged fission product inventory in the dropped assembly is conservatively multiplied by a radial peaking factor of 1.7. The entire gap inventory in the damaged rods is assumed to be release from the fuel. The licensee assumed that 12 percent of the I-131 inventory of the core was in the fuel rod gap (modified per NUREG/CR-5009), along with 30 percent of the Kr-85 and 10 percent of all other iodines and noble gases. This change reflects the assumption that the damaged nuclear fuel used in the FHA radiological analysis is assumed to exceed the RG 1.183, Table 3, Footnote 11, criteria. The gap fractions are those from RG 1.25 with the value for I-131 increased by 20 percent, consistent with the recommendation in NUREG/CR-5009. Alkali metals make a negligible contribution to dose for this analysis and are omitted.

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 assumed 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 iodine. The CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine from CsI re-evolves as elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

The fission product inventory in the fuel rod gap of the fuel rods assumed to be damaged is postulated to be instantaneously released because of the accident. The associated activity is assumed to be released from the containment refueling cavity or the SFP to the environment over two hours. The quantity of fission products released from the damaged fuel is reduced by passage through the refueling cavity or SFP water. The licensee assumed a decontamination factor (DF) of 200 which is consistent with the guidance provided in RG 1.183. This DF is applicable to PBNP because the 23 feet minimum water level requirement by RG 1.183 is met. The licensee assumed no credit for removing fission products by containment and SFP building ventilation systems nor is credit taken for isolation of release paths.

Since the assumptions and parameters used to model the release due to an FHA inside containment are identical to those for an FHA in the SFP, except for different control room air intake atmospheric dispersion factors for the different release paths, the activity released is the same regardless of the location of the accident. The licensee assumed the accident occurred in the Unit 2 containment building and the release was through the purge stack, resulting in a bounding analysis for an FHA in either location.

For CR isolation, it is assumed that the time to switch the VNCR system from normal operation to VNCR accident mode of operation is 10 minutes after receiving an isolation signal. Following isolation, there will be no unfiltered outside air makeup and the filtered makeup flow of 2500 cfm and recirculation flow of 1955 cfm is initiated. The licensee assumes an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 95 percent elemental, 95 percent organic, and 99 percent particulate.

<u>Summary</u>

The licensee evaluated the radiological consequences resulting from the postulated FHA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, by confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The FHA analysis assumption and parameters can be found in Table 3.2-7 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.2.2.7 <u>Reactor Vessel Head Drop Accident</u>

The reactor vessel head is assumed to drop onto the vessel causing fuel cladding damage to all of the fuel assemblies in the core, which results in a gap release. In addition, damage to the bottom-mounted instrumentation tubes is assumed such that approximately 300 gpm of reactor coolant is lost through these penetrations. This loss of inventory is well within the capacity of a single SI or RHR pump. Damage to the point of rupture or shearing of other connected piping, including the main RCS loops, pressurizer surge line, core deluge lines, accumulator dump lines, normal charging, and cold leg safety injection lines is not predicted.

The initial makeup of the RCS to the vessel is via suction from the RWST to the SI pumps, RHR pumps or charging pumps. Once the RWST volume is exhausted, the RHR system is realigned to recirculate the coolant in the containment sump to maintain the core sub-cooled. The reactor vessel head is assumed to be lifted immediately following shutdown.

Because there is no guidance in RG 1.183, PBNP's reactor vessel head drop (RVHD) radiological analysis utilizes a combination of input assumption guidance obtained from the LOCA and FHA radiological analyses as they apply to the accident scenario. The RVHD analysis assumes that containment closure is established prior to the event and the following initial conditions are assumed: 1) containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlock is functional); 2) purge supply/exhaust system fans are off and isolation valves closed; and 3) other containment penetrations that allow containment atmosphere to communicate with the environment or the primary auxiliary building (PAB) atmosphere are closed. The event does not result in the pressurization of the containment building because the event occurs while the reactor coolant system is open to the containment building atmosphere and sufficient RCS makeup is available to provide cooling to the core. There is no release via containment leakage.

The RVHD is assumed to occur immediately upon reactor shutdown, and 100 percent of the fuel assemblies are damaged to produce a complete gap release. The accident occurs at temperatures and pressures well below operating levels and the accident mitigation strategy ensures that the core is covered and cooled. The 5 percent gap fraction for iodines for the design basis LOCA is assumed to be applicable to the RVHD. No fuel melting is assumed to occur. The amount of activity released from the gap is determined from the total core inventory assumed for the LOCA analysis with no assumed post-shutdown adjustment for decay time.

ECCS Leakage

When ECCS recirculation is established following the RVHD, leakage is assumed to occur from ECCS equipment outside containment. Recirculation is conservatively initiated at 0 minutes. The leakage continues for the 30-day period following the accident. The amount of coolant available for recirculation is assumed to be equal to the amount of coolant that is injected (243,000 gallons of RWST inventory). There is no credit for the volume of coolant initially in the vessel or the RCS which provides a conservative concentration of iodine available for release during ECCS recirculation.

Activity enters the sump and flows out of containment in the ECCS recirculation flow and is released to the environment through leakage from the ECCS. Only elemental iodine (100 percent elemental) is released through this pathway since the noble gases are not assumed to dissolve in the sump and particulates would remain in the water of the ECCS leakage. It is assumed that the iodine is instantaneously and homogeneously mixed in the containment sump water at the time of release from the core.

The licensee assumes that the total ECCS recirculation leakage modeled in the analysis is 0.21 gpm (i.e., the current analysis value of 400 cubic centimeters per minute (cc/min) total ECCS leakage outside the containment is doubled to 800 cc/min consistent with RG 1.183 guidance). Of the 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST. The analysis models ECCS recirculation leakage into the PAB begins at 0 minutes. Ten percent of iodine in the leakage becomes airborne and is released to the outside environment without credit for any retention in the PAB. The activity is released from the RWST in proportion to the air displacement rate, due to diurnal heating and cooling, and the liquid/vapor partition.

In the event of a reactor vessel head drop, immediate manual actuation of the VNCR accident mode of operation is assumed to occur. The licensee states that there is sufficient time between accident recognition and release initiation to credit manual actuation of the VNCR accident mode of operation. In addition, the magnitude of the activity released to the environment is large enough to ensure automatic operation of the VNCR accident mode of operation via a high radiation signal. Therefore, no delay in manual actuation of the VNCR accident mode of operation is taken into consideration for the RVHD CR dose analysis.

Summary

The licensee evaluated the radiological consequences resulting from the postulated RVHD using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, by confirmatory calculation. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The RVHD analysis assumption and parameters can be found in Table 3.2-8 of this SE. The NRC staff, therefore, concludes that the proposed AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

2.1.3 Control Room Habitability and Modeling

2.1.3.1 Regulatory Evaluation

Control Room and Control Room Envelope

NRC Regulatory Guide 1.196, "Control Room Habitability at Light-water Nuclear Power Reactors," Revision 0, issued May 2003[4], uses the term "control room envelope" 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.*

[4] AN ML031490611

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 Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003[6], 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.

Control Room Emergency Filtration System (CREFS)

The CREFS (the term used at Point Beach units 1 and 2 for the Control Room Envelope Emergency Ventilation System, CREEVS) 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 provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity.

The CREFS consists of one emergency air filtration unit, two emergency fans, two recirculation fans, and required ducts, valves, instrumentation, doors, barriers, and dampers necessary to establish the required flow paths and isolation boundaries that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Doors, walls, floor, roof, penetrations, and barriers also form part of the system. The CREFS is an emergency system, parts of which operate during normal unit operations.

The CREFS (Mode 5) is required to be operable to ensure that the CRE habitability limits are met following limiting DBAs. Total system failure could result in exceeding the control room operator total effective dose equivalent (TEDE) limit of 5 rem in the event of a large radioactive release. The CREFS is considered operable when the individual components necessary to filter and limit in-leakage are operable. CREFS is considered operable when:

- Both emergency fans are operable;
- Both recirculation fans are operable
- Emergency filter unit HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

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[6] AN ML031490664

- Both emergency fan control dampers are operable;

- Both isolation dampers in the kitchen area exhaust duct are operable;

- Ductwork, valves, and dampers are operable, and air circulation can be maintained; and

- CREFS is capable of being automatically and manually initiated in the emergency mode of operation.

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.

Control Room Habitability

In Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," General Design Criteria (GDC) 1, 2, 3, 4, 5, and 19 apply to CRE habitability. A summary of these GDCs follows. Facilities not licensed under the GDC from 10 CFR Part 50 are licensed under similar plant-specific design criteria, as described in the facility's licensing basis documents.

In Section 5.1, "General design Criteria", of a letter dated December 8, 2008, the licensee stated the following:

> PBNP was licensed prior to the 1971 publication of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants." As such, PBNP is not licensed to the Appendix A GDCs. PBNP Updated FSAR, Section 1.3, lists the plant-specific GDCs to which the plant was licensed. The PBNP GDCs are similar in content to the draft GDCs proposed for public comment in 1967.

Section 1.3 of Point Beach UFSAR "General Design Criteria" states the following:

> The general design criteria define the principal criteria and safety objectives for the design of this plant. A complete set of these GDCs are stated explicitly in Table 1.3-1. Table 1.3-1 also identifies other locations in this report [Point Beach UFSAR] that repeat specific GDCs.
>
> Regarding the origin of these criteria, the Atomic Energy Commission (AEC) published proposed GDCs for public comment in 1967. The Atomic Industrial Forum (AIF) reviewed these proposed criteria and recommend changes. The Point Beach GDCs documented in this FSAR are similar in content to the AIF version of the Proposed 1967 GDCs.
>
> Appendix A of 10 CFR Part 50 contains a different set of GDCs which were published in 1971 (after Point Beach construction permits were issued). Note that the GDCs found in 10 CFR Part 50 Appendix A differ both in numbering and content from the GDCs adopted herein for PBNP.

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 performed.

GDC 2, "Design Basis for Protection Against Natural Phenomena," requires that SSCs important to safety be designed to withstand the effects of earthquakes and other natural hazards.

GDC 3, "Fire Protection," requires SSCs important to safety be designed and located to minimize the effects of fires and explosions.

GDC 4, "Environmental and Dynamic Effects Design Bases," requires SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

GDC 5, "Sharing of Structures, Systems, and Components," requires that SSCs important to safety 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, the orderly shutdown and cool-down of the remaining units.

GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions, including a LOCA. Adequate radiation protection is to be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of specified values.

Prior to incorporation of TSTF-448, Revision 3, the STS requirements addressing CRE boundary operability resided only in the following CRE ventilation system specifications:

- NUREG-1430, TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)"
- NUREG-1431, TS 3.7.10, "Control Room Emergency Filtration System (CREFS)"
- NUREG-1432, TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACS)"
- NUREG-1433, TS 3.7.4, "Main Control Room Environmental Control (MCREC) System"
- NUREG-1434, TS 3.7.3, "Control Room Fresh Air (CRFA) System"

In these specifications, the surveillance requirement associated with demonstrating the operability of the CRE boundary requires verifying that one CREFS train can maintain a positive pressure relative to the areas adjacent to the CRE during the pressurization mode of operation at a makeup flow rate. 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 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 measurement to satisfy the surveillance requirement (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 these specifications were 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) 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 9900; see latest guidance in Regulatory Issue Summary (RIS) 2005-20[7]. Further stated in RIS 2005-20, "imposing administrative controls in response to 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, 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 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), which requires a facility's TS to include surveillance requirements, which it defines as "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 condition 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.

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[7] AN ML073440103

Adoption of TSTF-448, Revision 3

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) described in Regulatory Guide 1.197, 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). Adoption will better assure that the Point Beach CRE will remain habitable during normal operation and DBA conditions, and ensure acceptability from a regulatory standpoint.

2.1.3.2 Technical Evaluation

The NRC staff reviewed the proposed changes against the corresponding changes made to the STS by TSTF-448, Revision 3, which the NRC staff has found to satisfy applicable regulatory requirements, as described above in Section 2.0. The emergency operational mode of the CREFS at Point Beach units 1 and 2 pressurizes the CRE to minimize unfiltered air inleakage. The proposed changes are consistent with this design.

Editorial Changes

The licensee proposed editorial changes to TS 3.7.9, "CREFS," to establish standard terminology, such as "control room envelope (CRE)" in place of "control room," except for the plant-specific name for the CREFS (plant specific name for CREEVS), and "radiological, chemical, and smoke hazards (or challenges)" in place of various phrases to describe the hazards that CRE occupants are protected from by the CREFS. These changes improve the usability and quality of the presentation of the TS, have no impact on safety, and therefore, are acceptable.

Other Proposed Changes

The proposed amendments would strengthen CRE habitability TS requirements by changing TS 3.7.9, "CREFS" and adding a new 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 Bases also includes editorial and administrative changes to reflect applicable changes to the corresponding STS Bases, which were made to achieve more consistency among the STS NUREGs. Except for plant specific differences, all of 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 verified that differences from the STS were adequately justified on the basis of plant-specific design or retention of current licensing basis. 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 Bases is not a condition for accepting the proposed amendments. However, TS 5.5.13, "TS Bases Control Program," provides assurance that the licensee has established and will maintain the adequacy of the Bases. The proposed Bases for TS 3.7.9 refer to specific guidance in NEI 99-03, "Control Room Habitability Assessment Guidance," Revision 0, dated June 2001, which the NRC staff has formally endorsed, with exceptions, through Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated May 2003.

To accommodate the request to adopt the AST and TSTF-448, the licensee proposed specific changes to the following sections of the TSs:

1. TS 1.1, "Definitions"

2. TS 3.7.9, "CREFS", Limiting Condition for Operation (LCO) 3.7.9, including Surveillance Requirements (SR) 3.7.9.2, 3.7.9.3 and 3.7.9.6

3. TS 5.5.15, "Containment Leakage Rate Testing Program"

4. Add TS 5.5.18, "Control Room Envelope Habitability"

<u>TS 1.1, "Definitions"</u>

With respect to "Definitions," the licensee is requesting that the maximum allowable primary containment leakage rate, L_a, shall be changed from 0.40 percent to 0.2 percent of primary containment air weight per day at the peak design containment pressure (P_a).

The NRC staff's assessment has determined that this change is conservative. The change allows for less containment leakage at peak containment pressure, and does not violate the acceptance criteria of NUREG-0800, Revision 2, Section 6.2.6 (II), which states that "the minimum acceptable design containment leakage rate shall not be less than 0.1% per day." Therefore the NRC staff has determined that this change is acceptable.

<u>TS 3.7.9, "CREFS"</u>

The license proposed changing LCO 3.7.9 and SRs 3.7.9.2, 3.7.9.3 and 3.7.9.6 as follows:

From:

3.7.9 Control Room Emergency Filtration System (CREFS)

LCO 3.7.9 CREFS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4,
During movement of irradiated fuel assemblies

<u>ACTIONS</u>

CONDITION: A CREFS inoperable

REQUIRED ACTION:	A.1	Restore CREFS to OPERABLE status.
COMPLETION TIME:	7 days	
CONDITION	B	Required Action and associated Completion Time not Met.
REQUIRED ACTION:	B.1	Suspend movement of irradiated fuel assemblies.
COMPLETION TIME:	B.1	Immediately.
	AND	
	B.2	Be in MODE 3
COMPLETION TIME:	B.2	6 hours
	AND	
	B.3	Be in MODE 5.
COMPLETION TIME:	B.3	36 hours.

SURVIELLANCE REQUIREMENTS

SR 3.7.9.1	Operate the CREFS for ≥ 15 minutes.
FREQUENCY:	31 days
SR 3.7.9.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).
FREQUENCY :	In accordance with VFTP
SR 3.7.9.3	Verify each CREFS emergency make-up fan actuates on an actual or simulated actuation signal.
FREQUENCY	18 months
SR 3.7.9.4	Verify each CREFS automatic damper in the emergency mode flow path actuates to the correct position on an actual or simulated actuation signal.
FREQUENCY:	18 months

SR 3.7.9.5 Verify CREFS manual start capability and alignment.

FREQUENCY: 18 months

SR 3.7.9.6 Verify each CREFS emergency make-up fan can maintain a positive pressure of ≥0.125 inches water gauge in the control room envelope, relative to the adjacent turbine building during the emergency mode of operation at a make-up flow rate of 4950 cfm ± 10%.

FREQUENCY: 18 months

To:

3.7.9 Control Room Emergency Filtration System (CREFS)

LCO 3.7.9 CREFS shall be OPERABLE with:

a. Two control room recirculation fans,
b. Two control room emergency fans,
c. One filter train,
d. Two control room emergency fan control dampers, and
e. Two isolation dampers in the kitchen area exhaust duct.

NOTE

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

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

ACTIONS

CONDITION: A. NOTE

Separate condition entry is allowed for each component.

One control room recirculation fan inoperable,

OR

One control room emergency fan inoperable,

OR

One control room emergency fan control damper inoperable.

REQUIRED ACTION: A.1 Restore inoperable fan or damper to OPERABLE status.

COMPLETION TIME: 7 days

CONDITION: B One isolation damper in the kitchen area exhaust duct inoperable.

REQUIRED ACTION: B.1 Restore isolation damper to OPERABLE status.

COMPLETION TIME: 7 days

OR

B.2 Place and maintain the other isolation damper in the same duct in the closed position.

COMPLETION TIME: 7 days

CONDITION: C. NOTE
Separate Condition entry is allowed for each

Two control room recirculation fans inoperable.

OR

Two control room emergency fans inoperable.

OR

Two control room emergency fan control dampers inoperable.

OR

Filter train inoperable for reasons other than Condition D.

REQUIRED ACTION: C.1 Initiate actions to implement mitigating actions.

COMPLETION TIME: Immediately

AND

C.2 NOTE
Not required following completion of Required Action C.3.

Suspend movement of irradiated fuel assemblies.

COMPLETION TIME: Immediately

AND

C.3 Verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits.

COMPLETION TIME: 24 hours

AND

C.4 Restore inoperable fans, dampers, or filter train to OPERABLE status.

COMPLETION TIME: 7 days

CONDITION: D. NOTE
Separate Condition entry is allowed for each component.

Filter train inoperable due to an inoperable CRE boundary.

OR

Two isolation dampers in the kitchen exhaust duct inoperable.

REQUIRED ACTION: D.1 Initiate actions to implement mitigating actions.

COMPLETION TIME: Immediately

AND

D.2 NOTE
Not required following completion of Required Action D.3

Suspend movement of irradiated fuel assemblies.

COMPLETION TIME: Immediately

AND

D.3 Verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.

COMPLETION TIME: 24 hours

	AND	
	D.4	Restore CRE boundary to OPERABLE status.
COMPLETION TIME:	90 days	
CONDITION:	E.	Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, 3, or 4 not met during movement of irradiated fuel assemblies.
REQUIRED ACTION:	E.1	Suspend movement of irradiated fuel assemblies.
COMPLETION TIME:	Immediately	
	AND	
	E.2	Be in MODE 3
COMPLETION TIME:	6 hours	
	AND	
	E.3	Be in MODE 5
COMPLETION TIME:	36 hours	

SURVIELLANCE REQUIREMENTS

SR 3.7.9.1	Operate the CREFS for ≥ 15 minutes.
FREQUENCY:	31 days
SR 3.7.9.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).
FREQUENCY:	In accordance with the VFTP
SR 3.7.9.3	Verify each CREFS emergency and recirculation fan actuates on an actual or simulated actuation signal.
FREQUENCY:	18 months
SR 3.7.9.4	Verify each CREFS automatic damper in the emergency mode flow path actuates to the correct position on an actual or simulated actuation signal.

FREQUENCY:	18 months
SR 3.7.9.5	Verify CREFS manual start capability and alignment.
FREQUENCY:	18 months
SR 3.7.9.6	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.
FREQUENCY:	In accordance with the Control Room Envelope Habitability Program.

In addition to the request to adopt TSTF-448, the licensee stated that these TS changes are being proposed as a result of AST assumptions crediting the CREFS and to address mitigating actions that will be implemented in the event that the CREFS become inoperable. To support this position the licensee provided the following in its April 20, 2010, letter:

> NextEra has performed a control room dose calculation for limiting AST radiological accidents without credit for the CREFS and including use of mitigating actions. The following are assumption applied to the calculations:
>
> - 1000 cubic feet per minute (cfm) filtered recirculation by a mitigating filtration unit and filter efficiencies of 99% for particulates and 95% for elemental and organic iodine. This filtration is assumed to be started one hour after the initiation of the accident.
> - Potassium iodide (KI) ingestion to reduce the dose to the thyroid from inhalation of iodines by a factor of 10.
> - No other changes are made to the release models in the analyses submitted with license amendment request 241.

The licensee stated that the results of this AST calculation meets the 5 rem total effective dose equivalent (TEDE), and represents the control room dose for the worst-case large break LOCA based on the use of a mitigating filtration unit and administration of KI without credit for the CREFS.

The licensee provided the following information as the design criteria for the mitigating filtration unit(s):

- A 1000 cfm high efficiency particulate (HEPA)/charcoal filter system needs to be installed outside the CR.
- Ductwork needs to be built between the CR wall and the filter system, and needs to conform to CR security requirements.
- Power for the system will be emergency diesel generator-backed power.

- Bubble tight manual dampers to assure maintaining the normal CRE will be provided to isolate the system when it is not in use.

- The system will be classified as augmented quality.

- The system will be seismically supported.

- System testing will be in accordance with the Ventilation Filter Test Program.

The licensee proposed to establish new action requirements in TS 3.7.9, "CREFS" for an inoperable CRE boundary. Currently, if the CREFS is determined to be inoperable due to an inoperable CRE boundary, existing Action A would apply and require restoring the CREFS (and the CRE boundary) to operable status in 7 days. The existing Action is more restrictive than would be appropriate in situations for which CRE occupant implementation of compensatory measures or mitigating actions would temporarily afford adequate CRE occupant protection from postulated airborne hazards. To account for such situations, the licensee proposed to revise the action requirements to add a Condition D "Filter train inoperable due to inoperable CRE boundary or two isolation dampers in the kitchen exhaust duct inoperable", in MODE 1, 2, 3 or 4. Condition D also includes a note stating a separate condition entry is allowed for each component. New Required Action D.4 would allow 90 days to restore the CRE boundary (and consequently, the affected CREFS) to operable status. D.1 initiates actions to ensure mitigating actions are immediately implemented, D.2 ensures the suspension of movement of irradiated fuel assemblies and D.3 verifies that mitigating actions are implemented within 24 hours to ensure 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.

The 24-hour Completion Time of new Required Action D.3 is reasonable based on the low probability of a DBA occurring during this time period and the use of mitigating actions. The 90-day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that the 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 of new Required Action D.4 is a reasonable time to diagnose, plan and possibly repair, and test most anticipated problems with the CRE boundary. Therefore, proposed Actions D1, D.2, D.3 and D.4 are acceptable.

To distinguish new Condition D from the existing Condition for CREFS inoperable, Condition A is revised, and new Conditions A and C include a NOTE stating a separate condition entry is allowed for each component. Conditions A and B allows redundant active CREFS components to be inoperable for 7 days. Condition C allows the CREFS to be inoperable for 7 days for reasons other than Condition D. The changes to existing Condition A and the addition of Conditions B and C are less restrictive because these Conditions will no longer apply in the event the CREFS is inoperable due to an inoperable CRE boundary during unit operation in Mode 1, 2, 3 or 4. This is acceptable because the new Action C establishes adequate remedial measures in this condition.

The licensee also proposed to modify the CREFS LCO by adding a NOTE allowing the CRE boundary to be opened intermittently under administrative controls. As stated in the LCO Bases, this NOTE "only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated." The allowance of this NOTE is acceptable because the administrative controls will ensure that the opening will be quickly sealed to maintain the validity of the licensing basis analyses of DBA consequences.

In a supplement to the LAR dated April 6, 2011, the licensee provided administrative changes to TS 3.7.9, and TS 5.5.18. Specifically, the licensee proposed to delete TS 3.7.9, NOTES C.2 and D.2, that allowed for continued movement of irradiated fuel assemblies once the mitigating actions of TS 3.7.9, C.3 and D.3, respectively, had been completed. The licensee also modified TS 3.7.9, D.3, to include that chemical exposures will not exceed limits by proposing the following:

> Verify mitigating actions ensure CRE occupant radiological and chemical exposures will not exceed limits, and CRE occupants are protected from smoke hazards.

The licensee also modified TS 5.5.18.d, to provide further clarification that the data will be trended on an 18-month interval, and that this information will be used to evaluate and ensure any CRE boundary degradation is identified. Enclosure 1 of the April 6, 2011, letter provides a revised markup of TS 3.7.9 and TS 5.5.18 to address these changes. The changes resolved NRC staff concerns related to the licensee's approach in addressing the guidance contained in TSTF-448, Revision 3. The NRC staff determined that suspension of irradiated fuel movement is conservative. Therefore, the NRC staff considers the licensee's proposed changes to be acceptable.

In the emergency make-up mode of operation, the CREFS isolates unfiltered ventilation air supply intakes, filters the emergency ventilation air supply to the CRE, and pressurizes the CRE to minimize unfiltered air inleakage past the CRE boundary. The licensee proposed to delete the CRE pressurization surveillance requirement (SR). This SR requires verifying that the CREFS, operating in the emergency make-up mode, can maintain a pressure of 0.125 inches water gauge, relative to the adjacent turbine building during the pressurization mode of operation at a makeup flow rate of 4950 cfm. The deletion of this SR is proposed because measurements of unfiltered air leakage into the CRE at numerous reactor facilities demonstrated that a basic assumption of this SR, an essentially leak-tight CRE boundary, was incorrect for most facilities. Hence, meeting this SR by achieving the required CRE pressure is not necessarily a conclusive indication of CRE boundary leak tightness, i.e., CRE boundary operability. In responses to GL 2003-01 dated August 11, 2003[8], and May 1, 2007[9], the

[8] AN ML032580984

licensee reported that it had determined that the PBNP Units 1 and 2 CRE pressurization surveillance, SR 3.7.9.6, was inadequate to demonstrate the operability of the CRE boundary and proposed to replace it with an inleakage measurement SR and a CRE Habitability Program in TS Section 5.5, in accordance with the approved version of TSTF-448. Based on the adoption of TSTF-448, Revision 3, the licensee's proposal to delete SR 3.7.9.6 is acceptable.

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 Regulatory Guide 1.197, Revision 0. 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. Therefore, the proposed CRE inleakage measurement SR is acceptable.

TS 5.5.15, Containment Leakage Rate Testing Program

In TS 5.5.15, "Containment Leakage Rate Testing Program," the licensee is revising TS 5.5.15.c in order to change the maximum allowable containment leakage rate, L_a at P_a from 0.4 percent to 0.2 percent of containment air weight per day. This is an editorial change and is acceptable because it is being changed to be made consistent with the change in definition that the NRC staff reviewed and found acceptable in Section 1.1, "Definitions".

TS 5.5.18, CRE Habitability Program

This proposed administrative controls program TS is consistent with the model program TS in TSTF-448, Revision 3. In combination with SR 3.7.9.6, 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 total effective dose equivalent (TEDE) for the duration of the accident.

A CRE Habitability Program TS 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 2.1 of this safety evaluation. Establishing what is meant by the CRE and the CRE boundary will preclude ambiguity in the implementation of the program.

[9] AN ML071210471

- 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 Regulatory Guide 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 Regulatory Guide 1.197, Revision 0, 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 Regulatory Guide 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, at a frequency of 18 months.

  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 3.7.9, 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.

Implementation of New Surveillance and Assessment Requirements by the Licensee

The licensee has proposed license conditions regarding the initial performance of the new surveillance and assessment requirements. The new license conditions adopted the conditions in Section 2.3 of the model application published in the *Federal Register* on January 17, 2007 (72 FR 2022). Plant specific changes were made to these proposed license conditions. The proposed plant specific license conditions are consistent with the model application, and are acceptable. A summary of the license conditions is provided in Attachment 2.

2.1.4 <u>Additional Technical Specification Changes</u>

TS 3.4.16, RCS Specific Activity

The licensee proposed to revise SR 3.4.16.2 to reduce the allowable specific activity for the reactor coolant from a DE I-131 $\leq$ 0.8 μCi/gm to $\leq$ 0.5 μCi/gm as follows:

- Current SR 3.4.16.2 states, "Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 0.8 μCi/gm."

- Revised SR 3.4.16.2 states, "Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 0.5 μCi/gm."

The AST reanalysis has determined that the proposed value provides acceptable radiological consequences for DBAs. The change is also conservative, in that it provides a more restrictive limit for the specific activity in the RCS. Based on the above, the NRC staff concludes that this TS change is acceptable.

TS 3.7.13, Secondary Specific Activity

The licensee proposes to revise LCO 3.7.13 and SR 3.7.13.1 to reduce the allowable specific activity for the secondary coolant from $\leq$ 1.00 μCi/gm to $\leq$ 0.1 μCi/gm DE I-131 as follows:

- Current LCO 3.7.13 states, "The specific activity of the secondary coolant shall be $\leq$ 1.00 μCi/gm DOSE EQUIVALENT I-131."

- Revised LCO 3.7.13 would state, "The specific activity of the secondary coolant shall be $\leq$ 0.1 μCi/gm DOSE EQUIVALENT I-131," and

- Current SR 3.7.13.1 states, "Verify the specific activity of the secondary coolant is $\leq$ 1.00 μCi/gm DOSE EQUIVALENT I-131."

- Revised SR 3.4.13.1 would state, "Verify the specific activity of the secondary coolant is $\leq$ 0.1 μCi/gm DOSE EQUIVALENT I-131."

The AST re-analysis has determined that the proposed values provide acceptable radiological consequences for DBAs. The changes are also conservative, in that they provide a more restrictive limit for specific activity in the SG. Based on the above, the NRC staff concludes that these TS changes are acceptable.

The AST re-analysis has determined that the new proposed TS values provide acceptable radiological consequences of DBAs. In addition, this change is a conservative change in that it provides a more restrictive limit for the specific activity in the secondary coolant. Based on the above, the NRC staff concludes that this TS change is acceptable.

2.1.5 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST at PBNP Units 1 and 2. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The licensee's limiting calculated DBA dose results are given in Table 3.2-1. The NRC staff finds with reasonable assurance that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that PBNP Units 1 and 2 as modified by this license amendment, 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. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the PBNP Units 1 and 2 design basis is superseded by the AST proposed by the licensee. The previous offsite and CR 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 fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the PBNP Units 1 and 2 design basis, and modified by the present amendment.

2.2. Materials and Chemical Engineering

2.2.1 Regulatory Evaluation

Implementation of the AST by the licensee required re-analyzing several DBAs using new source terms. The licensee performed these tasks by following the requirements of 10 CFR 50.67. The licensee also applied for a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." An acceptable accident source term is a permissible amount of radioactive material which could be released to the containment from the damaged core following an accident. Because of improved understanding of the mechanisms of the release of radioactivity, the current accident source term could be replaced by a less restrictive AST. However, this replacement is subject to performing a successful re-evaluation of the major DBAs. The guidance for implementation of an AST is provided in RG 1.183.

The NRC staff reviewed the portion of the amendment dealing with the licensee's analysis for maintaining sump pool pH greater than or equal to 7 for 30 days following a LOCA. As previously discussed in Section 2.1.2.2.1, the term pH (from potential of Hydrogen) is the logarithm of the reciprocal of hydrogen-ion concentration in gram atoms per liter and provides a measure on a scale from 0 to 14 of the acidity or alkalinity of a solution. As stated in RG 1.183, maintaining a pH basic (i.e., greater than 7) will minimize re-evolution of iodine from the sump pool.

2.2.2 <u>Technical Evaluation</u>

As specified in NUREG-1465, iodine released from the damaged core to the containment after a LOCA is composed of 95 percent CsI which is a highly ionized salt and soluble in water. Iodine in this form does not present any significant radiological concerns since it stays dissolved in the sump water and does not enter the containment atmosphere. However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is relatively insoluble in water and can therefore be released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is one of the more important. Maintaining pH basic in the sump water will ensure that this conversion will be minimized. The licensee used the method described in NUREG/CR-5950, "Iodine Evolution and pH Control" for calculating generation of this elemental iodine. Its calculations have indicated that at the higher sump water pH, less iodides are converted into elemental form and at pH 7 or higher, elemental iodine generated from this source becomes insignificant relative to the elemental iodine release directly to the containment from the damage core. The pH of the sump water in PBNP is controlled by the sodium hydroxide buffer which is formed by the addition of sodium hydroxide via the containment spray from the spray additive tank to the boric acid dissolved in the sump water after a LOCA.
After a LOCA, several acids are either generated or are added to the containment. Relative amounts of these acids and that of sodium hydroxide determine the value of pH reached by the containment sump water.

After a LOCA, boric acid from the reactor coolant system, accumulators, and refueling water storage tank is discharged into the sump. The licensee used a minimum and maximum boron concentration in each of those systems to perform the minimum and maximum pH calculations. Also, the value of pH will be continuously decreasing due to formation of hydrochloric and nitric acids in the containment. Hydrochloric acid (HCl) is formed from decomposition of chlorinated polymer cable insulation from radiation. The licensee conservatively overestimated the cable insulation weight. The licensee used a generation rate from NUREG-5950 of 4.6×10^{-4} moles of HCl per pound of insulation per Mrad. Nitric acid is formed in the containment by irradiation of water and air. The amount of nitric acid produced is proportional to the time-integrated dose rate for gamma and beta radiation. Both are strong acids and will contribute to lowering sump pH. In order to neutralize the boric, hydrochloric and nitric acids, the licensee chose to buffer the sump pool water by using a sodium hydroxide buffer. Such buffering action is intended to maintain basic pH in the sump pool despite the presence of the acids. The licensee has calculated that by adding 164.2 cubic feet (ft^3) of a 30 percent sodium hydroxide solution, at a reference temperature of 77 degrees Fahrenheit (°F), from the spray additive tank to the sump pool with an injection time between 69.11 and 149.55 minutes, it will maintain a basic pH in the sump water.

The NRC staff has independently verified the licensee's calculations and finds that by using sodium hydroxide as a buffer in the quantity specified, the pH of the sump water will remain above 7 for 30 days post-LOCA.

2.2.3 Conclusion

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of sump water and the corresponding fraction of the dissolved iodine in the sump water that is converted into the elemental form. The calculations were made for the 30 day period following a LOCA. The NRC staff performed independent calculations to verify the licensee's calculation. From the results of these calculations, the NRC staff concluded that although the value of pH varied with time it never dropped below 7.6. Maintaining pH above 7 resulted in a negligible fraction of the dissolved iodine being converted into elemental form and a low release of radioactive iodine to the environment. Based on these considerations, the NRC staff finds the licensee's proposal acceptable.

2.3 Electrical Engineering

2.3.1 Regulatory Evaluation

The following NRC requirements and guidance documents are applicable to the NRC staff's review of the LAR:

10 CFR Appendix A of Part 50, GDC 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies. GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that the safety related electrical equipment which are relied upon to remain functional during and following design basis events be qualified for accident

(harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

10 CFR 50.67, "Accident Source Term," provides an optional provision for licensees to revise the AST used in design basis radiological analyses.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This RG states that the licensees may use either the AST or the Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," assumptions for performing the required environmental qualification (EQ) analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID 14844) on EQ doses.

RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

2.3.2 Technical Evaluation

The licensee is proposing using an AST to determine the offsite and control room doses, resulting from a LOCA. The licensee stated that implementation of the AST for the LOCA will require adding a control room recirculation fan and control room charcoal filter fan to the initial loading of each train of the emergency diesel generators (EDGs).
Upon reviewing the license amendment request, the NRC staff requested additional information on any changes to the PBNP EDGs loading sequence. The licensee, in its response dated September 4, 2009, provided the EDG loading sequence and stated that the CS pumps and the VNPAB fans during the ECCS recirculation phase will not affect initial EDG load sequencing since these loads are added manually during a LOCA coincident with a LOOP.

The NRC staff further questioned whether any loads were being added to the PBNP EDGs and if so, how the loads being added would affect the capability and capacity of the EDGs. The licensee in its September 4, 2009, letter provided evaluations of the loading for CREFS and VNPAB. The licensee calculated the AST EDG loading for the Train A EDGs to be 2,779 kilowatts (kW) and concluded that the Train A EDGs will continue to operate within their 2,000-hour rating of 2,850 kW for the worst case DBA EDG electrical loading condition. The licensee calculated the AST Train B EDG loading to be 2,877 kW as stated in the October 2, 2009, letter. The licensee concluded in its September 4, 2009, letter that the Train B EDGs will continue to operate within the 200-hour rating of 2,951 kW for up to 24 hours and then remain within the 2,000-hour rating of 2,848 kW for the most limiting DBA EDG electrical loading condition. The licensee further stated that if LOCA occurs coincident with a LOOP, the operator will manually start the VNPAB system to ensure that the PAB vent stack is the source of the release associated with ECCS equipment leakage during the recirculation phase of event.

The PBNP FSAR states that the Train A EDGs are rated at 2,850 kW for 2,000 hours, 2,963 kW for 200 hours, 3,000 kW for 4 hours and 3,050 kW for a 30 minute period. The Train B EDGs are rated at 2,848 kW for 2,000 hours, 2,951 kW for 200 hours and 2,987 kW for 4 hours. The NRC staff requested additional details describing the testing of the EDGs to ensure the EDGs are capable of supporting the AST loading. The licensee stated in its October 2, 2009, letter that the current margin load testing for the Train A EDGs is performed over a range of 2,835 to 2,970 kW. Thus, the testing for the Train A EDGs bounds the AST loading of 2,779 kW and the Train A EDGs are capable of supporting the AST loading. The licensee stated that the current margin load testing for the Train B EDGs is over a range of 2,877 to 2,950 kW, which bounds the AST loading of 2,877 kW. A new license condition will require the load testing to be performed in a range that bounds the AST loading for Train B EDGs. Based on the above information, to ensure adequate testing of the EDGs, the license proposed the following license condition be incorporated in the Facility Operating License for PBNP:

> NextEra Energy Point Beach, LLC will perform Train B EDG load testing over a range of 2877 to 2950 kW at rated power factor following the implementation of LAR-241 in accordance with the testing schedule.

In its July 8 and August 12, 2010, responses to NRC staff requests for additional information (RAI) provided specifics regarding the power requirements for the mitigating filtration unit(s), and address loading requirements on the EDGs, the licensee stated the following:

"The mitigating filtration unit(s) will only be placed in service when the CREFS is out of service. The electrical power requirements for the mitigating filtration unit(s) are equivalent to the installed CREFS fans. Thus, EDG loading would not be affected if the mitigating filtration unit(s) were placed into service following a loss of offsite power."

The NRC staff requested the following additional information as a result of its review of the above licensee response:

a) Describe how the mitigating filtration unit(s) will be electrically connected and loaded onto the diesel generator (i.e., sequence of loading). Also describe what prevents the CREFS and mitigating filtration unit(s) from concurrently being loaded onto the diesel generator (alternatively: discuss how the diesel generator has sufficient capacity to support loading of both the CREFS and mitigating filtration unit(s)).

b) Describe how the mitigating filtration unit(s) will be electrically isolated and separated from the Class 1E system.

c) Describe how you determined that the electrical power requirements for the mitigating filtration unit(s) are equivalent to the installed CREFS fans (e.g., provide the ratings for the CREFS and mitigating filtration unit(s)).

In its response to question (a), the licensee stated that the mitigating filtration unit will be connected to the EDG via the same power source as the CREFS filter fans W-14A or W-14B via a manually-operated transfer switch. Since the mitigation filtration unit is not normally an automatically sequenced load, the EDG "sequence of loading" is not normally pertinent. The licensee further noted that manual actuation of the mitigating filtration unit will not take place until the automatic load sequence has been completed. The CREFS filter fans and mitigating filtration unit will be prevented from concurrent loading onto the EDG via the manual transfer switch, which will only allow one fan per EDG to operate.

In its response to question (b), the licensee stated that the mitigating filtration unit will be electrically isolated and separated from the Class 1E system via the motor control center (MCC) breaker (feeding either W-14A or W-14B, depending on which MCC is chosen) that serves as the point of electrical isolation between the Class 1E side (the MCC) and the non Class 1E side (downstream of the breaker).

In its response to question (c), the licensee stated that the electrical power requirements for the mitigating filtration unit are intended to be equivalent to the W-14A/B motors. The licensee's mitigating filtration unit specification requires that an equivalent motor to the W-14A/B motors will be procured for the mitigating filtration unit (equivalent horse power, efficiency, power factor, and voltage requirements).

The licensee proposed the following license condition to the Facility Operating License in its February 10, 2011, letter[10]:

> NextEra Energy Point Beach, LLC shall procure the CREFS mitigating filtration unit with electrical power requirements equivalent to the CREFS filter fan motors (i.e., equivalent horse power, efficiency, power factor, and voltage requirements).

Based on this information, the NRC staff finds that the PBNP EDGs have adequate capacity to support operation of an equivalently sized mitigating filtration unit.

Therefore, based on the above information and the license conditions, the NRC staff finds the response acceptable.

Since the VNPAB system is a non-safety related system, the NRC staff requested additional information regarding how the VNPAB system will be electrically separated from the safety-related system. Specifically, the NRC staff requested information on how a fault on the non-Class 1E electrical circuit would not propagate to the Class 1E circuit, affecting independence and redundancy of these systems. The licensee stated in its September 4, 2009, letter, that VNPAB fans are powered from independent and redundant safety related electrical sources and are aligned to the EDGs during LOOP conditions. The VNPAB fans are isolated from the safety-related Class 1E system by safety-related circuit breakers, and VNPAB load circuit

[10] AN ML110420103

breakers will isolate a fault on the non-Class 1E circuit and will not propagate to the Class 1E system. The VNPAB fans are supplied power through MCCs and maintained such that the redundant channels are not intermixed within the same raceway. Based on the above information, the NRC staff finds that the VNPAB system has sufficient redundancy and independence.

The NRC staff requested the licensee to describe how the VNPAB meets the single failure criterion. The licensee stated in its September 4, 2009, letter that modifications to CREFS and VNPAB in support of the AST LAR ensure that no single active or passive failure of an electrical or control component will prevent the system from performing its required function. Adequate redundancy is provided for electrical power supplies and controls. The licensee subsequently determined that additional clarification was required related to statements made in the September 4, 2009, letter. In a March 11, 2011, letter, NextEra provided clarification of the passive electrical failure assumptions on electrical conductors and cables for the CREFS and VNPAB systems.

The original PNPB VNPAB and CREFS systems were designed and installed as non-safety related systems. Since these systems are non-safety related, they were not required to be designed for active or passive single failures. Although not part of the current design basis of the VNPAB and CREFS systems, the licensee has improved these systems to allow for redundancy of active system components, seismic upgrades, and quality classification upgrades. In its March 11, 2011, letter, the licensee clarified that the electrical passive failures addressed in the modified design of the CREFS and VNPAB systems are non-mechanistic and do not include failure of one conductor, cable, or device causing a failure of another conductor, cable, or device in the same location or raceway.

The licensee noted that the new Mode 5 CREFS and VNPAB design considers relay failures; failures of contacts to change state; shorting of contacts that change state; and shorting of relays, solenoids, or starter coils that could cause a damper to change to an undesirable state or prevent starting of a fan.

For existing installed plant cables, components, and control panel items, the licensee did not implement design changes to improve separation of conductors, components, or internal control panel items associated with the CREFS and VNPAB. For the CREFS and VNPAB, the licensee did not consider shorts for existing plant cables and conductors that were not replaced. However, for new cables and conductors associated with the CREFS, the licensee considered failures of the cable to short line-to-ground or line-to-line, or open circuits on the cables and conductors. No new control cables or conductors were installed for VNPAB.

Considering that the original VNPAB and CREFS systems were designated as non-safety related systems and therefore not required to meet the single failure criterion, the staff finds that the modifications (i.e., redundancy of active system components, seismic upgrades, and quality classification upgrades) enhance the previous PNPB design by fulfilling active single failure criteria while also satisfying certain passive single failures. Based on this information, the NRC staff finds the licensee's improvements to the VNPAB and CREFS systems to be conservative in that they increase the redundancy and defense-in-depth of the PBNP to support the proposed AST.

The NRC staff requested additional information on how the operators would be notified in the event that the CREFS or VNPAB would become inoperable. In its September 4, 2009 letter, the licensee stated that the CR has the following alarms: low flow air condition for CR HVAC system and low flow air condition for the VNPAB system. Based on the above information, the NRC staff finds that there is adequate indication in the CR to show when the CREFS or VNPAB would become inoperable.

The NRC staff requested that the licensee provide a list and descriptions of components added to its 10 CFR 50.49 program due to the AST and additionally confirm that these components are qualified for the environmental conditions they are expected to be exposed to. The licensee in its September 4, 2009, letter provided a list of components to be added to its 10 CFR 50.49 program as a result of the AST adoption. These components were cables, motors, or motor terminations. The licensee further stated that the addition of these components to the 10 CFR 50.49 program is being implemented as part of the AST modifications and that these components are qualified for the environmental conditions. The NRC staff requested the licensee to specify whether these components were existing cables, motors, or motor terminations. The licensee stated in its October 2, 2009, letter, that the components added to the 10 CFR 50.49 program will be a combination of replacement and existing cables, motors, and motor terminations. The licensee provided a table indicating the component, whether it was existing or a replacement, the required qualification Rads, and the component qualification Rads. As the component qualification Rads is greater than the required qualification Rads for all the components, the NRC staff finds that the cables, motors, and motor terminations are qualified for the environmental conditions. Based on the above information, the NRC staff finds that the components being added to the PBNP 10 CFR 50.49 program are qualified for the environmental conditions expected resulting from the AST.

The licensee proposed to utilize the CS system during the sump recirculation phase and to modify operation of the CS and RHR systems to provide throttling capability of CS and RHR during the ECCS recirculation phase. The NRC staff requested the licensee to discuss the impact of the above proposed modifications on EQ. The licensee in its September 4, 2009, letter stated that the pressure and temperature LOCA profiles on record remain bounding at AST conditions and that there is no impact on the EQ program. Based on the above information, the NRC staff finds the response acceptable.

The licensee stated that a boron concentration of 3500 parts per million (ppm) was assumed in the RWST. The NRC staff requested the licensee to provide the impact of this chemical spray composition on EQ components. The licensee in its September 4, 2009, letter stated that its analysis concluded that the minimum sump pH at 3500 ppm is between 7 and 8, which is within the EQ evaluated pH range of 7.0 - 10.5. Therefore, there is no effect on EQ components located inside containment. Based on the above information, the NRC staff finds the response acceptable.

The NRC staff also reviewed the EQ portion of the LAR. The licensee used the methodology contained in TID 14844 to determine the radiation doses in the existing EQ analyses. As mentioned previously, the use of this methodology is consistent with the guidance contained in RG 1.183.

2.3.3 Conclusion

Based on the above evaluation, the NRC staff finds the proposed revision to the licensing basis provides reasonable assurance that an AST can be implemented at the PBNP. The NRC staff also concludes that the proposed changes are in accordance with 10 CFR 50.49, 10 CFR 50.65, 10 CFR 50.67, and the requirements of GDCs 17 and 18 and consistent with the guidance in RG 1.183 and RG 1.75. Therefore, the NRC staff finds the proposed changes acceptable.

2.4 Mechanical and Civil Engineering

2.4.1 Regulatory Evaluation

Pursuant to 10 CFR 50.67, a licensee may revise its current accident source term by re-evaluating the consequences of DBAs with the AST. Appendix A to 10 CFR 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," requires that SSCs necessary to assure the capability of the plant to mitigate the consequences of accidents which could result in exposures comparable to the guideline exposures provided in 10 CFR 100, "Reactor Site Criteria," be designed to remain functional during and after a safe-shutdown earthquake (SSE). The NRC staff's review in the area of mechanical and civil engineering focuses primarily on the structural integrity, including seismic adequacy, of SSCs credited in the implementation of the AST at PBNP.

The PBNP was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50. As such, PBNP is not licensed to the GDC of Appendix A. Table 1.3-1 of the PBNP UFSAR provides a listing and description of the plant-specific GDC to which the plant was licensed. The PBNP GDCs are similar in content to the draft GDCs proposed by the Atomic Energy Commission for public comment in 1967. As such, GDC 1 and GDC 2 for the PBNP are comparable to GDC 1 and 2 provided in 10 CFR 50, Appendix A.

The NRC staff's evaluation considered 10 CFR 50.55a, and PBNP GDCs 1 and 2. The NRC staff's review focused on verifying that the licensee provided reasonable assurance of the structural and functional integrity of the affected SSCs under postulated accident conditions, as analyzed with the implementation of an AST. The acceptance criterion are based on the continued conformance with the requirements of 10 CFR 50.55a, and PBNP GDC 1, as they relate to SSCs being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The acceptance criterion are also based on PBNP GDC 2, as it relates to SSCs important to safety being designed to withstand the effects the results from the loadings imposed on these SSCs due to the occurrence of extraordinary natural phenomena, such as earthquakes, combined with the effects of accident conditions.

The guidance associated with the implementation of an AST is provided in RG 1.183. With respect to the mechanical and civil engineering aspects of the AST implementation, RG 1.183 discusses that licensees must evaluate non-radiological impacts on a facility which are a consequence of the implementation of an AST methodology. For this particular AST LAR, the licensee is requesting to implement a full-scope AST as described in RG 1.183. Full-scope AST implementation refers to the licensee's request to re-calculate the dose consequences of selected DBAs to address all five characteristics of the AST (i.e., the composition, magnitude, chemical and physical forms of the radioactive material and the timing of the material's release). Additional guidance for the review can also be found in Section 15.0.1 of the SRP.

Finally, the NRC recently issued similar AST implementation license amendments for the St. Lucie Nuclear Plant, Unit 1, on November 26, 2008[11]; the Edwin I. Hatch Nuclear Plant, Units 1 and 2, on August 28, 2008[12]; the South Texas Project, Units 1 and 2, on March 6, 2008[13]; and the Salem Nuclear Generation Station, Units 1 and 2, on February 17, 2006[14].

2.4.2 Technical Evaluation

The NRC staff review of the licensee's application for the full-scope AST implementation primarily focused on the structural integrity, including the seismic adequacy, of SSCs that are credited in the proposed AST for the transport and removal mechanisms related to the source term. Specifically, the structural integrity of the VNPAB and CREFS, or portions of these HVAC SSCs, that are being credited in the licensee's AST application were reviewed to determine whether they would continue to satisfy the aforementioned regulatory requirements following the proposed AST implementation.

2.4.2.1 HVAC Systems

Overview of Seismic Evaluation of HVAC SSCs Credited for AST Implementation

In the course of its review, the NRC staff expressed concerns regarding the seismic adequacy of the HVAC SSCs credited for the proposed AST. In its September 10, 2009, response to concerns raised by the NRC staff, the licensee committed to evaluate the seismic adequacy of the CREFS and VNPAB credited in the AST analyses.

By letter dated November 20, 2009, the licensee supplemented its AST LAR with Stevenson and Associates (S&A) Report No. 09Q0839-R-001, Revision 0, "VNPAB and CREFS Seismic Verification"[15]. The licensee further supplemented its application with additional information by letter dated July 29, 2010, which provided Revision 1 of the S&A Report and responses to the NRC staff's RAIs. The S&A Report No. 09Q0839-R-001, Revision 1, dated July 15, 2010, and the licensee's responses to the NRC staff's RAIs, document the seismic verification of the Units 1 and 2 CREFS and VNPAB credited in the PBNP AST analyses. The licensee subsequently determined the need to further supplement its AST application with additional information by letters dated April 20, 2010, and July 8, 2010. In these supplements, the licensee introduced the addition of mitigating filtration units which will be placed in service when the CREFS is out of service. The licensee has proposed the following license condition for the addition of the mitigating filtration units:

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[11] AN ML082682060
[12] AN ML081770075
[13] AN ML080160013
[14] AN ML060040322
[15] AN ML093310309

> NextEra Energy Point Beach, LLC will install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) will be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992 [GIP-2], and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.

Installation and operation of this modification will be completed no later than the Unit 2 (2011) refueling outage.

The NRC staff notes that GIP-2 and its associated NRC SER (SSER No. 2) are included in the PBNP Units 1 and 2 licensing basis. Section A5.6 of the PBNP UFSAR states that "Modified, new, or replacement equipment classified as Seismic Class I may be seismically designed and verified (after installation) for seismic adequacy using... [GIP-2 and SSER No. 2]." EPRI Report No. 1014608 has not been approved for use by the NRC staff. However, as shown below, the NRC staff has found the EPRI Report No. 1014608 methodology acceptable in demonstrating the seismic adequacy of existing non-seismically analyzed ventilation systems that are credited in the proposed PBNP AST. It should be noted that the NRC staff's acceptance of utilizing the seismic experience-based methodology of the above referenced EPRI report is limited to its application for the proposed PBNP AST LAR and it is not an endorsement for its use in other applications at PBNP Unit 1 and Unit 2. The NRC staff finds the licensee's proposed methodology of evaluating the structural integrity, which includes verification of the seismic adequacy, of the new, additional mitigating filtration units, which have been proposed as part of the PBNP AST LAR, acceptable. This acceptance is based on the fact that the evaluation will be performed in accordance with the current PBNP licensing basis (i.e., GIP-2 and SSER No. 2) and with the methodology approved by this SER for the CREFS seismic verification.

The NRC staff's approval of this amendment is contingent upon a provision of confirmation of the design completion of the proposed filtration units. This confirmation shall indicate the method of evaluation used for the added filtration unit and its components and the number of filtration units required. This confirmation shall also include the anchorage factor of safety used for any modified, replacement or additional HVAC SSCs credited in the proposed AST and verify that it is in accordance with the UFSAR Section A5.6 which states that:

> "For new installations and newly designed anchorages in modifications or replacements, the GIP-2 criteria and procedures may also be applied, except that the factor of safety currently recommended for new nuclear power plants in determining the allowable anchorage loads shall be met."

The seismic verification of the HVAC SSCs was performed by a two member seismic review team (SRT). The NRC staff reviewed the SRT's qualifications presented in the seismic verification report S&A Report No. 09Q0839-R-001, Revision 1. The NRC staff found that the SRT members meet the requirements for Seismic Capability Engineers as defined in Section 2 of GIP-2. Therefore, the NRC staff considers the makeup of the SRT acceptable.

2.4.2.2 VNPAB and CREFS

The licensee credited the CREFS and the VNPAB as part of the implementation of the proposed AST at PBNP. The S&A Report No. 09Q0839-R-001, Revision 1, supplemented by the peer review report contained as an attachment to the report, and the licensee's July 29, 2010, letter in response to the NRC staff's RAIs, documents the seismic adequacy verification of the PBNP Units 1 and 2 CREFS and VNPAB credited HVAC systems in the PBNP AST analyses.

The licensee's evaluation methodology used to demonstrate the seismic ruggedness of the CREFS and VNPAB HVAC systems follows the guidelines found in the SQUG GIP-2

SQUG GIP-2 and its associated NRC SER (SSER No. 2), and NRC GL 87-02, Supplement 1, for fans, motors and heat exchangers, and the December 2006 EPRI Topical Report (TR) 1014608 (Reference 7) for ducts, duct supports, dampers, filter units and plenums. The SQUG GIP-2 is in the PBNP current licensing basis and, therefore, its use is acceptable. The EPRI guidelines are based on seismic experience data. The methodology is similar to GIP-2 and its associated SER in that it relies on the evaluation of seismic failure mechanisms for duct and damper systems from seismic experience data and includes methods to screen and identify the ductwork seismic vulnerabilities and weaknesses. Similarly with GIP-2, the EPRI methodology includes performing in-plant walkdowns and reviews to determine the seismic adequacy of the duct systems and supports, selecting representative duct and support analytical review samples and performing analytical reviews, and resolving outliers. The EPRI guidelines also include training and experience qualification requirements for individuals performing the seismic reviews similar to those of GIP-2. The EPRI TR 1014608 has not been submitted for NRC approval. However, based on its review, the NRC staff finds the EPRI TR 1014608 approach of utilizing the earthquake experience-based methodology, as supplemented by the peer review comments and by plant-specific seismic ruggedness evaluations, as an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed ventilation systems that are credited in the proposed PBNP AST. The NRC staff's acceptance of utilizing the seismic experience-based methodology of the above referenced EPRI TR is only limited to its application for the implementation of the proposed PBNP AST and it is not an endorsement for its use for other applications at PBNP Units 1 and 2.

The NRC staff reviewed the above enclosures and the licensee's responses to the NRC staff RAIs regarding the seismic ruggedness of the PBNP CREFS and VNPAB ventilation systems which are credited in the proposed AST and within the scope of review. The extent of the licensee's review included ductwork, duct supports and associated in-line components such as registers, dampers, damper actuators, fans, expansion joints, filter units and plenums. The licensee assessed the structural integrity of the HVAC SSCs by considering loads due to deadweight, seismic events and pressure.

The NRC staff found the design basis ground spectra and resulting in-structure response spectra that PBNP used for the purpose of the unresolved safety issue A-46 resolution to be conservative. The licensee used the same response spectra for the seismic verification of the subject HVAC SSCs. The NRC staff finds this approach acceptable.

The NRC staff reviewed the licensee's seismic report and verified that it is applicable to the screening and seismic verification guidelines of the EPRI TR, as it meets its temperature and material limitations. In its seismic report, the licensee has also demonstrated that the PBNP free field spectrum is enveloped by the GIP Seismic Motion Bounding Spectrum and the horizontal zero period acceleration at all HVAC support anchorage points are less than 2.0g for SSE, which are also applicability requirements for using the EPRI TR guidelines. Therefore, based on the above, the NRC staff considers the EPRI TR guidelines acceptable for the evaluation of HVAC SSCs credited in the proposed AST at PBNP.

The SRT walked down the HVAC SSCs which are within the scope of the proposed AST to screen and identify duct and support seismic vulnerabilities and undesirable conditions that could lead to damage or failure in an earthquake. The SRT included in the seismic verification report, the walkdown data sheets as well as the seismic walkdown evaluation and results. Several duct and duct support outliers were identified as a result of the walkdowns. The outliers were resolved either by evaluation or by modification. In its responses to the NRC staff's RAI, the licensee provided further outlier evaluations and acceptability justifications. Also in its response to the NRC staff's RAI, the licensee stated that installation of all identified modifications will be completed prior to the AST implementation. The NRC staff reviewed the licensee's responses related to this issue and found them acceptable.

The SRT's walkdown included the charcoal filter banks located in the same rooms with fans W30A and W30B. The SRT comments in the seismic report describe degraded conditions such as missing anchor nuts and raised anchor nuts on the sides that were accessible for walkdowns, while two sides on each unit were inaccessible. The NRC staff reviewed the provided justification for acceptance "as-is" and determined that the justification lacked rigor. In response to an NRC staff RAI, the licensee stated that these "VNPAB filters are not credited in the AST radiological analyses," "are not safety related" and "cannot seismically interact with (i.e., fall on) the fans in the room given its location." Therefore, based on this response, the NRC staff accepts the licensee's determination to "leave-as-is" for the subject filters.

The NRC staff identified in its review that resolution of the floor response spectrum outliers, where demand exceeds capacity for control room supply fans W13B1, W13B2, W14A and W14B, had not been provided. In response to NRC staff RAIs, the licensee stated that modification repairs for these fans are scoped under "Engineering Change (EC) 11690, Alternative Source Term Implementation and CREFS Upgrades to Support AST License Amendment Request, and are scheduled to be installed by the end of 2010, prior to implementation of the Alternative Source Term license amendment. The licensee further stated that the identified corrosion will be cleaned and the load path and anchorage will be inspected to confirm their effectiveness and repaired, as required, to ensure that seismic capacity exceeds seismic demand." The NRC staff finds the licensee's response acceptable as the outliers for these control room supply fans are scheduled to be resolved in a timely manner.

The SRT selected seven representative worst-case bounding samples of different types of duct support configurations and one bounding duct configuration that were identified during the walkdowns to be unusual or heavily loaded for Limited Analytical Review. These are documented in the seismic report. The selected ductwork and duct supports were evaluated in accordance with Section 4 of the EPRI TR. Two of the selected supports for analytical review that did not meet the criteria of the EPRI TR were identified as outliers needing modification. In response to an NRC staff RAI, the licensee revised its seismic report to show all outliers and recommended repair modifications and provided reference of controlled documentation which provides information needed to implement and track these repairs. Based on this supplemental information provided in the licensee's seismic report, the NRC staff finds the licensee's response acceptable.

2.4.3 Conclusion

The licensee employed the EPRI guidelines, which are similar to GIP-2 and the associated NRC SER, to demonstrate the seismic ruggedness for the non-seismically analyzed PBNP ventilation SSCs that are credited in the proposed AST, with the exception of the CREFS mitigating filtration units which were added during the NRC staff's review of the PBNP AST LAR. Therefore, based on its review as summarized above, the NRC staff finds the licensee's assessment of the seismic adequacy of the in-scope HVAC SSCs credited in the proposed AST reasonable and acceptable.

2.5 Human Factors

2.5.1 Regulatory Evaluation

The NRC staff reviewed the licensee's overall request using the guidance contained in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." With regard to the proposed changes to manual operator actions, the NRC staff used the guidance contained in NRC Information Notice (IN) 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times;" ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions;" and NUREG-0800, "Standard Review Plan," Chapter 18.0, "Human Factors Engineering."

2.5.2 Technical Evaluation

Changes to Manual Operator Actions

To support their request to implement an AST at PBNP, the licensee re-analyzed selected DBA, consistent with the requirements of 10 CFR 50.67, "Accident Source Term," and NRC guidance documents (e.g., Regulatory Guide 1.183). Contained within this re-analysis were two manual operator actions which differ from the current licensing basis:

1. Manual operator action to restore the VNPAB will occur within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation. If a loss of coolant accident (LOCA) occurs coincident with a loss of off-site power (LOOP), the VNPAB will be manually restarted to ensure that the auxiliary building vent stack is the source of the release associated with the emergency core cooling system (ECCS) leakage phase of the event.

2. The licensee requests NRC approval of revisions to the PBNP emergency operating procedures to direct operators to continue containment spray (CS) while on sump recirculation. These procedure changes will be implemented following installation of the CS

and RHR system modifications to provide throttling capability during the ECCS recirculation phase.

The licensee requested that the NRC approve these two changes in manual operator actions such that these changes become a part of the current licensing basis for PBNP.
NRC Bases for Approval of Changes to Manual Operator Actions

Using the guidance contained in NRC IN 97-78, ANSI/ANS 58.8-1994, and NUREG-0800, the NRC staff determined that the two proposed changes to manual operator actions are acceptable with respect to human performance.

Credited Operator Action: Restore the Primary Auxiliary Building Ventilation System (VNPAB) within 30 minutes following the alignment of residual heat removal (RHR) to containment sump recirculation mode of operation.

The first proposed change, to restore the VNPAB within 30 minutes following the alignment of RHR to containment sump recirculation mode of operation, is not a part of the current licensing basis at PBNP. A detailed review was conducted for this new manual operator action. The NRC staff concludes that crediting this manual operator action is acceptable, based on the following considerations.

1. Operator action is directed by plant procedures.

Procedure step sequencing will cue the operator to initiate VNPAB directly after stopping the high head safety injection pump. The proposed procedural Step 31 in emergency operating procedure 1.3, Transfer to Containment Sump Recirculation – Low Head Injection, directs the operator to ensure that VNPAB ventilation is in operation (i.e., two fans including either the A or B Auxiliary Building exhaust stack fan and the corresponding A or B Auxiliary Building filter fan).

2. Operator action to initiate VNPAB is a simple task.

Step 31 of the Emergency Operating Procedure (EOP)-1.3 directs the operator to ensure that either the A or B primary auxiliary building exhaust stack fan and the corresponding A or B primary auxiliary building filter fan are in operation. The fan switches are located on the back of the 1C-04 panel of the Unit 1 main control board.

3. The 30 minute operator action time to initiate VNPAB meets the time requirements of ANSI/ANS 58.8-1994.

In order to credit manual operator action, the time limit for taking the manual action must allow for the operator to diagnose plant conditions and take the necessary action. The proposed procedural Step 31 of EOP 1.3 directs the operator to initiate VNPAB directly after stopping the high head safety injection pump, eliminating the diagnosis aspect of the action. The licensee states that the required actions of starting two fans will be taken in the control room, will be direct action EOP steps after sump recirculation is established, and will be sequenced within the manual steps used to transition from the injection phase to the recirculation phase. The fan switches are located on the back of the 1C-04 panel of the Unit 1 main control board. The licensee committed to implement the changes to EOP-1.3 in accordance with the approved administrative procedure governing the EOP verification and validation process, during which timing requirements will be confirmed and documented.

4. Operators will be trained on the new required action.

Operators are routinely trained and evaluated on their ability to properly carry out actions specified in emergency operating procedures. AST-related simulator upgrades to switches, lights, alarms, software changes, and procedure revisions will be completed to support the first cycle of licensed operator requalification training in 2010. Training is accomplished as a part of the engineering change process associated with the AST modifications.

> Change to Emergency Operating Procedure: PBNP emergency operating procedure EOP 1.3 will be changed to direct operator actions to align the Containment Spray (CS) and Residual Heat Removal (RHR) systems for containment spray while on recirculation from the containment sump.

The second proposed change, revisions to the PBNP emergency operating procedures to direct operators to continue CS while on sump recirculation, is not a part of the current licensing basis of PBNP. A detailed review was conducted for these new manual operator actions. The NRC staff concludes that crediting these operator actions is acceptable, based on the following considerations.

1. The operator actions are directed by plant procedures.

The cues that alert the operators to align the CS and RHR systems for containment spray while on recirculation from the containment sump are provided in direct action procedure steps. Proposed EOP 1.3, Step 32, directs the operator to stop the injection phase CS when the reactor water storage tank (RWST) is depleted. Proposed EOP 1.3, Step 33, then directs the operator to establish the reduced RHR deluge (or upper plenum) recirculation flow and the flow-controlled CS recirculation flow path. The operator is not required to manually adjust CS or RHR flow.

2. Proper alignment of the CS pump discharge valves and throttle position is readily verified by control room indications.

Status lights are provided on the main control boards to allow the operator to confirm the proper alignment of the CS pump discharge valves and to confirm that the preset throttle position has been reached for RHR valves SI-852A or B, RHR pump core deluge valves. Regulatory Guide 1.97, Category 2, Type D flow instrumentation is available on the main control boards to allow the operators to monitor the operation of the CS and RHR systems during the ECCS recirculation phase of a LOCA.

3. The 20-minute operator action time to switch from injection to recirculation spray meets the time requirements of ANSI/ANS 58.8-1994.

The dose projections for the LOCA radiological analysis assume that CS is maintained throughout the injection phase and continued for two hours during the ECCS recirculation phase. It is also assumed that there will be no more than a 20-minute spray interruption to switch from injection to recirculation spray. In order to credit a manual operator action, the time limit for taking the manual action must allow for the operator to diagnose plant conditions and take the necessary action. The proposed EOP 1.3, Step 33, directs operator actions to align CS for recirculation, eliminating the diagnosis aspect of the action. The new EOP-1.3 actions are carried out in the control room. The licensee committed to validate the 20-minute interruption to establish CS recirculation in accordance with the approved administrative procedure governing the EOP verification and validation process. The licensee stated that alignment to recirculation spray has been demonstrated on the simulator to ensure that the necessary action can be accomplished well within 20 minutes. Timing requirements will be confirmed and documented as part of the verification and validation process.

4. Operators will be trained on the new required action.

The bases for two-hour time duration for CS recirculation and the 20-minute interruption will be contained in the proposed revisions to BG-EOP-1.3 and will be included as part of the licensed operator requalification training.

2.5.3 Conclusion

The NRC staff reviewed the proposed changes to credit manual operator actions associated with implementing an AST at PBNP. The NRC staff has concluded, based on the considerations discussed above, that (1) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed changes in manual operator actions, (2) such activities will be conducted in compliance with the Commission's regulations and guidance, and (3) the issuance of the license amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the NRC staff finds this request to be acceptable with respect to the proposed changes in operator manual actions.

2.6 Reactor Systems

2.6.1 Background

The licensee requested NRC approval to use of the Westinghouse RAVE methodology for the locked rotor (LR) analysis supporting implementation of the PBNP AST. The RAVE methodology will be used to determine fuel rods in departure from nucleate boiling (DNB) for the analysis of the LR event. The technical specification (TS) affected by this proposal is TS 5.6.4, "CORE OPERATING LIMITS REPORT (COLR)," and involves the addition of a topical report (TR) that documents the Westinghouse RAVE methodology.

The licensee's current analysis of record (AOR) is presented in the "Input Parameters and Assumptions for the Analysis of the Radiological Consequence of the Locked Rotor Accident," on pages 14.1.8-5 and 14.1.8-6 of the 2007 UFSAR. As indicated in the AOR, the licensee determined the offsite and control room doses following the LR event by conservatively assuming that 100 percent of the fuel rods in the core suffer sufficient damage such that all of their gap activity is released to the reactor coolant system (RCS). In support of implementation of the AST, the licensee proposed to use the RAVE methodology to support the input of the LR dose analysis that assumes 30 percent of the fuel rods in the core suffer damage due to DNB, sufficient that all of their gap activity is released to the RCS.

The use of the RAVE methodology would avoid the over-conservatism of the percentage of failed rods assumed in the AOR and provide additional margin to dose safety limits for the analysis of the LR event.

2.6.2 Regulatory Evaluation

Part 50.34 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Contents of applications; technical information," requires that safety analyses reports be submitted that analyze the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the licensing application process, licensees perform SEs to ensure that their safety analyses remain bounding or continue to meet the applicable acceptance criteria for the licensing application conditions. To achieve these goals, licensees confirm that key inputs (such as neutronic and thermal hydraulic parameters) to the safety analyses are, and will remain, conservative with respect to the current design bases. If key safety analysis parameters are not bounded, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied. The NRC staff review was based on the evaluation of technical merit and compliance with all applicable NRC staff guidance associated with reviews of analysis in support of the license amendment request, including NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

2.6.3 Technical Evaluation

In support of the AST application, the licensee proposed to use the Westinghouse RAVE methodology to determine fuel rods in DNB for the LR event. The NRC staff's review of the licensee's proposal and response to the requests for additional information (RAI) is discussed as follows.

2.6.3.1 Use of RAVE for the Locked Rotor Analysis

The RAVE methodology[16] uses inputs from three computer codes: SPNOVA[17] for the 3-D core neutronic kinetic calculation; VIPRE[18] for the core thermal-hydraulic calculation; and RETRAN[19] for the reactor coolant system loop thermal-hydraulic analysis. RAVE, VIPRE and RETRAN are Westinghouse codes that have been approved by the NRC but not been approved for the PBNP application to determine rods in DNB. SPNOVA is part of the advanced nodal code (ANC) which is currently included in the PBNP licensing basis.

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[16] WCAP-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," dated August 2006.
[17] WCAP-10965-P-A, "ANC – A Westinghouse Advanced Nodal Computer Code," dated September 1986.
[18] WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactors Non-LOCA Thermal-Hydraulic Safety Analysis," dated October 1999.
[19] WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," dated April 1999.

In support of the use of the RAVE methodology, the licensee provided compliance reports in Enclosures 5, 6, and 7 of the December 8, 2008, application. The compliance reports include tabulations listing: (1) applicable computer codes and methods; (2) all of the safety evaluation (SE) conditions and limitations within each TR to be used for the LR analysis; and (3) discussion of how these applicable conditions and limitations of the TRs are complied with in the LR analysis. The NRC staff's review of the disposition of each of the conditions and limitations listed in the SE approving the TRs is discussed as follows.

<u>Compliance Report for Safety Evaluations Approving WCAP-16259-P-A</u>

The NRC staff has reviewed the disposition in Table A-1, RETRAN-SPNOVA-VIPRE (RAVE), of Enclosures 5 and 6 of the December 8, 2008, application, for each of the conditions and limitations listed in the SE approving WCAP-16259-P-A (RAVE), and found that the licensee adequately addressed each of them. The following discussion presents the evaluation of the sensitivity studies the licensee provided to support its compliance with the SE conditions and limitations:
Condition 2 listed in the SE for use of the RAVE method states that:

> ... Since different core designs may exhibit different sensitivities, the first implementation of the RAVE sensitivity studies should be performed to ensure that limiting conditions have been identified. The sensitivity results will accompany the analyses using the RAVE methodology whenever the RAVE methodology is first implemented for a plant and must be presented to the NRC staff for review and approval.

In addressing compliance with Condition 2, the licensee performed sensitivity studies using methods in WCAP-16259-P-A to determine the conservative direction of the key inputs for the LR analysis for PBNP and provided the results which contain Westinghouse proprietary information in Enclosures 5 and 6 of the December 8, 2008, application. Tables A-1 and A-2 in the disposition to Condition 2 presented the sensitivity cases used to establish the reference limiting LR rods-in-DNB case and peak pressure case, respectively. Eleven cases were performed for the rod-in-DNB case with sensitivity of parameters including time step size, reactivity feedback coefficients, fuel burn-up, power level and power shape, and control rod insertion. Six cases were performed for the peak pressure case with sensitivity of parameters including time step size and various reactivity feedback coefficients. The NRC staff found that: the NRC-approved TR was used for the analysis; the scope of the sensitivity studied was sufficient for determining the limiting LR cases; and the conservation direction of key inputs was determined and the reference limiting cases were identified. Therefore, the NRC staff concluded that the sensitivity studies were acceptable for meeting the SE Condition 2.

Condition 5 listed in the SE for use of the RAVE method states that:

> ... Westinghouse performed sensitivity studies which demonstrated that the reactor power calculated by the RAVE methodology is insensitive to assumptions for core voiding up to a maximum steam void fraction of 30 percent. If the maximum void fraction in any RAVE reactivity feedback calculation exceeds 30 percent, additional justification will need to be provided for the steam/water

separation model utilized in the VIPRE whole-core model to the staff for additional review of that application of RAVE.

In addressing compliance with Condition 5, the licensee indicated that the impact of exceeding 30 percent void fraction limit was investigated and determined to be conservative to over pressurization during a LR event. The NRC staff requested the licensee provide the results of the analysis to support its conclusion. In response, the licensee summarized its response to Question 7 of the June 1, 2009, letter, stating that the maximum void fraction results obtained for the sensitivity cases presented for RAVE LR overpressure analysis: 6 cases in Table A of the June 1, 2009, letter and 4 additional cases in Table B. The result showed the effects of key input parameters on the LR overpressure event. The licensee performed the final LR overpressure analysis using the most conservative assumptions based on the results of the sensitivity study. The NRC staff, therefore, concluded that the licensee satisfactorily addressed Condition 5 for use of RAVE.

The NRC SE approving the TR, WCAP-16259-P-A, states that the basis of the NRC's acceptance of the TR is, in part, that "Westinghouse will maintain training guidelines that assure only qualified analysts perform and verify the analyses being performed." During the course of the review, the NRC staff requested the licensee to address how analysts meet the Westinghouse training guidelines for use of the RAVE methodology documented in the WCAP. In response, the licensee indicated in response to Question 1 of the June 1, 2009, letter that the RAVE methodology was implemented in accordance with the Westinghouse Quality Management System (QMS), which has been reviewed and approved by the NRC. The analysts and verifiers have been trained and are qualified to perform and verify the RAVE analyses according to Westinghouse QMS. Based on the licensee's response, the NRC staff has confidence that qualified and adequately trained analysts will perform the LR analysis using the RAVE method.

The Westinghouse RAVE methodology contains three Westinghouse computer codes, SPNOVA, VIPRE and RETRAN. The NRC staff requested the licensee to identify and provide the nodalization diagrams for use of the codes that deviate from those used for reference plant documented in applicable WCAP TRs, and justify the deviations. In response, the licensee indicated in response to Question 2 of the June 1, 2009, letter that the SPNOVA, RETRAN and VIPRE models used in the PBNP LR rod-in-DNB analysis utilize consistent nodalization with the same 3-loop plant model shown in the RAVE TR, WCAP-16259-P-A. Specifically, a one-to-one mapping is used for SPNOVA and VIPRE whole-core model nodalization consistent with WCAP-16259-P-A.

The PBNP RETRAN reactor vessel model for PBNP used in the RAVE analyses is consistent with the plant's RETRAN model (2-loop Westinghouse-designed from the plant model discussed in WCAP-14882-P-A). The licensee provided the RETRAN core model in Figure A of the submittal dated June 1, 2009. This model uses the same number of axial nodes as that used for sample 3-loop plant model in the RAVE TR. Since the approval of WCAP-14882-P-A, the reactor coolant system hot-leg modeling was changed to address temperature measurement interactions for pressurizer insurge and outsurge. The hot-leg model change consists of dividing each hot-leg control volume into three control volumes. The change applies to hot-legs in all RCS loop. Since this change has been applied in other RETRAN analyses performed by

Westinghouse and approved by the NRC, the NRC staff concludes that the model change remains acceptable.

The licensee provided the PBNP VIPRE whole core-model radial nodalization for reactivity feedback response in Figure B of Reference 2. This nodalization is consistent with the sample 3-loop whole-core nodalization in Section C of WCAP-16259-P-A. The only difference is that PBNP is a 2-loop plant and the core contains 121 fuel assemblies versus 157 assemblies for the 3-loop core.

The nodalization of the VIPRE subchannel model used in the DNBR calculation is the same as the model shown in Figure 4-2 of WCAP-14565-P-A.

Since the nodal schemes used for SPNOVA, VIPRE and RETRAN are consistent with that used in WCAP-16259-P-A and represent the PBNP plant and core configurations, the NRC staff concludes that the nodal schemes are acceptable for the PBNP LR analysis to determine fuel rods in DNB.

TS 5.6.4, Core Operating Limits Report (COLR)

In support of implementation of the use of the RAVE methodology for the LR analysis, the licensee proposed a change to TS 5.6.4.b by adding the following:

(13) WCAP-16259 P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis."

The approved analytical methods used to determine the core operating limits that are acceptable to the NRC are provided in TS 5.6.4.b. WCAP-16259 P-A is acceptable on the bases discussed in Section 2.8.7 for use in determining fuel rods in DNB for the PBNP LR analysis. The NRC staff concludes that the proposed TS change meets the TS 5.6.4.b requirements and, therefore, is acceptable.

Compliance Report for Safety Evaluations Approving WCAP-10965-P-A, WCAP-14565-P-A, and WCAP-14882-P-A

The NRC staff has reviewed the disposition in Enclosure 7 of the December 8, 2008, submittal, for each of the conditions and limitations listed in the SE approving RETRAN (WCAP-14882-P-A), VIPRE (WCAP-14565-P-A), and SPNOVA (WCAP-10965-P-A), and found that the licensee adequately addressed each of them. Condition 1 listed in the SE approving WCAP-14882-P-A allows use of the RETRAN for 15 events that listed in Table 1 of the SE. The NRC staff has found that the LR event is one of the events that are allowed to use RETRAN for RCS response analysis. Therefore, the NRC staff concludes that the SE compliance report of Enclosure 7 is acceptable to support the use of RETRAN, VIPRE and SPNOVA in the LR analysis using the RAVE method.

2.6.3.2 Steam Generator Tube Rupture Analysis That Calculates the Margin to Steam Generator Overfill

Section 6.2 of Enclosure 3 to the December 8, 2008, submittal, documents the analysis of the SG tube rupture (SGTR) event. The SGTR analysis indicates that "the equilibrium primary-to-secondary break is assumed to persist until 30 minutes after the initiation of the SGTR, at which time the operators have completed the actions necessary to terminate the steam release from the ruptured SG. Pressure between the ruptured SG and primary system is such that the ruptured SG is not overfilled." The NRC staff notes that the consequences of a SGTR depend largely on the ability of the operator to take necessary actions to terminate the primary-to-secondary break flow. The licensee does not indicate what are the operator actions and the associated completion times. If the break flow continues for an extended period of time, the secondary side of the SG may be filled and water may enter the steam lines, which results in unanalyzed conditions.

In response to a RAI requesting the licensee to address the margin to SG overfill during a SGTR event, the licensee presented in the June 1, 2009, letter, the results of a supplemental SGTR analysis analyzed at extended power uprate conditions with a core power of 1811 MWt for PBNP Units 1 and 2. The accident analyzed is a double-ended rupture of a single SG tube, which is the limiting break size identified in the AOR. The analysis does not consider the effect of single failures. The licensee based its decision not to assume the worst single failure on the fact that its current licensing basis does not include a single failure.

The licensee uses the NRC-approved computer code, LOFTTR2, to analyze the SGTR. It models operator responses based on the PBNP simulator exercise results and emergency operating procedure EOP-3. The PBNP-specific operator actions and the associated action times modeled in the analysis are shown in Table 1 of the June 1, 2009, letter, for ruptured SG isolation, initiation of RCS cooldown and RCS depressurization, and emergency core cooling system flow termination.
The licensee applies to the analysis a set of operating conditions to provide a minimum margin to SG overfill. In the analysis, the licensee uses the low end (558 °F, higher initial reactor coolant density) of the RCS average temperature range (558 °F to 577 °F) and maximum SG tube plugging level of 10 percent (lower initial secondary pressure) for maximizing the primary-to-secondary break mass flow rate. The analysis models the Unit 1 Model 44F SG to bound the Unit 2 Model Δ47 SG. It includes consideration of maximum safety injection and auxiliary feedwater flow rates for maximizing ruptured SG water inventory. Also, the licensee performed a sensitivity study of the decay heat model on the SG overfill margin, and determined that the nominal 1971 American Nuclear Society (ANS) decay heat used in the SGTR analysis would be more limiting than 1971+20% ANS decay heat for the PBNP margin to SG overfill analysis.

Since the higher decay power levels associated with the extended power uprate (EPU) of 1800 MWt will result in a longer cooldown and accumulation of additional break flow in the secondary side of the ruptured SG, the licensee assumed that the initial power is at the EPU power level in the margin-to-overfill (MTO) analysis to bound that of the current licensed thermal power (CLTP) level of 1540 MWt. However, the licensee subsequently indicated in its September 28, 2010, letter, in the response to a RAI during a separate NRC-review of the licensee's EPU application, that it had performed a sensitivity study and concluded that there would be a smaller margin at CLTP level than that at the EPU power level. The sensitivity study reveals that the effect of the higher decay power on the MTO will be offset by the lower initial secondary mass inventory associated with the higher EPU power level. In addressing the non-conservatism of using the EPU power level, the NRC staff requested the licensee to provide an MTO analysis at both CLTP and EPU conditions for limiting MTO cases performed with the NRC-approved methods. In response, the licensee provided a conservative analysis in its December 1, 2010, letter and demonstrated that no SG overfill would occur at both EPU and CLTP conditions. The NRC staff reviewed the analysis and provided the following evaluation focused on the MTO analysis at CLTP conditions as it is applicable to the AST application.

Analytical Methodology

The thermal hydraulic analysis is performed using the LOFTTR-2 code in a manner consistent with the methodology described in WCAP-10698-P-A with the exception of assuming a single failure. The results of this evaluation indicates that at AST (with CLTP level of 1540 MWt) conditions, there will be 30 ft^3 of available SG water volume.

The NRC staff audited the calculation notebooks supporting the result of the MTO analysis, and found that no departures from the approved WCAP-10698-P-A analytic methodology, with the exception of the omission of the single failure assumption, and more conservative changes to the decay heat modeling, were identified. The licensee based its decision not to assume the worst single failure on the fact that its current licensing basis did not include a single failure. The NRC staff also found that the licensee did, however, use a plant-specific value for the target cooldown temperature, as opposed to a generic value. This enabled the licensee to assume that the operators were responding to the event more quickly. This assumption is acceptable because it is based on PBNP-specific operating procedures. Therefore, the NRC staff determines that the MTO analysis was performed using appropriate analytic methods with conservative assumptions that results in a smaller MTO.

Use of Non-Safety-Related Equipment

In the September 28, 2010, letter, the licensee indicated in response to RAI SRXB-5 that all equipment credited for mitigating the SGTR event is safety-related, with the exception of main feedwater pumps and discharge MOVs, condenser steam dump valves, instrument air (IA) compressors, and main steam system radiation monitors.

Under the analyzed limiting MTO case for an SGTR event with concurrent LOOP, the main feedwater pumps and the condenser steam dump valves will not be available. This equipment is not considered available in the MTO analysis for an SGTR event concurrent with LOOP.

The main steam system radiation monitors, of which there are three sets, are augmented quality and are used to identify the ruptured SG. Abnormal level deviations and SG sampling can also be used to identify a ruptured SG.

The SG atmospheric dump valves (ADVs), which are safety-related, are credited for mitigating the MTO event. The motive force for manual operation of these valves is provided, however, by the non-safety-related instrument air system.

In response to NRC Question 2, as provided in Enclosure 1 in to the December 1, 2010, letter, the licensee stated that ADV operability would be assured by the following means:

1. With LOOP on the affected unit, instrument air from the other unit is available.

2. With LOOP on both units, there is available volume in the IA receiver, in the meantime, the IA compressors are loaded on the emergency diesel generators by steps in the applicable abnormal operating procedure and alarm response procedures.

3. In addition, local operation of the ADVs is available.

In its assessment of the licensee's assumption of ADV operability under LOOP conditions, the NRC staff verified that each of the above items would assure that the ADV would function as analyzed in the MTO evaluation.

Regarding Item 1, although this assumption is not consistent with typical design basis events requiring the assumption of a LOOP, it is consistent with the design basis analysis addressing radiological consequences. If the assumed, concurrent LOOP were to affect only the single unit, Item 1 would assure ADV operability to perform its safety function.

For Item 2, the NRC staff requested that the licensee identify the sources of power under dual-unit LOOP conditions to the ADV actuation circuitry, including to the control room valve switch and to the control system that regulates the instrument air. The licensee stated in its January 13, 2010, letter that the ADV controls are powered from 120V instrumentation (battery backed) AC buses. The ADV position indication lights are powered from 125V DC vital buses, which are powered from safety-related batteries.

The licensee also stated that the ADVs require instrument air to operate remotely from the control room. The ADVs receive air from the instrument air system headers, which supply air to both Unit 1 and 2 ADVs. The IA compressors are powered from the safety-related 480V AC buses, which can be powered from EDGs. On a LOOP, the IA compressors are manually energized by cycling the control switches in the control room to the off position to reset the breaker, and then to the on position to restart the compressor. This is accomplished in the control room with no local field action required.

During the January 6, 2011 audit at the plant site, the NRC staff observed that the licensee demonstrated successful restoration of instrument air following a dual unit LOOP during the PBNP simulator exercises. In the simulated scenario, the NRC staff observed that the instrument air pressure gradually declined until a uniquely colored annunciator alerted the operators to the instrument air loss. Alarms also indicated the loss of instrument air to the MSIVs. When the annunciator was acknowledged, an operator immediately checked to ensure there was available load on the EDG, and re-loaded instrument air to the DG. This was accomplished by turning a switch in the control room. The entire evolution from LOOP to restoration of instrument air occurred in less than five minutes. Based on its observation of the licensee's simulator exercises, the NRC staff determines that, while a dual unit LOOP is an unlikely event concurrent with a SGTR, the operators at the unaffected unit would successfully restore instrument air, establishing a defense-in-depth provision of instrument air availability.

The licensee also provided information to address the operability of the IA system during various LOOP scenarios, and its ability to provide motive force to the ADV. If only a single IA compressor is lost, the remaining compressor is sufficient to provide the necessary air supply to serve both units. Although the loss of both IA compressors would cause a gradual reduction in IA pressure, the licensee estimated that, following a total reduction in IA pressure, a single restored compressor could repressurize the IA system in less than five minutes to provide sufficient pressure for normal ADV operation.

The NRC staff determines that its site visit activities, along with the licensee's RAI response in the January 13, 2010, letter, provided reasonable assurance that above Item 2 is a credible defense-in-depth to provide that the ADV on the intact SG of the affected unit will be operable from the control room.

Finally, the NRC staff reviewed Item 3, above. In the January 13, 2010, letter and during the January 6, 2011, NRC staff site visit and simulator observation, the licensee demonstrated the actions that would be necessary to operate an ADV using a remote, manual action. Overall, if remote, manual operator action were required to open the ADV, the valve would not open within the analytically assumed 17 minutes. The NRC staff finds, based on the licensee's demonstration, that the ADV would be opened within 20-22 minutes if it were required to be opened locally. Although this exceeds the analytically assumed 17 minutes, the NRC staff concludes that there is reasonable assurance that the ADV would remain operable from the control room based on its review of Items 1 and 2 above.

Based on its review, audit, and simulator observation, the NRC staff concludes that there is reasonable assurance that the ADV would operate despite the non-safety-related status of the IA system, because both the IA receiver and loading the compressor to the EDG would assure, with diversity, that instrument air is available to operate the ADV.

In conclusion, the NRC staff finds that (1) the licensee's MTO analysis has adequately accounted for operation of the plant at the CLTP conditions, (2) the analysis was performed with appropriately conservative analytical methods and approved computer codes, (3) the assumptions used in the analysis are conservative, resulting in a smaller MTO, and (4) the results show that the SGTR event would likely not result in an overfill of the ruptured SG. Therefore, the NRC staff concludes that the MTO analysis is acceptable to show no SG overfill will occur during an SGTR event for the AST application.

2.6.4 Conclusion

Based on its review, the NRC staff finds that (1) the RAVE is a NRC-approved method for Westinghouse manufactured plants to determine fuel rods in DNB during a LR event, and (2) the PBNP licensee satisfactorily shows compliance with the conditions and restrictions listed in the SE approving the RAVE method. Therefore, the NRC staff concludes that the proposed use of RAVE is acceptable.

The NRC staff also finds that the NRC-approved methodology is used in the SG MTO analysis with conservative assumptions, resulting in a smaller MTO and the results show that the SG overfill will not occur during an SGTR event. Therefore, the NRC staff concludes that the MTO analysis is acceptable

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 12, 2010 (75 FR 62602). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 14, 2011

ATTACHMENT 1

Table 3.1-1
Point Beach Nuclear Plant Units 1 and 2
Control Room Atmospheric Dispersion Factors
(X/Q Values, sec/m^3)

	0 -2 hrs	2 – 8 hrs	8 – 24 hrs	1 - 4 days	4 – 30 days
Unit 2 Containment Wall	1.39×10^{-3}	9.80×10^{-4}	3.84×10^{-4}	3.46×10^{-4}	3.02×10^{-4}
Auxiliary Building Vent	1.80×10^{-3}	1.31×10^{-3}	5.15×10^{-4}	4.03×10^{-4}	3.03×10^{-4}
Unit 2 RWST	9.89×10^{-3}	7.98×10^{-3}	2.88×10^{-3}	2.75×10^{-3}	2.35×10^{-3}
Unit 2 "A" MSSVs	4.66×10^{-3}	3.40×10^{-3}	1.17×10^{-3}	1.07×10^{-3}	9.05×10^{-4}
Unit 2 Containment Façade Penetration	1.87×10^{-2}	1.50×10^{-2}	5.11×10^{-3}	4.94×10^{-3}	4.23×10^{-3}
Unit 2 Purge Stack	6.94×10^{-3}	-----	-----	-----	-----

Table 3.1-2
Point Beach Nuclear Plant Units 1 and 2
EAB and LPZ Atmospheric Dispersion Factors
(X/Q Values, sec/m^3)

EAB	
0-2 hrs	5.0×10^{-4}
LPZ	
0-8 hrs	3.0×10^{-5}
8-24 hrs	1.6×10^{-5}
1-4 days	4.2×10^{-6}
4-30 days	8.6×10^{-7}

Table 3.2-1
Point Beach Nuclear Plant Units 1 and 2
Calculated Radiological Consequences TEDE [1] (rem)

Design Basis Accident	EAB [2]	LPZ[3]	CR
Loss of Coolant Accident	14.2	1.6	4.93
Steam Generator Tube Rupture Accident			
pre-accident iodine spike	2.0	0.2	1.9
Accident initiated iodine spike	0.6	0.1	0.5
Locked Rotor Accident	2.0	0.5	4.6
Main Steam Line Break			
pre-accident iodine spike	0.14	0.03	1.9
concurrent iodine spike	0.20	0.08	4.0
Control Rod Ejection Accident	2.3	0.8	2.9
Fuel Handling Accident	2.7	0.2	4.3
Reactor Vessel Head Drop Accident	0.1	0.1	0.5

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone

Table 3.2-2
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the LOCA

Parameter	Value
Reactor power	1811 MWt
Iodine Chemical Form in Containment	
Elemental	4.85%
Organic (methyl)	0.15%
Particulate (cesium iodide)	95%
Containment Net Free Volume	1.0E+06 ft^3
Containment Sprayed Volume	5.82E+05 ft^3
Fan Cooler Units	
Number in operation	2
Flow rate (per unit)	33,500 cfm
Delay time to start	90
Containment Leak Rates	
0 - 24 hours	0.2 weight %/day
> 24 hours	0.1 weight %/day
Spray Operation	
Injection Sprays Initiated	0 - 90 seconds
Injection Sprays Terminated	1.00 hour
Delay Time to Recirculation Sprays	20 minutes
Recirculation Spray Duration	2 hours
Average Spray Fall Height	65.58 feet
Spray Flow Rates	
Injection	1,070 gpm
Recirculation	900
Containment Spray Removal Coefficients	
Spray elemental iodine removal	
Injection	20.0 hr-1
Recirculation	9.20
Spray particulate removal	
Injection	4.42
Recirculation	3.72

Table 3.2-2
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the LOCA (continued)

Parameter	Value
Containment Spray DF	
Elemental	200
Particulate	1000
Sedimentation Particulate Removal (Unsprayed region: From start of event; Sprayed region: When sprays not operating.)	0.1 hr^{-1}
Containment Sump Volume	2.43E+05 gal
Containment Sump pH	≥ 7.0
RWST Minimum Water Volume	25,500 gal
RWST Maximum Air Volume	270,000 gal
RWST Minimum Temperature	40°F
RWST Maximum Temperature	100°F
RWST Maximum Boron Concentration	3500 ppm
Time to Initiate ECCS Recirculation	0 min
ECCS Leak Rate	800 cc/min
PAB Leak Rate	300 cc/min
RWST Leak Rate	500 cc/min
Iodine Species ECCS Leakage Released to the Atmosphere	
Elemental	97%
Organic	3%
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-3
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the SGTR

Parameter	Value
Reactor Coolant Iodine Activity (Initial)	
Pre-Accident Spike	60 µCi/gm DE I-131
Accident-Initiated Spike	0.5 µCi/gm DE I-131
Noble Gas	520 µCi/gm DE Xe-133
Alkali Metal	Corresponds to 0.5 µCi/gm DE I-131
Accident-Initiated Iodine Spike Factor	335
Primary-to-Secondary Leak Rate	1000 gm/min per SG
Duration of Accident-Initiated Iodine Spike	8 hrs
Secondary Coolant Activity (Initial)	
Iodine	0.1 µCi/gm DE I-131
Alkali Metal	Corresponds to 0.1 µCi/gm DE-I-131
Reactor Coolant Initial Mass	1.06E+08 gm
Steam Generator Initial Mass (each)	2.99E+07 gm/SG
Offsite power	Lost at time of reactor trip (220 sec)
Primary-to-Secondary Leakage Duration for intact SG	30 hours
Ruptured Steam Generator	
Pre-trip Break Flow	21,300 lbm (0 - 220 sec)
Post-trip Break Flow	103,200 lbm (220 sec - 30 min)
Pre-trip Flashed Break Flow	4,690 lbm (0 - 220 sec)
Post Trip Flashed Break Flow	13,420 lbm (220 sec - 30 min)
Steam Release	1130 lbm/sec (0 - 220 sec) 88,100 lbm (220 sec - 30 min)
SG Iodine Partition Factor:	
Non-flashed	0.01
Flashed	1.0
SG Particulate Retention	
Non-flashed	0.0025
Flashed	1.0
Condenser Partition Factor	0.01

Table 3.2-3
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the SGTR (continued)

Parameter	Value
Intact Steam Generator	
Primary-to-Secondary Leakage	1000 gm/min per SG
Steam Release	1130 lbm/sec (0 - 220 sec) 257,700 lbm (220 sec - 2 hr) 584,000 lbm (2 - 8 hr) 866,000 lbm (8 - 24 hr) 54,100 lbm/hr (>24 hr)
SG Iodine Partition Factor	0.01
SG Particulate Retention	0.0025
Condenser Partition Factor	0.01
Iodine Species Released to the Atmosphere	
Elemental	97%
Organic	3%
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-4
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the LR

Parameter	Value
Fraction of Fuel Rods in Core Assumed to Fail for Dose Considerations	30% of core
Gap Fractions	
I-131	0.08
Kr-85	0.10
Other Iodines and Noble Gases	0.05
Alkali Metals	0.12
Radial Peaking Factor	1.7
Reactor Coolant Activity (Initial)	
Iodine	0.5 µCi/gm DE I-131
Noble Gas	520 µCi/gm DE Xe-133
Alkali Metal	Corresponds to 0.5 µCi/gm DE I-131
Secondary Coolant Activity (Initial)	
Iodine	0.1 µCi/gm DE I-131
Alkali Metal	Corresponds to 0.1 µCi/gm DE I-131
Primary-to-Secondary Leakage	2000 gm/min total
SG Iodine Partition Factor	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Species Released to the Atmosphere	
Elemental	97%
Organic	3%
RCS Mass	1.06E+08 gm
Secondary Side mass	
0 - 2 hours	5.98E+07 gm total
> 2 hours	7.37E+07 gm total
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-5
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the MSLB

Parameter	Value
Reactor Coolant Activity (Initial)	
Pre-Accident Iodine Spike	60 µCi/gm DE I-131
Accident-Initiated Iodine Spike	0.5 µCi/gm DE I-131
Noble Gas	520 µCi/gm DE Xe-1 33
Alkali Metal	Corresponds to 0.5 µCi/gm DE I-131
Accident-Initiated Iodine Spike Factor	500
Duration of Accident-Initiated Iodine Spike	4 hours
Secondary Coolant Activity (Initial)	
Iodine	0.1 µCi/gm DE I-131
Alkali Metal	Corresponds to 0.1 µCi/gm DE I-131
Primary-to-Secondary Leakage	1000 gm/min per SG
Steam Release from Faulted SG	5.7E+07 gm
Time to Release Initial Mass in Faulted SG	2 min
Time to Cool RCS Below 212°F (Releases from Faulted SG)	60 hrs
Iodine Form (Atmospheric Release)	
Elemental	97%
Organic	3%
Steam Releases to Environment	
0 - 2 hours	221,153 lbm
2 - 24 hours	1,048,064 lbm
24 - 30 hours	201,570 lbm
SG Iodine Partition Factor	
Faulted SG	1.0
Intact SG	0.01
SG Particulate Retention Factor	0.0025
RCS Mass	1.06E+08 gm
Initial Intact SG mass	2.99E+07 gm
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-6
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the CRDA

Parameter	**Value**
Fraction of Fuel Rods in Core that Fail	10% of core
Gap Fractions	
Iodine	0.10
Noble Gas	0.10
Alkali Metals	0.12
Fraction of Fuel Melting	0.25% of core
Radial Peaking Factor 1.7	1.7
Fraction of Activity Released from Melted Fuel	
Containment Leakage	
Iodine	50%
Noble Gas	100%
Alkali Metals	50%
Primary-to-Secondary leakage	
Iodine	50%
Noble Gas	100%
Alkali Metals	50%
Reactor Coolant Activity (Initial) See Table 6.	
Iodine	0.5 μCi/gm DE I-131
Noble Gas	520 μCi/gm DE Xe-133
Alkali Metal	Corresponds to 0.5 μCi/gm DE I-131
Secondary Coolant Activity (Initial) See Table 6.	
Iodine	0.1 μCi/gm DE I-131
Alkali Metal	Corresponds to 0.1 μCi/gm DE I-131
Containment Net Free Volume	1.0E+06 ft^3
Containment Leak Rates	
0 - 24 hours	0.2 weight %/day
> 24 hours	0.1 weight %/day
Iodine Chemical Form in Containment	
Elemental	4.85%
Organic	0.15%
Particulate (cesium iodide)	95%
Spray Removal in Containment	Not Credited

Table 3.2-6
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the CRDA (continued)

Parameter	Value
Sedimentation Removal in Containment	
Iodines	Not Credited
Alkali metals	0.1 hr^{-1}
Primary-to-Secondary Leakage	
Leakage Rate	2000 gm/min total
Duration	2000 sec
Steam Release to Environment	
0 - 2 hours	213,295 lbm
2 - 14 hours	719,045 lbm
14 - 30 hours	561,112 lbm
SG Iodine Partition Coefficient	0.01
SG Alkali Metal Retention Factor	0.0025
Iodine Chemical Form After Release to Atmosphere	
Elemental	97%
Organic	3%
RCS Mass	1.06E+08 gm
Total SG Mass	5.98E+07 gm
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-7
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the FHA

Parameter	Value
Radial Peaking Factor	1.7
Fuel Damaged	1 assembly
Time from Shutdown before Fuel Movement	65 hrs
Activity Released from Water Pool	
I-130	1.95E-01 Ci
I-131	3.50E+02
I-132	2.97E+02
I-133	8.78E+01
I-135	7.45E-01
Kr-85m	8.32E-01
Kr-85	2.59E+03
Kr-87	1.59E-11
Kr-88	6.53E-03
Xe-131m	7.69E+02
Xe-133m	2.81E+03
Xe-133	1.18E+05
Xe-135m	2.43E+01
Xe-135	2.46E+03
Iodine chemical form. in pool	
Elemental	99.85%
Organic (methyl)	0.15%
Gap Fractions	
I-131	0.12
Kr-85	0.30
Other Iodines and Noble Gases	0.10
Water Depth	23 feet
Overall Pool Iodine Scrubbing Factor	200
Filter Efficiency	No filtration assumed
Isolation of Release	No isolation assumed
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation (VNCR accident mode) After Receiving an Isolation Signal	10 minutes
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-8
Point Beach Nuclear Plant Units 1 and 2
Parameters and Assumptions for the RVHD

Parameter	Value
Fuel Damaged	100%
Fuel Melt	0%
Time from Shutdown before Head Movement	Immediate
Gap Fraction - Iodine	0.05
Iodine Form Released to Environment	
Elemental	100%
Organic	0%
Recirculation Initiation Time	Immediate
ECCS Leak Rate to Auxiliary Building	300 cc/min
ECCS Leak Rate to RWST	500 cc/min
Containment Sump Volume	2.43E+05 gal
RWST Minimum Water Volume	25,500 gal
RWST Maximum Air Volume	270,000 gal
RWST Minimum Temperature	40°F
RWST Maximum Temperature	100°F
RWST Maximum Boron Concentration	3500 ppm
Control Room Isolation	Immediate Manual Operator Action
Control Room Atmospheric Dispersion Factors	See Table 3.1-1
Control Room Parameters	See Table 3.2-9

Table 3.2-9
Point Beach Nuclear Plant Units 1 and 2
Control Room Parameters

Volume	65,243 ft^3
Control Room Unfiltered Inleakage	
LOCA and MSLB	200 cfm
Remaining Non-LOCA events	300 cfm
Normal Ventilation Flow Rates (VNCR Mode 1)	
Filtered Makeup Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Makeup Flow Rate	2000 cfm
Emergency Mode Flow Rates (VNCR accident mode)	
Filtered Makeup Flow Rate	2500 cfm
Filtered Recirculation Flow Rate	1955 cfm
Unfiltered Makeup Flow Rate	0 cfm
Filter Efficiencies	
Elemental Iodine	95%
Organic (Methyl) Iodine	95%
Particulate	99%
Delay to Switch CR HVAC from Normal Operation to Post Accident Operation after receiving an isolation signal (sec)	60 seconds
Breathing Rate - Duration of the Event	3.5E-04 m^3/sec
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

ATTACHMENT 2

Summary of License Conditions

(Applicable to Units 1 and 2)

Additional Conditions	Implementation Date
Upon implementation of Amendment Nos. 240/244 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.6, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.1 8.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 3.7.9.6, in accordance with Specification 5.5.18.c.(i), shall be within 18 months of implementation of this amendment. b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within three (3) years of completion of the testing prescribed in item a. above. c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 18 months of implementation of this amendment.	Immediately
NextEra Energy Point Beach, LLC shall modify the PBNP control room (CR) radiation shielding to ensure CR habitability requirements are maintained.	No later than the Unit 2 (2011) refueling outage
NextEra Energy Point Beach, LLC shall revise PBNP Emergency Operating Procedures (EOPs) to direct continued containment spray while on sump recirculation.	No later than the Unit 2 (2011) refueling outage
NextEra Energy Point Beach, LLC shall modify the control room emergency filtration system (CREFS) to create a new alignment for the accident mode that provides a combination of filtered outside air and filtered recirculation air. The modifications shall include redundancy for all CREFS active components that must reposition from their normal operating position, and auto-start capability on loss of offsite power in conjunction with a containment isolation or high control room radiation signal from an emergency diesel generator supplied source for the CREFS fans required for the new system alignment.	No later than the Unit 2 (2011) refueling outage

Additional Conditions	Implementation Date
NextEra Energy Point Beach, LLC shall modify the primary auxiliary building (PAB) ventilation system (VNPAB) to ensure redundancy of active components needed to operate the PAB exhaust system. VNPAB components required to direct radioactive releases in the PAB to the vent stack shall be upgraded to an augmented quality status. No credit is taken by AST for the PAB charcoal filters. NextEra Energy Point Beach, LLC shall revise PBNP EOPs to address starting the VNPAB fans.	No later than the Unit 2 (2011) refueling outage
NextEra Energy Point Beach, LLC shall perform Train B Emergency Diesel Generator load testing over a range of 2877 to 2950 kW at rated power factor. This license condition will remain in effect until implementation of LAR 261 for Unit 2.	No later than the Unit 2 (2011) refueling outage
NextEra Energy Point Beach, LLC shall install and support CREFS mitigating filtration unit(s) and associated ductwork and bubble tight dampers to Seismic Class I requirements as defined in FSAR Appendix A.5. The mitigating filtration unit(s) shall be seismically qualified in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable.	No later than the Unit 2 (2011) refueling outage
NextEra Energy Point Beach, LLC shall procure the CREFS mitigating filtration unit with electrical power requirements equivalent to the CREFS filter fan motors (i.e., equivalent horse power, efficiency, power factor, and voltage requirements).	No later than the Unit 2 (2011) refueling outage

ATTACHMENT 3

Summary of Licensee Commitments

Letter Number (ADAMS AN)	Commitment	Commitment Change Letter Number (ADAMS AN)	Revised Commitment
NRC 2009-0023 (ML090540860)	The VNPAB system will be added to the scope of the Maintenance Rule (10 CFR 50.65) and the scope of the License Renewal Program (10 CFR 54.37(b)). These actions will be completed during the Unit 1 (Spring 2010) refueling outage that implements the LAR.	NRC 2010-0016 (ML100360077)	The VNPAB system will be added to the scope of the Maintenance Rule (10 CFR 50.65) and the scope of the License Renewal Program (10 CFR 54.37(b)). These actions will be completed no later than the Unit 2 (2011) refueling outage that implements the LAR.
NRC 2009-0076 (ML092540146)	A seismic adequacy review of ventilation systems credited in the AST analyses will be conducted in accordance with the guidelines provided in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected on February 14, 1992, and in the December 2006, Electric Power Research Institute (EPRI) Final Report 1014608, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems: Revision to 1007896," as applicable. This seismic verification review, including independent peer review, will be conducted on the Point Beach Units 1 and 2 CREFS and exhaust portion of the VNPAB system credited in the AST analyses and the associated seismic verification report will be provided to the NRC within 30 days of completion. Any required modifications identified by the review will be completed as part of implementation of the AST modifications.	N/A	

Letter Number (ADAMS AN)	Commitment	Commitment Change Letter Number (ADAMS AN)	Revised Commitment
NRC 2010-0011 (ML100360065)	Administrative controls will be established to ensure that CREFS and the primary auxiliary building ventilation (VNPAB) system will not be in concurrent Technical Specification Action Conditions (TSACs) during planned preventive maintenance activities. These controls will be implemented following NRC approval of LAR 241, no later than the Unit 2 (2011) refueling outage.	NRC 2011-0029 (ML110730295)	Administrative controls will be established to ensure that CREFS and the primary auxiliary building ventilation (VNPAB) system will not be in concurrent Technical Specification Action Conditions (TSACs) during planned preventive maintenance activities on components of the CREFS and VNPAB systems. These administrative controls are not applicable to planned preventive maintenance activities performed on common support system components. These controls will be implemented following NRC approval of LAR 241, no later than the Unit 2 (2011) refueling outage.
NRC 2010-0051 (ML101100605)	Written procedures will be available describing mitigating actions to be taken in the event of an intentional or unintentional entry into TSACs 3.7.9.C or 3.7.9.D. These procedures will be implemented following NRC approval of LAR 241, and no later than the completion of the Units 2 (2011) refueling outage.	N/A	
NRC 2010-0051 (ML101100605)	A description of mitigating actions to be taken in the event of an intentional or unintentional entry into TSACs 3.7.9.C or 3.7.9.D will be incorporated into the PBNP FSAR in accordance with 10 CFR 50.71(e).	N/A	

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

Sincerely,

/RA /R Pascarelli for

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 240 to DPR-24
2. Amendment No. 244 to DPR-27
3. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
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HWalker, NRR
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ADAMS Accession No.: ML110240054 *SE memo dated

OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	DIRS/ITSB/BC	DCI/CSGB/BC	DE/EEEB/BC
NAME	TBeltz	BTully	RElliott	RTaylor *	GWilson *
DATE	03/09/2011	03/09/2011	04/08/2011	06/15/2009	10/02/2009
OFFICE	DRA/AADB/BC	DRA/AADB/BC	DRA/AADB/BC	DSS/SRXB/BC	DE/EMCB/BC
NAME	MGavrilas *	TTate *	TTate *	AUlses *	MKhanna *
DATE	10/30/2009	12/23/2010	02/02/2011	01/02/2011	08/27/2010
OFFICE	DSS/SCVB/BC	OGC (NLO w/comments)	NRR/LPL3-1/BC	NRR/LPL3-1/PM	
NAME	RDennig *	LSubin	RPascarelli	TBeltz RJP for	
DATE	12/09/2010	04/08/2011	04/14/2011	04/14/2011	

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