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

January 22, 2013

Mr. James E. Lynch
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089-9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NOS. ME2609 AND ME2610)

Dear Mr. Lynch:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 206 to Renewed Facility Operating License No. DPR-42 and Amendment No. 193 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated October 27, 2009, as supplemented by letters dated April 29, May 25, June 23, August 12 and December 17, 2010; June 22, July 11, August 9, and December 8, 2011; February 13, February 24, and September 13, 2012, respectively.

The amendments modify the PINGP TSs and licensing basis that supports a full scope application of the Alternative Source Term Methodology. In addition, the amendments incorporate TS Task Force-490, "Deletion of E-Bar Definition and Revision to RCS [Reactor Coolant System] Specific Activity Tech Spec," Revision 0.

The amendments include license conditions that defer the implementation of these amendments until after installation of the Unit 2 Replacement Steam Generators.

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

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 206 to DPR-42
2. Amendment No. 193 to DPR-60
3. Safety Evaluation

cc w/encls: Distribution via ListServ

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated October 27, 2009, as supplemented by letters dated April 29, May 25, June 23, August 12 and December 17, 2010; June 22, July 11, August 9, and December 8, 2011; February 13, February 24, and September 13, 2012, respectively, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 206, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

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

   The Alternative Source Term (AST) License Amendments 206/193 will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).

   NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.

   Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.

4. Implementation Requirements:

   1) Prior to implementation of the Alternative Source Term license amendment, NSPM will revise the Prairie Island Nuclear Generating Plant design and licensing bases to indicate that the Steam Generator Water Level – Narrow Range Instruments are required to meet Regulatory Guide 1.97, Revision 2 requirements.

   2) Within 90 days after completion of the outage in which the Unit 2 Replacement Steam Generators are installed, NSPM will implement an administrative control to require Auxiliary Building Special Ventilation Zone boundary integrity during movement of heavy loads over an open reactor vessel containing irradiated fuel assemblies when the containment atmosphere is open to the outside (as described in Updated Safety Analysis Report Section 12.2.12).

This license amendment is effective as of the date of its issuance. The licensee shall implement the license conditions within 30 days.

The balance of the license amendment shall be implemented in accordance with the terms of the license conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: January 22, 2013

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 193
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated October 27, 2009, as supplemented by letters dated April 29, May 25, June 23, August 12 and December 17, 2010; June 22, July 11, August 9 and December 8, 2011; February 13, February 24, and September 13, 2012, respectively, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.193, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

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

   The Alternative Source Term (AST) License Amendments 206/193 will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).

   NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.

   Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.

4. Implementation Requirements:

   1) Prior to implementation of the Alternative Source Term license amendment, NSPM will revise the Prairie Island Nuclear Generating Plant design and licensing bases to indicate that the Steam Generator Water Level – Narrow Range Instruments are required to meet Regulatory Guide 1.97, Revision 2 requirements.

   2) Within 90 days after completion of the outage in which the Unit 2 Replacement Steam Generators are installed, NSPM will implement an administrative control to require Auxiliary Building Special Ventilation Zone boundary integrity during movement of heavy loads over an open reactor vessel containing irradiated fuel assemblies when the containment atmosphere is open to the outside (as described in Updated Safety Analysis Report Section 12.2.12).

This license amendment is effective as of the date of its issuance. The licensee shall implement the license conditions within 30 days.

The balance of the license amendment shall be implemented in accordance with the terms of the license conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: January 22, 2013

ATTACHMENT TO LICENSE AMENDMENT NOS. 206 AND 193

RENEWED FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Renewed Facility Operating License Nos. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by marginal lines.

REMOVE	INSERT
DPR-42, License Pages 3 and 4	DPR-42, License Pages 3 and 4
DPR-60, License Pages 3 and 4	DPR-60, License Pages 3 and 4

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

REMOVE	INSERT
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
3.3.7-1	---
3.3.7-2	---
3.3.7-3	---
3.3.7-4	---
3.4.17-1	3.4.17-1
3.4.17-2	3.4.17-2
3.4.17-4	---
3.7.12-1	3.7.12-1
3.7.12-2	3.7.12-2
---	3.7.12-3
3.7.13-1	3.7.13-1
3.7.13-2	---
3.9.4-1	3.9.4-1
3.9.4-2	---
5.0-23	5.0-23
5.0-28	5.0-28
5.0-30	5.0-30

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

REMOVE	INSERT
DPR-42, License Page B-3	DPR-42, License Pages B-3 and B-4
DPR-60, License Page B-3	DPR-60, License Pages B-3 and B-4

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

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM 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 and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

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

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 206, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

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

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM 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 and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

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

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 193, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.S4(p). The combined set of plans, which contains

Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," submitted by letters dated October 18, 2006, and January 10, 2007, and as supplemented by letters dated March 18 and June 2, 2011, and approved by NRC Safety Evaluation dated August 16, 2011.

NSPM shall fully implement and maintain an effect all provisions of the Commission-approved Northern States Power Company - Minnesota (NSPM) Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NSPM CSP was approved by License Amendment No. 202.

(4) Fire Protection

NSPM shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 21, 1980, December 29, 1980, July 28, 1981, October 27, 1989, and October 6, 1995, subject to the following provision:

> NSPM may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 206, are hereby incorporated into this license. NSPM shall operate the facility in accordance with the Additional Conditions.

(6) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

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

Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," submitted by letters dated October 18, 2006 and January 10, 2007, and as supplemented by letters dated March 18 and June 2, 2011, and approved by NRC Safety Evaluation dated August 16, 2011.

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved Northern States Power Company - Minnesota (NSPM) Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NSPM CSP was approved by License Amendment No. 193.

(4) Fire Protection

NSPM shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 21, 1980, December 29, 1980, July 28, 1981, October 27, 1989, and October 6, 1995, subject to the following provision:

> NSPM may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 193, are hereby incorporated into this license. NSPM shall operate the facility in accordance with the Additional Conditions.

(6) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

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

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

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Amendment Number	Additional Conditions	Implementation Date
158	The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 158 shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.	October 31, 2002
158	The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table LR, "Less Restrictive Changes – Relocated Details," and Table R, "Relocated Specifications," attached to the NRC staff's safety evaluation dated July 26, 2002. Those requirements shall be relocated to the appropriate documents no later than October 31, 2002.	October 31, 2002
206	The Alternative Source Term (AST) License Amendments 206/193 will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs)	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Amendment Number	Additional Conditions	Implementation Date
206	NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed
206	Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Amendment Number	Additional Conditions	Implementation Date
149	The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 149 shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.	October 31, 2002
149	The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table LR, "Less Restrictive Changes – Relocated Details," and Table R, "Relocated Specifications," attached to the NRC staff's safety evaluation dated July 26, 2002. Those requirements shall be relocated to the appropriate documents no later than October 31, 2002.	October 31, 2002
193	The Alternative Source Term License Amendments ~~206~~/~~193~~ will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Amendment Number	Additional Conditions	Implementation Date
193	NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed
193	Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor output as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose when inhaled as the combined activities of isotopes I-131, I-132, I-133, I-134 and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

1.1 Definitions (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
LEAKAGE	LEAKAGE from the Reactor Coolant System (RCS) 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. 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;

1.1 Definitions

c. Pressure Boundary LEAKAGE (continued)

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 channel 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 steps.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Appendix J of the USAR, Pre-Operational and Startup Tests;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Specific Activity

LCO 3.4.17 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 30 µCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

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 > 30 μCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 580 μCi/gm.	7 days
SR 3.4.17.2	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

3.7.12 Auxiliary Building Special Ventilation System (ABSVS)

LCO 3.7.12 Two ABSVS trains shall be OPERABLE.

----------NOTE----------
The ABSVS boundary may be opened under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ABSVS train inoperable.	A.1 Restore ABSVS train to OPERABLE status.	7 days
B. Two ABSVS trains inoperable due to inoperable ABSVS boundary in MODES 1, 2, 3, or 4.	B.1 Restore ABSVS boundary to OPERABLE status.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Two ABSVS trains inoperable due to inoperable ABSVS boundary during movement of irradiated fuel assemblies. OR Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABSVS train for ≥ 15 minutes with the heaters operating.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.12.2	Perform required ABSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABSVS train can produce a negative pressure within 20 minutes after initiation.	92 days
SR 3.7.12.4	Verify each ABSVS train actuates on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.13 Not Used

3.9 REFUELING OPERATIONS

3.9.4 Decay Time

LCO 3.9.4 The reactor shall be subcritical for at least 50 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor core.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical for less than 50 hours.	A.1 Suspend movement of irradiated fuel assemblies within the reactor core.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify the reactor has been subcritical for at least 50 hours.	Once prior to movement of irradiated fuel in the reactor core following reactor shutdown

5.5 Programs and Manuals (continued)

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of the Control Room Special Ventilation System (CRSVS), Auxiliary Building Special Ventilation System (ABSVS), and Shield Building Ventilation System (SBVS) at least once each 24 months.

Demonstrate for the ABSVS, SBVS, and CRSVS systems that:

a. An inplace DOP test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% (for DOP, particles having a mean diameter of 0.7 microns);

b. A halogenated hydrocarbon test of the inplace charcoal adsorber shows a penetration and system bypass < 0.05% (SBVS not applicable);

c. A laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than: 1) 10% penetration for ABSVS, and 2) 2.5% penetration for the CRSVS when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and 95% relative humidity (RH);

d. The pressure drop across the combined HEPA filters and the charcoal adsorbers (SBVS not applicable to charcoal adsorbers) is less than 6 inches of water at the system flowrate ± 10%; and

e. A laboratory test of a sample of the charcoal adsorber shall have filter test face velocities greater than or equal to the following values for each system: 1) 54 fpm for the CRSVS, and 2) 72 fpm for the ABSVS.

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

2. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J", Section 9.2.3, to allow the following:

(i). The first Unit 1 Type A test performed after December 1, 1997 shall be performed by December 1, 2012.

(ii). The first Unit 2 Type A test performed after March 7, 1997 shall be performed by March 7, 2012.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.06% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.006% of primary containment air weight per day at pressure P_a.

5.5 Programs and Manuals (continued)

5.5.16 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 Special Ventilation System (CRSVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

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

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

c. Requirements for (i) determining the unfiltered air in-leakage 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. Licensee controlled programs that will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the periodic assessments of the CRE boundary.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 206 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-42

AND

AMENDMENT NO. 193 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY – MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated October 27, 2009[1], as supplemented by additional letters[2], Northern States Power Company, a Minnesota corporation (NSPM, the licensee), doing business as Xcel Energy, requested changes to the technical specifications (TSs) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The supplements[2] 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, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 6, 2010 (75 FR 17466).

The proposed changes would revise the TSs to fully implement an alternative source term (AST) methodology at PINGP, Units 1 and 2. The application provides the TS changes and evaluations of the radiological consequences of design-basis accidents (DBAs) for implementation of a full-scope AST in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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."

The licensee also requested the adoption of Technical Specification Task Force (TSTF)-490, Revision 0, "Deletion of E-Bar Definition and Revision to reactor coolant system (RCS) Specific Activity Technical Specification" for pressurized water reactors. This TSTF involves changes to the limits on RCS gross specific activity limits with the addition of a new limit for noble gas

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. (AN) ML093160605.

[2] April 29, 2010 (AN ML101200083), May 25, 2010 (AN ML101460064), June 23, 2010 (AN ML101760017), August 12, 2010 (AN ML102300295, December 17, 2010 (AN ML103510322), June 22, 2011 (AN ML111740145), July 11, 2011 (AN ML111930157), August 9, 2011 (AN ML112220098), December 8, 2011 (AN ML113430091), February 13, 2012 (AN ML120460484), February 24, 2012 (AN ML12058A069), and September 13, 2012 (AN ML12258A057).

specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xenon-133 (DEX) definition that replaces the current E Bar ($\bar{E}$) average disintegration energy definition. In addition, the current dose equivalent Iodine-131 (DEI) definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs).

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ATTACHMENT 1

ATTACHMENT 2

2.0 EVALUATION

2.1 Radiological Consequences Analyses

2.1.1 Regulatory Evaluation

2.1.1.1 TSTF-490

By letter dated September 13, 2005 (Agencywide Documents Access and Management System (ADAMS) AN ML052630462), the TSTF submitted TSTF-490 for NRC staff review. This TSTF involves changes to NUREG-1430, "Standard Technical Specifications – Babcock and Wilcox Plants," NUREG-1431, "Standard Technical Specifications – Westinghouse Plants", and NUREG-1432, "Standard Technical Specifications – Combustion Engineering Plants", Standard Technical Specification (STS) Section 3.4.16 RCS gross specific activity limits with the addition of a new limit for noble gas specific activity. The noble gas specific activity limit would be based on a new DEX definition that replaces the current E Bar ($\bar{E}$) average disintegration energy definition. In addition, the current DEI definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs).

The NRC staff evaluated the impact of the proposed changes as they relate to the radiological consequences of affected DBAs that use the RCS inventory as the source term. The source term assumed in radiological analyses should be based on the activity associated with the projected fuel damage or the maximum RCS TS values, whichever maximizes the radiological consequences. The limits on RCS specific activity ensure that the offsite doses are appropriately limited for accidents that are based on releases from the RCS with no significant amount of fuel damage.

The Steam Generator Tube Rupture (SGTR) accident and the Main Steam Line Break (MSLB) accident typically do not result in fuel damage and therefore the radiological consequence analyses are based on the release of primary coolant activity at maximum TS limits. For accidents that result in fuel damage, the additional dose contribution from the initial activity in the RCS is not normally evaluated and is considered to be insignificant in relation to the dose resulting from the release of fission products from the damaged fuel.

2.1.1.2 Alternative Source Term (AST)

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) requires that the licensee's analysis 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 safety evaluation (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 Criteria (GDC) for Nuclear Power Plants": GDC 19, "Control room";
- 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), 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, June 2001;
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982;
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000;
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, June 2003;
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 0, May 2003;
- NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985;
- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Revision 3, March 2007;

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

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

- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, 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, July 1981;

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

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

The NRC staff also considered relevant information in the PINGP, Units 1 and 2, updated safety analysis report (USAR) 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 PINGP, Units 1 and 2, control room (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. 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 staff's acceptance is based on the licensee's analyses.

2.1.2 Technical Evaluation

2.1.2.1 TSTF-490 TS Changes

2.1.2.1.1 Revision to the definition of DEI

The list of currently acceptable DCFs for use in the determination of DEI includes the following:

- Table III of Technical Information Document (TID)-14844, Atomic Energy Commission (AEC), 1962, "Calculation of Distance Factors for Power and Test Reactor Sites,"

- Table E-7 of RG 1.109, Revision 1, NRC, 1977,

- International Commission on Radiological Protection (ICRP) 30, 1979, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity,"

- Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) DCFs from Table 2.1 of Environmental Protection Agency (EPA) Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

The licensee's current definition allows DEI to be calculated using multiple dose conversion factors (DCFs) from the above list. For the proposed amendment, the licensee is revising the DEI definition to reference the CEDE DCFs from Table 2.1 of EPA FGR-11. Therefore the proposed definition of DEI will read as follows:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose when inhaled as the combined activities of isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

The NRC staff conducted a confirmatory analysis, using various sources of DCFs and the RCS concentration values from the licensee's AST submittal. The NRC staff's analysis confirmed that for a given RCS concentration, use of the DCFs FGR-11 will yield a conservative DEI value. In addition, maintaining the DEI value to the TS limit of 0.5 microCuries per gram (μCi/gm) will be conservative relative to the coolant concentrations used in the design basis dose consequence analyses for PINGP, Units 1 and 2.

2.1.2.1.2 <u>Deletion of the Definition of E Bar ($\bar{E}$) and the Addition of a New Definition for DEXe-133 (DEX)</u>

The proposed definition for DEX is similar to the definition for DEI. The determination of DEX will be performed in a similar manner to that currently used in determining DEI, except that the calculation of DEX is based on the acute dose to the whole body and considers the noble gases Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138, which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half life, or small DCF. The calculation of DEX will use the effective DCFs from Table III.1 of EPA Federal Guidance Report No. 12 (FGR-12), 1993, "External Exposure to Radionuclides in Air, Water, and Soil." Using this approach, the limit on the amount of noble gas activity in the primary coolant would not fluctuate with variations in the calculated values of $\bar{E}$. If a specified noble gas nuclide is not detected, the new definition states that it should be assumed the nuclide is present at the minimum detectable activity. This will result in a conservative calculation of DEX.

When $\bar{E}$ is determined using a design basis approach in which it is assumed that 1.0 percent of the power is being generated by fuel rods having cladding defects and it is also assumed that

there is no removal of fission gases from the letdown flow, the value of $\bar{E}$ is dominated by Xe-133. The other nuclides have relatively small contributions. However, during normal plant operation there are typically only a small amount of fuel clad defects and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of $\bar{E}$ that is very different than would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation and the limiting condition for operation (LCO) becomes essentially meaningless. It also results in a TS limit that can vary during operation as different values for $\bar{E}$ are determined.

The proposed change will implement an LCO that is consistent with the whole body radiological consequence analyses, which are sensitive to the noble gas activity in the primary coolant but not to other non-gaseous activity currently captured in the $\bar{E}$ definition. The current surveillance requirement (SR) 3.4.17.1 for LCO 3.4.17 specifies the limit for primary coolant gross specific activity as 100/$\bar{E}$ µCi/gm. SR 3.4.17.1 currently requires the licensee to verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ µCi/gm. The current $\bar{E}$ definition includes radioisotopes that decay by the emission of both gamma and beta radiation. The current Condition B of LCO 3.4.17 would rarely, if ever, be entered for exceeding 100/$\bar{E}$ µCi/gm since the calculated value is very high (the denominator is very low) because beta emitters such as tritium (H-3) are included in the determination, as required by the $\bar{E}$ definition.

TS Section 1.1 definition for E - AVERAGE DISINTEGRATION ENERGY ($\bar{E}$) is deleted and replaced with a new definition for DEX which states:

> DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

The change incorporating the newly defined quantity for DEX is acceptable from a radiological dose perspective since it will result in an LCO that more closely relates the non-iodine RCS activity limits to the dose consequence analyses which form their bases. The licensee has proposed to use the TSTF-490 approved DCFs in FGR-12 for the calculation of DEX values. The NRC staff also confirmed, by confirmatory calculation, that the licensee's proposed value of 580 µCi/gm DEX accurately reflects the mix of nuclides used in the dose consequence analysis.

2.1.2.1.3 <u>LCO 3.4.17, "RCS Specific Activity"</u>

The licensee proposed a modification to LCO 3.4.17 to specify that iodine-specific activity in terms of DEI and noble gas-specific activity in terms of DEX shall be within limits. Currently, the limiting indicators are not explicitly identified in the LCO, but are instead defined in current Condition A and SR 3.4.17.2 for iodine specific activity and in current Condition B and SR 3.4.17.1 for gross non-iodine-specific activity. The change states, "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits." This change is

consistent with TSTF-490. Therefore, the NRC determined that the proposed revision is acceptable.

2.1.2.1.4 LCO 3.4.17 Applicability Revision

The licensee has proposed to modify the TS 3.4.17 Applicability to include all of MODE 3 and MODE 4. The licensee stated that it is necessary for the LCO to apply during MODES 1 through 4 to limit the potential radiological consequences of an SGTR or MSLB that may occur during these MODES. The licensee also stated that in MODES 5 and 6, the steam generators are not used for decay heat removal, the RCS and steam generators are depressurized, and primary-to-secondary leakage is minimal. Therefore, the monitoring of RCS-specific activity during MODES 5 and 6 is not required. The proposed change to modify the TS 3.4.17 Applicability to include all of MODE 3 and MODE 4 is necessary to limit the potential radiological consequences of an SGTR or MSLB that may occur during these modes and is, therefore, acceptable. The NRC staff determined that the proposed revision is acceptable.

2.1.2.1.5 TS 3.4.17 Condition A

The licensee proposed a modification to TS 3.4.17 Condition A to replace the DEI site-specific limit "> 0.5 μCi/gm" with the words "not within limit" to be consistent with the revised TS 3.4.17 LCO format. The site-specific DEI limit of ≤ 0.5 μCi/gm is contained in SR 3.4.17.2. This proposed format change will not alter current TS requirements and is acceptable from a radiological dose perspective.

The licensee proposed a modification to TS 3.4.17 Required Action A.1 to remove the reference to Figure 3.4.17-1, "Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER" and insert a limit of less than or equal to the site specific DEI spiking limit of 30 μCi/gm. The pre-accident iodine spike analyses assume a DEI concentration 60 times higher than the corresponding long term equilibrium value, which corresponds to the specific activity limit associated with 100 percent RATED THERMAL POWER (RTP) operation. The NRC staff concludes that the proposed TS 3.4.17 Required Action A.1 should be based on the short term site specific DEI spiking limit to be consistent with the assumptions contained in the radiological consequence analyses. Therefore, the NRC staff determined that the proposed revision to TS 3.4.17 required Action A.1 is acceptable.

2.1.2.1.6 TS 3.4.17 Condition B Revision to Include Action for DEX Limit

The licensee proposed that TS 3.4.17 Condition B be replaced with a new Condition B for DEX not within limits. This change is made to be consistent with the change to the TS 3.4.17 LCO which requires the DEX specific activity to be within limits as discussed above in Section 2.1.2.1.3. The DEX limit is site-specific and the numerical value in units of μCi/gm is contained in revised SR 3.4.17.1, as described below in Section 3.1.2.1.8. The site specific limit of DEX in μCi/gm is typically based on the maximum accident analysis RCS activity corresponding to 1 percent fuel clad defects with sufficient margin to accommodate the exclusion of those isotopes based on low concentration, short half-life, or small dose conversion factors. The primary purpose of the TS 3.4.17 LCO on RCS-specific activity and its associated Conditions is to support the dose analyses for DBAs. The whole body dose is primarily dependent on the noble gas activity, not the non-gaseous activity currently captured in the Ē definition.

The Completion Time for revised TS 3.4.17 Required Action B.1 will require restoration of DEX to within limit in 48 hours. This is consistent with the Completion Time for current Required Action A.2 for DEI. The radiological consequences for the SGTR and the MSLB accidents demonstrate that the calculated thyroid doses are generally a greater percentage of the applicable acceptance criteria than the calculated whole body doses. It then follows that the Completion Time for noble gas activity being out of specification in the revised Required Action B.1 should be at least as great as the Completion Time for iodine specific activity being out of specification in current Required Action A.2. Therefore the Completion Time of 48 hours for revised Required Action B.1 is acceptable from a radiological dose perspective. A Note is also added to the revised Required Action B.1 that states, "LCO 3.0.4.c is applicable." This Note would allow entry into a MODE or other specified condition in the LCO Applicability when LCO 3.4.17 is not being met and is the same Note that is currently stated for Required Actions A.1 and A.2. The proposed Note would allow entry into the applicable MODES from MODE 4 to MODE 1 (power operation) while the DEX limit is exceeded and the DEX is being restored to within its limit. This MODE change is found to be acceptable by the NRC staff because a significant conservatism is incorporated into the DEX specific activity limit, the low probability of an event occurring which is limiting due to exceeding the DEX specific activity limit, and the ability to restore transient specific excursions while the plant remains at, or proceeds to power operation.

2.1.2.1.7 <u>TS 3.4.17 Condition C</u>

The licensee proposed to revise TS 3.4.17 Condition C to include Condition B (DEX not within limit) if the Required Action and associated Completion Time of Condition B is not met. This is consistent with the changes made to Condition B which now provide the same completion time for both components of RCS specific activity as discussed in the revision to Condition B. The revision to Condition C also replaces the limit on DEI from the deleted Figure 3.4.17-1, with a site-specific value of > 30 μCi/gm. This change makes Condition C consistent with the changes made to TS 3.4.17 Required Action A.1.

The proposed change to TS 3.4.17 Required Action C.1 requires the plant to be in MODE 3 within six hours and adds a new Required Action C.2 which requires the plant to be in MODE 5 within 36 hours. These changes are consistent with the changes made to the TS 3.4.17 Applicability. The revised LCO is applicable throughout all of MODES 1 through 4 to limit the potential radiological consequences of an SGTR or MSLB that may occur during these MODES. In MODE 5 with the RCS loops filled, the steam generators are specified as a backup means of decay heat removal via natural circulation. In this mode, however, due to the reduced temperature of the RCS, the probability of a DBA involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS-specific activity is not required. In MODE 5 with the RCS loops not filled and MODE 6, the steam generators are not used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

A new TS 3.4.17 Required Action C.2 Completion Time of 36 hours is added for the plant to reach MODE 5. Based on engineering judgment, the NRC staff finds that this Completion Time is reasonable, based on operating experience, to reach MODE 5 from full power conditions in an orderly manner and without challenging plant systems and the value of 36 hours is consistent with other TS which have a Completion Time to reach MODE 5.

2.1.2.1.8 SR 3.4.17.1 DEX Surveillance Requirement (SR)

The licensee proposed a change to replace the current SR 3.4.17.1 action for RCS gross specific activity with a surveillance requirement to verify that the site-specific reactor coolant DEX specific activity is ≤ 580 μCi/gm. This change provides actions for the new LCO limit added to TS 3.4.17 for DEX. The revised SR 3.4.17.1 action requires performing a gamma isotopic analysis as a measure of the noble gas-specific activity of the reactor coolant at least once every seven days, which is the same frequency required under the current SR 3.4.17.1 action for RCS gross non-iodine specific activity. The surveillance provides an indication of any increase in the noble gas specific activity. The results of the surveillance on DEX allow proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

The current surveillance required the licensee to verify reactor primary coolant gross specific activity to ≤ 100/$\bar{E}$ μCi/gm. In the proposed license amendment request (LAR), the licensee proposed to delete the definition and reference to $\bar{E}$, the average disintegration energy, and add a limit for primary coolant noble gas activity based on DOSE EQUIVALENT XE-133, and take into account only the noble gas activity in the primary coolant. The change states, "Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 580 μCi/gm." The results of the surveillance to determine the DEX value allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

The licensee proposed to modify SR 3.4.17.1 with a Note that only requires the surveillance to be performed in MODE 1. This change allows entry into MODE 4, MODE 3, and MODE 2 prior to performing the surveillance and allows the surveillance to be performed in any of those modes, prior to entering MODE 1. This is similar to the current surveillance SR 3.4.17.2 for DEI.

By letter dated May 20, 2010, (ADAMS AN ML101380011), the NRC staff requested the licensee to justify why there is an apparent disparity between the modes of applicability (MODES 1, 2, 3, and 4) and the limited mode (MODE 1) under which the surveillance is required. By supplement letter dated June 23, 2010, the licensee stated that the Note only requiring the SR 3.4.17.1 to be performed in MODE 1 is deleted.

As described in Sections 3.1.2.1.4 and 3.1.2.1.6 above, and in accordance with SR 3.0.1, the Applicability requirement of LCO 3.4.17 applies to operation in MODES 1, 2, 3, and 4 for PINGP, Units 1 and 2. SR 3.4.17.1 is also affected by the inclusion of a Note to TS 3.4.17 Required Action B.1, which permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is found to be acceptable by the NRC staff due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation. This allows entry into MODE 4, MODE 3, and MODE 2 prior to performing the surveillance. This allows the surveillance to be performed in any of those MODES, prior to entering MODE 1.

2.1.2.1.9 SR 3.4.17.2 DEI Surveillance Requirement

As described in Section 3.1.2.1.4 above, and in accordance with SR 3.0.1, the Applicability requirement of LCO 3.4.17 applies to operation in MODES 1, 2, 3, and 4 for PINGP, Units 1 and 2. Currently, a Note exists in SR 3.4.17.2 which reads, "Only required to be performed in

MODE 1." In accordance with the licensee's response to a request for additional information (RAI) dated June 23, 2010, and to be consistent with the applicability and requirements of the SR 3.4.17.1, the licensee proposed to delete this Note from SR 3.4.17.2. Hence, with the proposed change, the DEI SR is required to be met during all MODES of Applicability (MODES 1, 2, 3, and 4). The NRC staff finds the proposed deletion aligns the surveillance requirements with the MODES of Applicability and is, therefore, acceptable.

2.1.2.1.10 SR 3.4.17.3 Deletion

The current SR 3.4.17.3 which required the determination of $\bar{E}$ is proposed to be deleted. TS 3.4.17 LCO on RCS specific activity supports the dose analyses for DBAs, in which the whole body dose is primarily dependent on the noble gas concentration, not the non-gaseous activity currently captured in the $\bar{E}$ definition. With the proposed elimination of the limit for RCS gross specific activity and the addition of the new LCO limit for noble gas-specific activity, this SR to determine $\bar{E}$ is no longer required. This change is consistent with TSTF-490. Therefore, the NRC staff determined that the proposed revision is acceptable.

2.1.2.2 TS 3.7.12 Changes

By letter dated February 24, 2012, the licensee responded to an RAI dated November 2, 2011 (ADAMS AN ML113110094) regarding ventilation within the Auxiliary Building and Common Area of the Auxiliary Building (CAAB). In response to NRC staff questions, the licensee re-examined the radiological accidents and the possible operating conditions of the Auxiliary Building Normal Ventilation System (ABNVS) to identify any other potential for an accident release from the Auxiliary Building normal exhaust stacks. The licensee identified a potential vulnerability from having a common ABSVS shared between the two units, in conjunction with a common TS for that system.

Normally, with either unit in MODES 1-4, TS 3.7.12 would ensure the integrity of the Auxiliary Building Special Ventilation Zone (ABSVZ) prior to the accident and thereby ensure that a radiological release from a fuel handling accident (FHA) and a heavy load drop (HLD) would not escape from the CAAB into the ABSVZ. With respect to the proposed AST license amendment request (LAR), this ABSVZ integrity ensures the CAAB is the limiting release point for a FHA and a HLD.

However, TS 3.7.12 was determined to be inadequate for the proposed amendments in that it does not ensure ABSVZ integrity for the event that neither PINGP unit is operating in MODES 1-4. For example, if Unit 1 experienced a MODE 6 FHA when Unit 2 was in MODE 5 (or 6) with the ABSVZ compromised, the radioactive plume could travel from containment or the spent fuel pool (SFP) into the CAAB, through the compromised ABSVZ boundary, and into the ABNVS exhaust. This path would compromise the assumption that the CAAB was the limiting release path for the FHA.

The licensee proposes revising TS 3.7.12 to extend the ABSVS operability statement to include the fuel handling operations that are precursors to the FHA. The specific changes are:

- Add the following Note, "During movement of irradiated fuel assemblies" to the APPLICABILITY requirements.

- Revise CONDITION B from "Two ABSVS trains inoperable due to inoperable ABSVS boundary" to, "Two ABSVS trains inoperable due to inoperable ABSVS boundary in MODES 1, 2, 3, or 4.

- Revise CONDITION C, from, "Required Action and Associated Completion Time not met", to read, "Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4".

- Add a new CONDITION D that will read, "Two ABSVS trains inoperable due to inoperable ABSVS boundary during movement of irradiated fuel assemblies. OR Required Action associated Completion Time of Condition A not met during movement of irradiated fuel assemblies."

- Add a new REQUIRED ACTION D.1 that will read, "Suspend movement of irradiated fuel assemblies".

- Add a new COMPLETION TIME for REQUIRED ACTION D.1 that will read, "Immediately".

NRC staff finds the above revision to TS 3.7.12 to be an acceptable approach for addressing the potential bypass of analyzed release points, of a radiological plume from a FHA or a HLD, when neither Unit 1 or Unit 2 is in MODE 1, 2, 3, or 4 and ABSVS is inoperable per TS 3.7.12. By requiring that the ABSVS be OPERABLE during movement of irradiated fuel assemblies, the revision to TS 3.7.12 assures that the radioactive release from the FHA and HLD will be enveloped by the CAAB until released from the location assumed in the analysis.

The licensee also revised the TS basis for TS 3.7.12 by adding the background for the TS change and updated the applicable safety analysis section addressing the changes.

2.1.2.3 <u>License Condition</u>

By letter dated February 24, 2012, the licensee responded to the RAI dated November 2, 2011 (ADAMS AN ML113110094) regarding ventilation within the Auxiliary Building and CAAB. In RAI 2, NRC staff requested confirmation that the staff understanding that there are no ventilation systems providing fresh air to the CAAB or that exhaust from the CAAB was correct. In their response, the licensee identified two possible ventilation paths where radioactivity could bypass the isolation of the ABNVS and release to the ABNVS exhaust stack.

The 121 Laundry Dryer Exhaust Fan provides a release path for an FHA, HLD, or an MSLB to the Unit 2 exhaust stack. This release path was not previously recognized and was not analyzed in the current licensing basis. The licensee entered this information into their corrective action program. By letter dated September 13, 2012, the licensee submitted a license condition to implement a plant modification or procedure modification. The proposed license condition states:

> Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.

Therefore, based on the above information and the proposed license condition, the NRC staff finds the licensee provided acceptable means to address the potential radiological release path represented by the 121 Laundry Dryer exhaust fan. The proposed license condition provides the NRC staff with reasonable assurance that an unanalyzed radiological release path will not occur during a design-basis FHA, HLD, or MSLB.

2.1.2.4 Atmospheric Dispersion Estimates

2.1.2.4.1 Meteorological Data

The licensee used five years of onsite meteorological data collected during calendar years 1993 through 1997 to develop new and revised CR and Technical Support Center (TSC) atmospheric dispersion factors (χ/Q values). The CR χ/Q values were used as input to the loss-of-coolant (LOCA), FHA, HLD, MSLB, SGTR, control rod ejection (CREA), and locked rotor accident (LRA) dose assessments in the current PINGP LAR (current LAR), dated October 27, 2009. The TSC χ/Q values were used for the LOCA dose assessments. The 1993 through 1997 data, in the form of hourly meteorological data files in the ARCON96 atmospheric dispersion computer code input format (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"), were previously provided for a prior licensing action by letter dated January 20, 2004 (ADAMS AN ML040270067). The data are discussed further in the SE associated with PINGP, Unit 1 and Unit 2, Amendment Nos. 166 and 156, respectively, dated September 10, 2004 (ADAMS AN ML042430504).

For calculation of doses at the EAB and LPZ, the licensee used current licensing basis χ/Q values based upon meteorological data collected in the early 1970's and derived from the PINGP USAR. The EAB and LPZ χ/Q values are those which were used at initial facility licensing and are discussed further in Section 2.1.2.4.3 below.

2.1.2.4.2 Control Room Atmospheric Dispersion Factors

To assess CR post-accident radiological consequences for the LOCA, FHA, HLD, MSLB, SGTR, CREA, and LRA accidents, the licensee generated χ/Q values using ARCON96 and guidance provided in RG 1.194. RG 1.194 asserts that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in design basis accident radiological analyses. NRC staff evaluated the applicability of the ARCON96 model and determined that there are 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 PINGP.

As described in the Enclosure to the current LAR, the licensee initially considered 26 source-receptor pairs, then performed additional assessment, as discussed in Enclosure 1 to a letter dated August 12, 2010, to identify a reduced set of pairs for use in calculation of doses to the CR operators. Consequently, the licensee considered source-receptor pairs for an accident in either PINGP, Unit 1 or Unit 2, to the normal intakes for the CR ventilation system, designated as 121 CR and 122 CR, from the following locations:

- Unit 1 and Unit 2 SBVS
- Unit 1 and Unit 2, Group 1 and Group 2, MSSVs/ SG PORVs.

- Unit 1 and Unit 2 Auxiliary Building Normal Ventilation Make-Up Air Intake. The Auxiliary Building Normal Ventilation Make-Up Intake is a potential release point for back leakage to the Refueling Water Storage Tank (RWST) during post-LOCA mitigation.

- CAAB. Two possible source locations, the closest point of the northwest and northeast CAAB walls, were modeled to account for the proximity of the CAAB to both CR ventilation intakes.

When calculating the dose to personnel in the TSC, the licensee also considered postulated releases from the following release points to the make-up air intake for the TSC Ventilation System:

- Unit 1 and Unit 2 SBVS.

- Unit 1 and Unit 2 Auxiliary Building Normal Ventilation Make-Up Air Intake.

With regard to release locations, the licensee asserted that all were less than 2.5 times the height of their adjacent buildings and, in accordance with RG 1.194, modeled them as ground level releases as follows. Buoyancy or mechanical jet high energy releases to the environment were not credited in the χ/Q analyses. Limiting χ/Q values were selected to maximize the predicted radiological consequences for each accident sequence, accounting for the accident scenario and progression, loss of offsite power and single failure considerations. Source release points and receptor locations were selected to be conservative regardless of the availability of offsite power. Single-failure considerations were in addition to consideration of a loss of offsite power. The licensee performed further analysis regarding ventilation within the Auxiliary Building and CAAB as described in supplemental submittals dated June 23, 2010, and August 12, 2010. The licensee modeled postulated releases from the SBVS and main steam safety valves and power-operated relief valves (MSSVs/PORVs) as point releases and releases from the Auxiliary Building Normal Ventilation (ABNV) Make-Up Air Intakes and CAAB as diffuse releases. Sections 2.1.2.4.2.1 and 2.1.2.4.2.2 provide additional information.

2.1.2.4.2.1 Point Source Releases

Releases from the SBVS were modeled as point releases. Releases from the MSSVs/PORVs were initially modeled as diffuse releases, but, following further assessment, the licensee remodeled assumed releases from the MSSVs/PORVs as point releases. The licensee provided tables of inputs which included information regarding the source and receptor heights, and distance and direction relationships between the point source-receptor pairs. The licensee also supplied figures to provide visual confirmation of the location of the sources and receptors.

NRC staff reviewed the licensee's assessment of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling for the point source releases. This included review of the inputs and assumptions, which the staff found generally consistent with site configuration drawings, input tables, RG 1.194 criteria, and NRC staff practice. In addition, NRC staff generated a sample set of comparative χ/Q estimates and found the resultant χ/Q values to be similar to those calculated by the licensee for the cases considered.

2.1.2.4.2.2 Diffuse Source Releases

The licensee modeled postulated releases from the ABNV Make-Up Air Intakes and CAAB as diffuse releases. The licensee provided tables which included information regarding the initial diffusion coefficients related to each diffuse source, in addition to source and receptor heights, and distance and direction relationships between the diffuse source-receptor pairs. The licensee also supplied detailed text describing characteristics and figures showing the relationship of the sources and receptors to facility structures and components.

- ABNV Make-Up Air Intakes

  The licensee postulated that the ABNV Make-Up Air Intakes could be potential release points to the environment for back leakage to the RWST during post-LOCA mitigation. The licensee also noted that the horizontal distance between the Unit 2 ABNV Make-Up Air Intake and the 122 CR intake is less than 10 meters.

  - The ABNV Make-Up Air Intakes are louvered. RG 1.194 asserts that modeling a louvered panel or opening as a diffuse source may be appropriate when (1) the release rate from the building interior is essentially equally dispersed over the entire surface of the panel or opening and (2) assumptions of mixing, dilution, and transport within the building necessary to meet condition 1 are supported by the interior building arrangement.

    The licensee asserted that the RWST for each unit includes a vent at the top of the tank which discharges into the cylindrical concrete structure which encloses the tank. The concrete enclosure has an access opening which allows a path for released activity to be discharged into the ABNVS equipment room. If the normal ventilation system is isolated due to the safety injection signal and the start of the ABSVS, there will be no forced air movement in the room other than any natural circulation or differential pressure induced flow between the room and ductwork in the room. With no forced ventilation, the activity is assumed to slowly distribute throughout the room, seep into the ventilation system supply duct through small openings, mix with the air in the duct, and be slowly dispersed through the ABNV Make-Up Air Intake louvers into the environment.

  - The licensee asserted that the horizontal distance from the Unit 2 ABNV Make-Up Air Intakes to 122 CR is approximately 9.2 meters. RG 1.194 asserts that, if the distance from a source to a receptor is less than about 10 meters, the ARCON96 code and the procedures in Regulatory Position 4 of RG 1.194 should not be used to assess χ/Q values. Such situations should be addressed on a case-by-case basis. However, as described in the Enclosure to the LAR, since the distance is within 10 percent of the recommended 10 meters in RG 1.194, the licensee judged that this was a relatively minor difference and it would still be consistent with RG 1.194 to use the ARCON96 computer code. The licensee noted that, in addition, conservatism was present in its choice of the source-receptor combinations, such that the modeled dispersion factors are conservative.

With regard to postulated releases from the ABNV Make-Up Air Intakes, NRC staff reviewed the figures and other information associated with the inputs and assumptions provided by the licensee. Because of the circuitous path and unforced flow between the RWST and interior face of the ABNV Make-Up Air Intake if the normal ventilation system was isolated, NRC staff has determined that the effluent should be fairly evenly distributed throughout the interior building space and dispersed to the environment over the entire surface of the ABNV Make-Up air Intake louvers. NRC staff estimated a "taut string" distance from the source to the receptor, as discussed in RG 1.194. Staff estimated this distance, which factored in the difference in elevation between the source and receptor, to be slightly more than 10 meters and generated χ/Q values using ARCON96. Thus, NRC staff has determined that the physical arrangement and separation of the Unit 2 ABNV Make-Up air Intake and the 122 CR intake is adequate to use ARCON96 for this specific case.

- Common Area of the Auxiliary Building

The licensee modeled postulated releases to the environment from the CAAB as diffuse releases through the CAAB walls. With regard to assuming a diffuse release through a building wall, RG 1.194 asserts, in part, that diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used.

The licensee discussed releases associated with the FHA, HLD, and MSLB in the Enclosure to the LAR. As described in supplemental submittals dated February 13, 2012 and February 24, 2012, the licensee provided additional detailed information regarding the Auxiliary Building and CAAB and the modeled release scenarios involving these buildings. The submittals also provided a description of the licensee's further assessment to identify potential release locations that could be more limiting than assuming a diffuse from the CAAB. This is discussed in Sections 2.1.2.2 and 2.1.2.3 of this SE.

In addition, in RAI 4 of the RAI dated November 2, 2011, NRC staff requested information regarding the Auxiliary Building Exhaust Stacks (ABES). It was not clear to the staff if there were any potential for releases from exhaust systems utilizing these stacks. The licensee responded in its letter dated February 24, 2012 that no post-accident radiological releases from the ABES are considered in the radiological consequence analysis because, at the initiation of a radiological event that affects the Auxiliary Building, the exhaust fans are tripped, fan discharge dampers are closed, and actuation of the ABSVS draws a negative pressure on the ABNVS. These functions occur automatically. The NRC staff finds the response an acceptable justification for not considering the ABES as potential post-accident radiological release points.

For calculation of χ/Q values associated with diffuse releases, the ARCON96 computer code uses two initial diffusion coefficients, σ_y and σ_z, entered by the user, to represent the diffuse source. RG 1.194 asserts that if the area source and the intake are on the same building surface such that wind flowing along the building surface would transport the release to the intake and if the included angle between the source-receptor line of sight and either the vertical or horizontal axis of the assumed source is less than 45 degrees, the initial dispersion coefficient will need to be adjusted. With regard to postulated releases from the CAAB, the licensee found that the included angle between the source-receptor line of sight and the vertical axis is less than 45 degrees for the CAAB to both the 121 CR and 122 CR Vent Intakes. Consistent with the intent of RG 1.194, for the case where the subtended angle is less than 45 degrees, the licensee set the value for σ_z to 0.0. The licensee noted that, for this case, while RG 1.194 indicates that σ_y should be set to 0.0, they believed this to be in error and, therefore, set σ_z to 0.0 to be consistent with the definition of σ_z.

The licensee modeled several scenarios with postulated diffuse releases to the environment through the CAAB walls. These scenarios were associated with an FHA in the spent fuel pool near the south end of the Auxiliary Building, an FHA inside of containment with a release to the Auxiliary Building near the mid-section of the Auxiliary Building, and a MSLB inside the Auxiliary Building, near the north end of the Auxiliary Building. Control room intakes 121 CR and 122 CR are located near the northeast and northwest ends of the CAAB, respectively.

- Fuel Handling and Heavy Load Drop Accidents

  In the current LAR, the licensee asserted that, for a postulated FHA or HLD associated with the SFP, the release from the pool would be to the SFP enclosure. A release from the SFP enclosure could occur through several possible locations depending on the ventilation system configuration: the Shield Building vent stack if the SFP Special Ventilation System (SFPSVS) is operating, the SFP Normal Vent Stack (SFPNVS) if the SFPNVS is operating, or the CAAB if no ventilation systems are operating. The licensee evaluated the three locations relative to the CR intakes and provided a comparative listing of the associated χ/Q values in Table 3.4-1 of the current LAR which identified the postulated release from the CAAB to be the limiting case.

  For a postulated FHA or HLD associated with the refueling pool inside of containment, the licensee asserted that the release from the pool would be to the containment. Such a release in the containment could result in releases to the environment from several possible locations: the Equipment Hatch, the Shield Building vent stack, or the CAAB. The licensee evaluated these locations relative to the CR intakes and provided a comparative listing of the associated χ/Q values in Table 3.4-1 of the current LAR which identified the postulated release from the CAAB to be the limiting case.

- MSLB Accident

  The licensee modeled the limiting release for the postulated MSLB as originating inside the Auxiliary Building and being transported to the CAAB, from which

effluent was assumed to be released as a diffuse source to the environment. The licensee provided additional detailed justification to address this scenario as described in the February 13 and 24, 2012, supplemental submittals and determined that the modeled release from the CAAB was the limiting case, which it summarized as follows:

- The dose analysis takes no credit for mixing of the effluent in the Auxiliary Building prior to its transit to the CAAB even though this mixing would be promoted by the steam release from the break.

- The area of the Auxiliary Building in which the MSLB release is postulated to occur is separated from the CAAB by four sets of double doors which have breakaway ceramic latch pins that allow the doors to open during a high energy line break, thus providing a pathway for Auxiliary Building pressure relief. A MSLB in the Auxiliary Building will pressurize all of the volumes and result in all four sets of doors opening from the Auxiliary Building into the CAAB. One set of doors is on the north side of the CAAB, the end nearer the 122 CR intake. As the licensee modeled the scenario, the steam release into the CAAB takes no credit for steam release into the CAAB through the other three sets of doors which would also open. These other doors are at the opposite side of the CAAB from 121 CR and 122 CR and would further promote mixing within the building.

- The steam flow path from the set of doors to the closest point of the CAAB to the 122 CR Vent Intake is not a likely direct flow path given the relative orientation of the door to this point of the building.

- The steam release to the CAAB would promote mixing within the building based on: (1) the relative orientation of the release into the building (i.e., the steam would flow towards the center of the building as it entered through the doors), and (2) the pressure and temperature differences of the incoming steam as compared to the environment in the CAAB.

- Based on building construction and the orientation of the steam flow path into the CAAB, there are no concerns with steam impingement causing a localized breach in the building.

- Should pressure build up in the CAAB, based on building design, pressure differential protection for the building (i.e., blow out panels) will result in openings at the south side of the CAAB; which have smaller χ/Q values than the assumed diffuse release location at the north end of the CAAB.

For postulated releases from the CAAB, NRC staff reviewed the figures and other information associated with the inputs and assumptions provided by the licensee and generated a sample set of χ/Q values using ARCON96 for comparison to the licensee's assessment.

Regarding the FHA and HLD, NRC staff review included a comparison of postulated releases from the Shield Building vent stack, the SFPNVS, and the Equipment Hatch in addition to the CAAB. The licensee modeled the FHA and HLD diffuse releases to occur from the closest points of the CAAB to the CR intakes although the locations in which a release might occur within the Auxiliary Building are from the containment maintenance airlocks in the mid-section or in the SFP area at far end of the Auxiliary Building. NRC staff has confirmed that the modeled diffuse release from the CAAB is more limiting than the other postulated FHA and HLD releases identified by the licensee.

NRC staff review of the MSLB included consideration of the detailed information provided by the licensee regarding the complex release scenario within the Auxiliary Building to transport effluent to and disperse effluent within the CAAB. The NRC staff has determined that the scenario modeled and described by the licensee is adequate to meet RG 1.194 diffuse release criteria. This is a specific case, based on factors such as the design of the PINGP Auxiliary Building and CAAB, arrangement and features of internal structures and components which would require the effluent to move along a circuitous path within the Auxiliary Building through design pressure relief doors to get to the CAAB, and factors generating further mixing, dilution and transport within the CAAB prior to release to the environment.

2.1.2.4.2.3 Control Room Atmospheric Dispersion Factors Conclusions

NRC staff reviewed the licensee's assessment of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. This included a review of the inputs and assumptions, which the staff found generally consistent with site configuration drawings and input tables, NRC staff practice, and RG 1.194 criteria, which includes consideration of scenario-specific cases. In addition, NRC staff generated sample comparative χ/Q value estimates and found the resultant χ/Q values to be similar to those calculated by the licensee for the cases considered.

On the basis of this review, the NRC staff has determined that the CR χ/Q values, as listed in Table 2.1-1 of this SE, which the licensee used to identify the scenarios resulting in the bounding doses are acceptable for use in the dose assessment for the current LAR. However, NRC staff notes that the conclusions related to the CR χ/Q values for the FHA, HLD and MSLB are based upon the specific scenarios and should not be interpreted to suggest that the CAAB can be considered as a diffuse source for other scenarios.

2.1.2.4.3 Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Atmospheric Dispersion Factors

For calculation of doses at the EAB and LPZ, the licensee input current licensing basis χ/Q values derived from the PINGP USAR, which were those used at initial facility licensing. These χ/Q values were based on guidance in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." They were discussed further in the SEs associated with PINGP, Unit 1 and Unit 2, License Amendments 166 and 156, respectively (ADAMS AN ML042430504), and Amendments 191 and 180, respectively (ADAMS AN ML091490611). As noted in these amendments, the licensee asserted that the dose assessment used guidance in RG 1.4, but did not assert that the current licensing basis EAB and LPZ χ/Q values were based upon RG 1.4. Further, NRC

staff had observed several differences in assumptions between the inputs used in the χ/Q calculations that appear in PINGP USAR and the RG 1.4 default guidance.

- With regard to the EAB dose assessment, RG 1.183 asserts that the maximum EAB total effective dose equivalent for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the applicable dose criteria. Therefore, the EAB χ/Q value should be representative of the limiting two hour time period.

- The shortest time interval for which χ/Q values are explicitly calculated in RG 1.4 is a 0-8 hour time period. Since the standard Gaussian short-term centerline atmospheric dispersion equation and inputs and assumptions discussed in the PINGP USAR are also suitable for a 0-2 hour time period, the NRC staff determined that it remains acceptable for the licensee to use the value calculated for the 0-8 hour time period as defined in RG 1.4 in the dose estimate for a 0-2 hour time period associated with the current LAR. However, NRC staff continues to note that if a different methodology or different assumptions were used, it may not be appropriate, in general, to apply χ/Q values calculated for a 0-8 hour time period to a 0-2 hour dose assessment.

- The current LAR also utilizes the LPZ χ/Q values used in the dose assessment that supports Amendments 191 and 180 for PINGP, Units 1 and 2, respectively. As noted in the SE associated with these Amendments, the licensee derived LPZ χ/Q values at a distance of 2,414 meters for time periods greater than 8 hours by interpolation from χ/Q values in Table XIV of the PINGP USAR, which lists distances ranging from 400 to 100,000 meters.

As part of the review associated with Amendments 191 and 180, NRC staff made a scoping comparison calculation using RG 1.145, 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 current staff practice. The resultant 0-2 hour EAB and 0-8 hour and 8-24 hour LPZ χ/Q values calculated by the NRC staff were lower than the licensing basis χ/Q values used by the licensee in its dose assessment, while the 1-4 and 4-30 day LPZ χ/Q values were higher than the licensing basis X/Q values used by the licensee.

The NRC staff qualitatively estimated the effect on the resultant dose estimates for the current LAR of the differences in the EAB and LPZ χ/Q values with respect to the assumed temporal release of effluent to the environment and determined that the 10 CFR 50.67 dose acceptance criteria would not be exceeded as a result of the χ/Q differences. While the NRC staff determined that the licensee's use of the PINGP licensing basis χ/Q values is acceptable for the current LAR, the NRC staff notes that this may not be the case for other dose applications and should be evaluated for each case. The PINGP current licensing basis EAB and LPZ χ/Q values are presented in Table 2.1-2 of this SE.

2.1.2.4.4 <u>Secondary Containment Drawdown – Meteorology</u>

Regulatory Guide 1.183, asserts, in part, that the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5

percent of the total number of hours in the data set. The licensee reviewed the 1993 through 1997 onsite meteorological measurements used to calculate the CR χ/Q values and determined that wind speed values of 16.5 mph at the 10-meter and 22.3 mph at the 60-meter measurement levels are exceeded less than 5 percent of the total number of hours in the data set.

NRC staff also reviewed the 1993 through 1997 meteorological data set and estimated that wind speed values exceeded less than 5 percent of the time were less than the values cited by the licensee. Thus, NRC staff has found the licensee's conclusion acceptable that wind speed values of 16.5 mph at the 10-meter and 22.3 mph at the 60-meter measurement levels were exceeded less than 5 percent of the time at PINGP.

2.1.2.5 Radiological Consequences of Design Basis Accidents

The licensee has proposed a licensing basis change for its offsite and CR DBA dose consequence analysis for PINGP, Units 1 and 2. The proposed change will implement an AST methodology for determining DBA offsite and CR doses. 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, 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:

- LOCA
- FHA
- HLD
- MSLB
- SGTR
- CREA
- LRA

Although the current licensed maximum reactor core power level for PINGP is 1677 MWt, the above analyses assume a maximum core power of 1852 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.

2.1.2.5.1 Loss-of-Coolant Accident (LOCA)

The LOCA event is assumed to be caused by an abrupt failure of a main reactor coolant pipe and the Emergency Core Cooling System (ECCS) fails to prevent the core from experiencing significant degradation. This sequence cannot occur unless there are multiple failures and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released into the containment and then to the environment by means of containment leakage and leakage from the ECCS.

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 of the chemical forms of radioiodine released from the RCS to the containment are 95 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic. Whereas, the radioiodine that is postulated to be available for release to the environment is assumed to be 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 environment
- ESF leakage outside containment

2.1.2.5.1.1 Containment Leakage

The licensee assumes the radioactivity released from the core is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released. The radioactivity release into the containment is assumed to terminate at the end of the early in-vessel phase, which occurs at 1.8 hours after the onset of a LOCA.

The licensee assumes that the PINGP containment is projected to leak at the proposed TS allowable leakage rate of 0.15 percent of its contents by weight per day for the first 24 hours and then 0.075 percent for the remainder of the 30-day accident duration. The NRC staff determined that this approach is consistent with the guidance in RG 1.183. The licensee indicated that the analysis does not credit containment spray for reducing the amount of radionuclides available for leakage from the containment. Containment leakage to the shield building (SB) is treated by the SBVS and released from the Shield Building vent stack. Containment leakage to the ABSVZ is treated by the Auxiliary Building Special Ventilation System (ABSVS) and released from the Shield Building vent stack.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. The licensee credited an elemental iodine natural deposition removal coefficient of 2.999 hr^{-1} until an elemental iodine DF of 200 is reached at three hours post-LOCA. Elemental

iodine natural deposition is not credited after three hours. As a result of this removal mechanism, a large fraction of the released activity will be deposited in the containment sump.

The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above seven. The licensee states that the pH is above seen for the duration of the analysis period. The NRC staff finds that the licensee followed the guidance in RG 1.183, and, therefore, finds the licensee's approach acceptable.

2.1.2.5.1.2 Engineered Safety Features (ESF) Leakage

The ESF systems that recirculate containment sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components in the ECCS located in the ABSVZ.

ECCS leakage can occur from the following potential sources:

- Leakage from the ECCS pump seals, valve packing, mechanical joints, etc., and
- Backleakage to the RWST through valve seats.

Leakage rates through the ECCS are controlled by periodic surveillance procedures that establish maximum leakage limitations. The maximum permitted total leakage from both trains of ECCS is 2 gallons per hour (gph). Per guidance in RG 1.183, the licensee doubled the engineered safety feature (ESF) leakage of 2 gph to 4 gph, and then used the different temperature-dependent iodine flashing factors to determine the ESF leakage iodine flashing rates. The RWST backleakage of 5 gph is also doubled in the analysis. The NRC staff finds these assumptions to be conservative.

For determination of the dose contribution from ESF leakage, all radionuclides assumed to be released from the core (except noble gases) are assumed to be instantaneously and homogeneously mixed in the containment sump. Actual leakage from the RCS sump through ESF equipment would not start until after ESF leakage is assumed to begin. For the backleakage to the RWST, the leakage needs to transit long lengths of liquid-filled piping prior to reaching the RWST. A conservative transit time of 35 hours is used before the leakage reaches the RWST. Leakages from the ECCS and backleakage to the RWST are assumed to continue for the full 30 day duration of the accident.

The fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of leakage that flashes to vapor. According to RG 1.183, if the temperature of the leakage exceeds 212 °F, the fraction of total iodine in the liquid that becomes airborne is assumed equal to the fraction of liquid that flashes to vapor. The licensee determined the flash fraction using a constant enthalpy process based on the maximum time-dependent temperature of the sump liquid circulating outside of containment.

The licensee's calculation shows that after 8.33 hours, the sump liquid temperature decreases below 212 °F. Based on the constant enthalpy equation, the calculated flash fraction at

212.3 °F is 0.1 percent. RG 1.183, Appendix A, Item 5.5, states that, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates." Because the temperature of the leakage is less than 212 °F, RG 1.183 Appendix A, Item 5.5, provides the option of a static model that releases 10 percent of the iodine activity during the accident, or it allows for the potential for the application of smaller partitioning factors based upon a technically justifiable plant-specific model. The licensee chose the latter approach.

The licensee performed an evaluation based on the plant-specific post-LOCA sump water temperature. The licensee showed that an ESF leakage flashing fraction of 3 percent is conservative because it is at least 6 times greater than the analytically derived average limit of 0.48 percent and 2.68 times greater than the experimentally measured limit of 1.12 percent. The NRC staff concludes that this approach is conservative and therefore, acceptable.

2.1.2.5.1.3 Control Room Doses

In the licensee's submittal the CR envelope is described as consisting of the CR and the two mechanical equipment rooms. The CR ventilation system is entirely located within the two mechanical equipment rooms, with the exception of the outside air supply. The outside air supply dampers are located at the envelope boundary. There are no other ventilation systems that penetrate the CR envelope.

The licensee indicates that during normal operation one train is running and the other train is in standby. For the operating train, the air handler is operating and the clean-up fan would be in standby with no air flow through the Particulate, Absolute, Charcoal (PAC) filter. During normal operation, the operating train recirculates the CR air and draws in fresh air. This recirculation flow during normal system operation is not filtered. Air is exhausted from the CR Envelope at a rate equivalent to the quantity of fresh air brought in. The design flow rates are 10,000 cfm recirculation flow rate and 2000 cfm fresh air for a total air handler flow rate of 12,000 cfm. For the LOCA dose analysis, the licensee did not use the normal recirculation flow rate. The normal recirculation flow is not modeled as it has no impact to the dose analysis. The input for normal operation in the dose analysis is the fresh air supply flow rate. An equivalent input is also included for the exhaust flow rate.

In its submittal, the licensee provided information about PINGP's emergency mode. PINGP's emergency mode alignment is initiated either by a safety injection (SI) signal or a high radiation signal in the CR. The SI signal is almost immediate for the design-basis LOCA. A "high" signal will automatically switch the CR ventilation system from the normal mode of operation to the emergency mode. In response to SI or high radiation signal, both trains of the CRSVS start and are automatically aligned to isolate the fresh air intake and exhaust, and start and align a portion of the recirculation air flow through the clean-up fan. The portion of the air that is drawn by the clean-up fan passes through a PAC filter that is credited by the licensee in the dose analysis. In this alignment, the system is recirculating and filtering the CR atmosphere. To account for a single active failure, only one train of CR ventilation system is credited in the dose analysis. In the emergency mode, the clean-up fan is designed to provide 4000 cfm ± 10 percent. For the AST dose analysis, the NRC staff determined that the licensee used the lower bound for filtered air flow rate, 3600 cfm.

For the purposes of determining radiological consequences in the AST dose analysis, the licensee assumed that all inleakage to the CR is drawn in through the CR outside air intake. In order to be drawn in through the outside air intake, the licensee determined that the release needs to leak by the redundant bubble tight dampers.

According to its submittal, the licensee calculated the air immersion and inhalation dose to a hypothetical individual in the CR based on the time-integrated concentration in the CR compartment in RADTRAD 3.03. The Murphy-Campe geometric factor, GF, relates the dose from an infinite cloud to the dose from a cloud of volume as described in Section 2.3.2 of NUREG/CR-6604.

The gamma shine dose contribution to CR personnel consists of the following parts:

- Gamma shine from containment airborne activity
- Gamma shine from activity in the external radioactive cloud surrounding the plant structures
- Gamma shine from trapped activity on filters

The post-LOCA containment leakage activity confined in the containment dome air space above the operating floor can contribute to the direct shine dose to the CR operator. The licensee describes the containment steel cylinder wall as being 1.5" thick and the SB concrete wall is 2'-6" thick. The CR concrete ceiling and Auxiliary Building roof are 2'-0" and 1'-0" thick respectively. The direct shine from the post-LOCA activity confined above the containment operating floor to the CR operator encounters a slant line-of-sight path through the 1.5" of the containment steel, 2.5' of the SB concrete wall, 3.0' of the CR ceiling plus Auxiliary Building roof, and multiple other concrete roof and walls of the auxiliary building. The encountered concrete and steel barriers provide shielding for the post-LOCA containment shine dose to the CR operator. The licensee concluded that this component of the CR operator dose can therefore be ignored. Based on the discussion above and engineering judgment, the NRC staff determined that there is adequate shielding of the post-LOCA containment shine dose to the CR operator. Therefore, the NRC staff agrees with the licensee's assessment of this streaming source and finds it acceptable.

The licensee asserts that the post-LOCA radioactive plumes released from the SB vent stack and auxiliary building air intake louvers contain the radioactive sources from the containment and ESF leakages. The licensee determined the RWST leakage dose contribution is extremely small; therefore, the radiation shine from the RWST radioactive plume is excluded from this analysis. The CR concrete shielding (walls and ceiling) directly exposed to the environment is reviewed to determine the minimum thickness of concrete shielding between the external cloud and the CR personnel. The CR walls are within the auxiliary building and turbine building and are further surrounded by other concrete walls in the auxiliary building and turbine building. The ceiling of the CR is 24" thick concrete, which is directly exposed to the radioactive cloud external to the CRE.

The licensee indicates that the CRE will be submerged in the post-LOCA radioactive plume and the CR operator will be exposed to gamma dose through the walls and roof. The resulting cloud shine dose to a 6-foot tall CR operator standing on the floor of the shift supervisor's office provides the limiting location in the CR. The licensee used RADTRAD 3.03 code to calculate

the site boundary whole body gamma dose based on the semi-infinite cloud immersion. Therefore, the X/Qs for the LPZ receptor modeled in RADTRAD model are modified by replacing them with the CR intake X/Qs. The semi-infinite gamma dose calculated at the modified LPZ receptor actually represents the dose at the CR air intake location. The NRC staff finds that using the CR air intake X/Qs in the model to represent the center of the CR is conservative because the CR air intake locations are closer to the leakage release points than the CR center.

According to the licensee's submittal, the CRSVS charcoal filter trains are located in the Mechanical Equipment Rooms above the CR. The RADTRAD 3.03 code calculates the cumulative elemental and organic iodines and the aerosol mass are assumed to deposit on the CRSVS recirculation charcoal/HEPA filters. The CR recirculation filter iodine and aerosol activities are calculated for the containment leakage. The relationship between the iodine and aerosol mass and activity are established based on the information obtained from RADTRAD model. Using the assumptions from RG 1.183, the licensee asserts that post-LOCA ESF leakage consists of a non-aerosol iodine release (97 percent of the elemental iodine and 3 percent of the organic iodide). Therefore, there is no aerosol mass deposited on the CRSVS HEPA filter. The total iodine (elemental and organic) is deposited on the CR charcoal filter. The total iodine release due to the containment and ESF leakages are combined and then converted into the isotopic iodine activities. Similarly, the total aerosol mass is deposited on the CRSVS recirculation filter and converted into aerosol isotopic activities.

The licensee developed a MicroShield geometric model based the location of the charcoal filter, charcoal tray dimensions, and by positioning a 6 feet tall CR operator just below the charcoal bed in the CR supervisor office in the CR. Then NRC staff finds this positioning to be conservative because it is the most limiting position. The charcoal trays are modeled with approximate dimensions of 24" (H) x 48" (W) x 30" (L) and placed on the CR ceiling with the CR operator standing right below the center of charcoal bed. The CRSVS charcoal filter shine dose to the CR operator is calculated using the the MicroShield computer code. The 720-hrs direct dose from the CR filter shine is calculated using the CR occupancy factors and added to doses from other post-LOCA sources. Based on the above discussion, the NRC staff finds that the licensee used conservative assumptions for the calculation of the gamma shine contribution from trapped activity on CR charcoal and high-efficiency particulate air (HEPA) filters. Therefore, the NRC staff agrees with the licensee's modeling and finds the licensee's assessment of this streaming source acceptable.

2.1.2.5.1.4 Technical Support Center (TSC) Dose Consequence Assessment

As described in the licensee's submittal, during normal heating, ventilation, and air conditioning (HVAC) system operation, the system is operating on recirculation using the return fans. Fresh air is supplied through the normal outside air intakes. The recirculation flow path using the return fans is not filtered. Air is exhausted from the TSC at a rate equivalent to the quantity of fresh air supplied. There is no air flow through the clean up and PAC filters. The design outside air supply flow is 1000 cfm. An equivalent value is used for the exhaust air flow rate.

The licensee indicates that the system is manually realigned for the emergency mode. In the emergency mode, the return fans are shut off, the normal outside air dampers are closed, the clean-up fan is started and the emergency outside air supply damper modulates to provide a preset flow rate to maintain a positive pressure in the TSC. The recirculation and emergency supply air is filtered through a PAC filter. The TSC HVAC model schematic uses a single filtered outside air intake and a filtered recirculation flow which mixes with part of the recirculating air in the TSC. Unfiltered inleakage and filtered make-up air are assumed to be drawn into the TSC outside air intake. No filtering is applied to the unfiltered inleakage.

In its submittal, the licensee describes how the TSC 30-day inhalation and immersion dose was analyzed for the LOCA and each of the other DBAs. In examining their post-accident TSC dose consequences, the licensee finds that the 30-day doses do not exceed 5 rem TEDE. The NRC staff reviewed the licensee's assumptions and results and determined that the licensee followed the regulatory requirements of Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 and the regulatory guidance of NUREG-0737, "Clarification of TMI Action Plan Requirements". Therefore, the NRC staff finds the licensee's analysis of the TSC radiological dose to be acceptable.

2.1.2.5.1.5 LOCA 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 CR 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 staff relied upon information provided by the licensee and staff experience in performing similar reviews. 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 2.1-3 of this SE.

2.1.2.5.2 Fuel Handling Accident (FHA)

The FHA analysis postulates that a spent fuel assembly is dropped and damaged during fuel handling. This accident may take place either in the containment or in the spent fuel pool (SFP). As discussed in its submittal, the licensee does not take credit for containment isolation or filtration by the SFP special ventilation system (SFPSVS) in its analysis. Based on confirmatory calculations, the NRC staff determined that the licensee has chosen the analysis inputs and assumptions such that the results of the single FHA analysis are bounding for the accident occurring in either the containment or the SFP.

The entire gap activity from the damaged assembly is assumed by the licensee to be released directly to the outside atmosphere at a constant rate over a 2-hour period. The licensee calculated the activity in the gap of the fuel rods assuming the assembly has been operated at 110 percent of the maximum core thermal power times a radial peaking factor of 1.9, and the fuel has undergone radioactive decay for 50 hours. The analysis assumed the RG 1.183, Table 3, non-LOCA gap fractions. Although the licensee could not ensure that the fuel would meet the footnote 11 maximum linear heat generation rate limitation, the licensee chose to follow the footnote's alternative to calculate fission gas release fractions with an NRC-approved methodology. The NRC finds the licensee's approach acceptable because it is consistent with

the guidance in RG 1.183.

To calculate the gap release fraction for AST, the NRC staff considers the use of approved methodologies and bounding power histories to be acceptable. The NRC staff endorses (via NUREG/CR-7003, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model") the American National Standards Institute/American Nuclear Society-5.4-1982 model, entitled "American National Standard Method for Calculating the Fractional Release of Volatile Fission Product from Oxide Fuel," as an acceptable gap fractional release model. Based on this model, the licensee developed a computer code, GAP, to perform the gap release fraction calculation. The GAP code was found to be acceptable to the NRC staff by License Amendment Nos. 166 and 156 to Facility Operating License Nos. DPR-42 and DPR-60 for PINGP, Units 1 and 2, respectively (ADAMS AN. ML042430504). The NRC staff determined that the analytical approach is consistent with the ANSI/ANS-5.4-1982 model, and that the GAP code is acceptable for analyzing the gap release fraction.

The licensee assumed the iodine species released from the fuel gap to the water was 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee assumed that the depth of water above the damaged fuel is 23 feet, therefore, the effective iodine decontamination factor (DF) for the water pool is 200. These assumptions are consistent with the guidance in RG 1.183.

The licensee assumed that the CR heating, ventilation, and air conditioning system was initially in the normal operation mode. Within a few seconds, the activity level in the CR would cause a high radiation signal, which in turn activates the isolation of the CR envelope and filtration of recirculated air. The licensee's analysis assumed this isolation and recirculation occurred 5 minutes after the FHA, which takes into account delays in radiation detection and subsequent isolation. The licensee assumed 300 cfm of unfiltered inleakage into the CR envelope during the emergency mode of operation. This assumption bounds the results of tracer gas testing of the CR envelope inleakage that was performed by the licensee in January 1998, which was done in accordance with Generic Letter (GL) 2003-01, "Control Room Habitability."

Although the licensee's analysis does not take credit for filtration by the SFPSVS or SBVS, the systems are not prevented from operating after a FHA. The licensee is proposing to remove the SFPSVS and the charcoal adsorbers testing from the PINGP TS. The revised FHA analysis uses the most limiting X/Q for the common area of auxiliary building without crediting any charcoal filtration. The NRC staff finds that the calculated dose consequences for the post-FHA unfiltered releases from the SFP and containment will be below the regulatory dose criteria. The NRC staff finds the licensee's discussion technically sound with respect to the radiological consequences of a FHA.

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 CR are within the dose criteria specified in 10 CFR 50.67 and accident-specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee and staff experience in performing similar reviews. The 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 2.1-4 of this SE.

2.1.2.5.3 Heavy Load Drop Accident (HLD)

In the postulated HLD, the reactor vessel upper internals are assumed by the licensee to be dropped into the reactor vessel, damaging all fuel rods in two impacted fuel assemblies. This accident would take place in the containment. The postulated drop of the reactor vessel head and the upper internals were both examined. The limiting case was determined to be the drop of the upper internals. The licensee does not take credit for filtration by the SBVS filter in its analysis and has chosen the analysis inputs and assumptions such that the results of the HLD analysis are bounding for the accident occurring in the containment.

For the HLD, the accident is postulated to occur at seven days after shutdown. For the first seven days after shutdown, movement of a heavy load over an open reactor vessel, with irradiated fuel in the vessel, is not allowed with containment open. Following accident initiation at seven days after shutdown, the radionuclide inventory from the damaged fuel pins is assumed to be released over a 2-hour time period. Because the FHA and the HLD are similar accidents, the NRC staff finds the release period assumption to be acceptable, because it is consistent with the assumptions described in RG 1.183. The licensee calculated the activity in the gap of the fuel rods assuming the assembly has been operated at 110 percent of the maximum core thermal power times a radial peaking factor of 1.9. The analysis assumed the RG 1.183 Table 3 non-LOCA fractions. Although the licensee could not ensure that the fuel would meet the footnote 11 maximum linear heat generation rate limitation, the licensee chose to follow the footnote's alternative to calculate fission gas release fractions with an NRC-approved methodology. The GAP code was found to be acceptable to the NRC staff by License Amendment Nos. 166 and 156 to Facility Operating License Nos. DPR-42 and DPR-60 for PINGP, Units 1 and 2, respectively (ADAMS AN ML042430504). The NRC staff determined that the analytical approach is consistent with the ANSI/ANS-5.4-1982 model, and that the GAP code is acceptable for analyzing the gap release fraction.

The licensee assumed the iodine species released from the fuel gap to the water was 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee assumed that the depth of water above the damaged fuel is 23 feet, therefore, the effective iodine DF for the water pool is 200. These assumptions are consistent with the guidance in RG 1.183.

The licensee assumed that the CR HVAC system was initially in the normal operation mode. Within a few seconds, the activity level in the CR would cause a high radiation signal, which in turn activates the isolation of the CR envelope and filtration of recirculated air. The licensee's analysis assumed this isolation and recirculation occurred 5 minutes after the FHA, which takes into account for delays in radiation detection and subsequent isolation. The licensee assumed 300 cfm of unfiltered inleakage into the CR envelope during the emergency mode of operation. This assumption bounds the results of tracer gas testing of the CR envelope inleakage that was performed by the licensee in January 1998, which was done in accordance with GL 2003-01.

Although the licensee's analysis does not take credit for filtration by the SBVS, the system is not prevented from operating after a HLD. The licensee is proposing to remove the charcoal adsorbers testing from PINGP TS 5.5.9. The HLD analysis uses the most limiting X/Q for the common area of auxiliary building without crediting any charcoal filtration. The NRC staff finds that the calculated dose consequences for the post-FHA unfiltered releases from the SFP and

containment will be below the regulatory dose criteria. The NRC staff finds the licensee's discussion technically sound with respect to the radiological consequences of a HLD.

The licensee evaluated the radiological consequences resulting from the postulated HLD using the AST and concluded that the radiological consequences at the EAB, LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee and staff experience in performing similar reviews. The 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 HLD analysis assumptions and parameters can be found in Table 2.1-5 of this SE.

2.1.2.5.4 MSLB

In the postulated MLSB accident, the licensee considers the complete severance of a main steam line outside the containment but upstream of the main stean isolation valves (MSIV). Upon a MSLB, the affected steam generator (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 licensee assumes that the secondary water in the affected SG completely flashes to steam within 10 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 value of 30 µCi/gm Dose Equivalent I-131. The accident initiated iodine spike case assumes the primary coolant activity was initially at the proposed TS limit of 0.5 µCi/gm Dose Equivalent I-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 eight 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 150 gallons per day (gpd) for the intact SG and 1.0 gpm for the faulted SG. The leakage to the affected SG is conservatively assumed to immediately flash to steam and be released to the environment without holdup or dilution. The leakage in the unaffected SG mixes with secondary water and is released at the assumed steaming rate of the intact SG. The licensee assumes that within 75 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 0.1 in the unaffected SG which is more conservative than the value of 0.01 from RG 1.183. All noble gas activity carried over to the secondary side through SG tube leakage is conservatively assumed to be immediately released to the outside atmosphere.

No fuel damage is predicted for the MSLB. Activity release due to iodine spiking is release homogeneously throughout the primary coolant. Iodine releases from SGs to the environment is assumed to be 97 percent elemental and 3 percent organic. These assumptions are consistent with the guidance in RG 1.183.

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 CR are within the dose criteria specified in 10 CFR 50.67 and accident-specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee and staff experience in performing similar reviews. 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 2.1-6 of this SE.

2.1.2.5.5 Steam Generator Tube Rupture (SGTR)

As discussed in the licensee's submittal, for this postulated accident a double-ended rupture of a single SG tube is assumed to occur by the licensee. At the start of the accident, radionuclides from the primary coolant are assumed to enter the SG via the ruptured tube and primary-to-secondary leakage. The radionuclides are then conservatively assumed to release into the condenser prior to reactor trip. The primary-to-secondary break flow results in depressurization of the RCS. Reactor trip and safety injection (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 through the associated PORV/safety valves.

In accordance with RG 1.183 guidance, two iodine spiking cases were considered. The first is a pre-accident iodine spike case. The pre-accident iodine spike case assumes that a reactor transient occurs prior to the SGTR and has raised the primary coolant iodine concentration to a conservative value of 30 µCi/gm Dose Equivalent I-131. The primary-to-secondary coolant leakage of 150 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 associated SG PORV. The licensee assumes an iodine partition coefficient of 100 in the intact SG in accordance with RG 1.183 guidance. One hundred percent of the noble gases are conservatively 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, while the remainder mixes with secondary coolant for subsequent steaming through the PORV. The licensee assumes that the steam release from the intact SG continues for 14 hours until the residual heat removal (RHR) system is initiated.

The second case considered is an accident initiated iodine spike case. The accident initiated iodine spike case assumes the primary coolant activity was initially at the proposed TS limit of 0.5 µCi/gm Dose Equivalent I-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. These assumptions are in accordance with the guidance in RG 1.183.

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 CR are within the dose criteria specified in 10 CFR 50.67 and accident specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In

performing this review, the NRC staff relied upon information provided by the licensee and staff experience in performing similar reviews. 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 assumption and parameters can be found in Table 2.1-7 of this SE. The SGTR event is further evaluated in Section 2.6 of this SE.

2.1.2.5.6 Control Rod Ejection Accident (CREA)

As discussed in the licensee's submittal, the CREA is postulated by the licensee as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod control assembly and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. It is assumed that prior to the postulated accident the plant is operating at an equilibrium level of radioactivity in the primary and secondary systems as a result of coincident fuel defects and SG tube leakage.

Following a postulated CREA, two activity release paths contribute to the total radiological consequences of the accident. The first release path is via containment leakage resulting from release of activity from the primary coolant to the containment. The release path from containment is the same as that modeled for the post-LOCA containment release in Section 2.1.3.1 of this SE. The second path is assumed to be from the contribution of steam in the secondary system dumped through the SG PORVs and Turbine Driven Auxiliary Feedwater Pump steam exhaust, since offsite power is assumed to be lost.

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 are in the fuel gap, consistent with the guidance provided in RG 1.183. A radial peaking factor of 1.9 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 25 percent of the iodine 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 contained in the melted fuel are released to the secondary. These assumptions are consistent with RG 1.183.

For the containment leakage case, the ejected control rod is conservatively assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break LOCA. All activity from damaged fuel that has been mixed with the primary coolant of the RCS is assumed to leak directly to the containment volume. This flashed release is assumed to instantaneously and homogeneously mix with the containment atmosphere and subsequently be available for release to the environment via the assumed containment leak rate limit as used for the LOCA analysis. Credit for mitigation of the release by containment spray is not taken. No credit is taken for plateout of iodine inside of containment. These assumptions are consistent with the guidance in RG 1.183.

For the secondary release case, no breach of the RPV is assumed following the rod ejection. In this case, RCS integrity is maintained and all activity from damaged fuel that has been mixed with the RCS leaks to the secondary side coolant through the SG tubes via the primary-to-secondary coolant leakage of 150 gpd per SG. Activity is available for release from the SGs to the environment by steaming of the SG PORVs. The licensee assumed that the

chemical form of the radioiodine released to the environment would be 97 percent elemental and 3 percent organic which is 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 the 1 percent of the radionuclides that enter the unaffected SG from the RCS enter the vapor space and are released to the environment. This assumption is also consistent with the guideline provided in RG 1.183. Steam releases from the SG are assumed by the licensee to continue for 45.5 hours.

The licensee evaluated the radiological consequences resulting from the postulated CREA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident-specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee and staff experience in performing similar reviews. 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 CREA analysis assumption and parameters can be found in Table 2.1-8 of this SE.

2.1.2.5.7 Locked Rotor Accident (LRA)

As discussed in the licensee's submittal for the LRA, an instantaneous seizure of a reactor coolant pump (RCP) rotor is assumed to occur by the licensee. The reactor is tripped on the subsequent low flow signal. Following the trip, heat stored in fuel rods continues to pass into the reactor coolant, causing the coolant to expand. The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the SGs, causes an insurgence of coolant into the pressurizer and a pressure increase throughout the RCS. The insurgence of coolant into the pressurizer compresses the steam volume, which in turn actuates the automatic spray system, opens the pressurizer PORVs, and also opens the pressurizer safety valves.

The instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel damage. The dose analysis for this event assumes 20 percent fuel damage. Therefore, the source term available for release is associated with this fraction of damaged fuel and the fraction of core activity existing in the gap, plus the iodine in the RCS due to a design basis pre-accident iodine and noble gas activity associated with assumed 1 percent fuel defects. The 20 percent clad damage is the acceptance criteria for the PINGP Locked RCP Rotor transient analysis according to PINGP's current licensing basis. Therefore, the NRC staff finds that using the transient analysis acceptance criteria as the input for the radiological dose consequence analysis is acceptable.

A radial peaking factor of 1.9 is applied. The radionuclides released from the fuel are conservatively assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage of 150 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 and is directly released to the environment in proportion to the steam release rate, with credit taken for iodine partitioning in the SG liquid. For the secondary liquid iodine release from the SGs, an iodine partition coefficient of 100 is assumed which is consistent with RG 1.183. The secondary coolant iodine release continues until the residual heat removal system can be used to complete the cooldown at approximately 45.5 hours.

The licensee evaluated the radiological consequences resulting from the postulated LRA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident-specific dose criteria described in SRP Section 15.0.1. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee, the NRC staff experience in performing similar reviews, and, where deemed necessary, confirmatory calculation. The 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 LRA analysis assumption and parameters can be found in Table 2.1-9 of this SE.

2.1.3 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 PINGP, Units 1 and 2. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.1.1 above. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.1.1. The licensee's limiting calculated DBA dose results are summarized in Table 2.1-11. 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 staff further finds reasonable assurance that PINGP, 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 PINGP, 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, Section 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 PINGP, 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 design basis accidents using new source terms. The licensee performed these tasks by following the requirements of 10 CFR, Section 50.67, "Accident source term." It also applied for a license amendment under 10 CFR,, Section 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 that could be released to the containment from the damaged core following an accident. As a

result of improved understanding of the mechanisms of the release of radioactivity, 10 CFR 50.67 permits licensees to voluntarily replace their current Technical Information Document (TID) 14844 accident source term with the AST. However, this replacement is subject to performing a successful re-evaluation of the major design basis accidents. 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 suppression pool pH ≥7 for 30 days following a LOCA. According to RG 1.183, maintaining pH basic will minimize re-evolution of iodine from the suppression pool water.

2.2.2 Technical Evaluation

After a LOCA, a variety of different chemical species are released from the damaged core. One of them is radioactive iodine. This iodine, when released to the outside environment, will significantly contribute to radiation doses. It is, therefore, essential to keep it confined within the plant's containment. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," iodine is released from the core in three different chemical forms; at least 95 percent is released in ionic form as cesium iodide (CsI) and the remaining 5 percent as elemental iodine (I_2) and hydriodic acid (HI); the release contains at least 1 percent of each I_2 and HI. CsI and HI are ionized in water and are, therefore, soluble. However, elemental iodine is scarcely soluble. To sequester the iodine in water, it is desirable to maintain as much as possible of the released iodine in ionic form. In radiation environments existing in containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At a higher pH, conversion to elemental form is lower and at pH >7 it becomes negligibly small. The relationship between the rate of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control."

PINGP, Units 1 and 2, uses sodium hydroxide to buffer post-LOCA sump pH. The sodium hydroxide is added through the containment spray system. Prairie Island TSs require a minimum volume of 2590 gallons of sodium hydroxide at a concentration of 9 percent be available for injection. This minimum available quantity of sodium hydroxide was used in the licensee's calculations of the minimum sump pH following a LOCA. The post-LOCA sump pH calculations were performed using the MAAP-AST computer code. The code calculates the sump pH as a function of time based on specific fluids contributing to the sump volume.

The licensee provided information regarding the assumptions and calculations used to verify that the sump pH would remain greater than 7.0 following a LOCA. The licensee's analysis considered maximum boron concentrations and volumes for the refueling water storage tank, accumulator, and reactor coolant system. As previously stated, the quantity of sodium hydroxide was limited to the minimum allowed by TS. This combination of inputs maximizes the acid and minimizes the base available to impact the sump pH, thus establishing the minimum pH case to ensure that the pH does not drop below 7.0. Additional inputs to the calculation included the impact of strong acids generated by radiation of cable insulation and sump water.

Strong acids generated from cable insulation and sump fluids under accident radiation levels were inputs to the licensee's pH calculation. Radiolysis of chloride-bearing cable insulation results in the generation of hydrochloric acid. The licensee used plant databases to determine the quantity of cable material in containment. The quantity of cable material was then increased

by more than 100 percent to add conservatism and allow margin for any cables that may be added in the future. The licensee assumed that the entire mass of cable insulation was chloride-bearing and that it was all subject to radiolysis. This is noted as a conservatism in the calculation since many cables will have non-chlorinated insulation material. In addition, many cables have chlorinated jackets only, with non-chlorinated conductor insulation. The licensee assumed that 10,000 pounds of cable material was available to contribute to acid generation. In addition to hydrochloric acid generated by cables, radiolysis of the sump water results in the generation of nitric acid. The maximum possible water volume was used to calculate nitric acid generation. The NRC staff finds the licensee's calculations for strong acid generation acceptable.

The minimum pH evaluation used the maximum borated water source volumes and concentrations and the contribution of acid from radiolysis of cables and sump fluid to determine the minimum sodium hydroxide mass needed to ensure an equilibrium sump pH greater than 7.0. The calculation determined that existing TSs required quantity of sodium hydroxide would be sufficient to maintain pH greater than 7.0.

The NRC staff reviewed the licensee's assumptions and analysis and concluded that conservative values were used for the key parameters of the calculation. In addition, the NRC staff verified that the proposed TS requirements for sodium hydroxide volume and concentration will ensure sufficient buffering of the sump pool such that the pH will not drop below 7.0.

2.2.3 Conclusion

The NRC staff reviewed the licensee's assumptions to minimize iodine re-evolution as presented in the re-analysis of the radiological consequences for a LOCA. The methodology relies on using buffering action of sodium hydroxide. The assumptions are appropriate and consistent with the methods accepted by the staff for the calculation of post-accident containment sump pH. In addition, the staff verified that the post-accident containment sump pH will be maintained above 7.0 for 30 days following a LOCA.

2.3 Electrical Engineering

2.3.1 Regulatory Evaluation

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

In 10 CFR Appendix A of Part 50, General Design Criterion (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.

In 10 CFR Appendix A of Part 50, 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.

In 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.

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

In 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 licensees may use either the AST or the TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," assumptions for performing the required environmental qualification 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 environmental qualification doses.

2.3.2 Technical Evaluation

The NRC staff reviewed the electrical and environmental qualification portions of the LAR.

Based on its review of the LAR, the NRC staff requested additional information regarding whether electrical non-safety related systems were credited in the AST analyses to assess their design compliance with GDCs 17 and 18. The licensee stated in its April 29, 2010, letter that the AST does credit nonsafety-related exhaust dampers, in the auxiliary building ventilation system which are closed by spring force when electrically actuated. However, the exhaust damper electrical controls are part of the safety-related ABSVS with redundant circuits. No other nonsafety-related electrical components are credited. The NRC staff finds the electrical design acceptable.

The NRC staff further requested additional information on whether there would be any change in loading on the PINGP emergency diesel generators (EDGs) and if so, confirm adequate margin with respect to EDG loading and that EDG testing envelopes loading requirements post-AST in accordance with GDCs 17 and 18. In its April 29, 2010, letter the licensee stated that no loads were added to the PINGP EDGs, either automatically sequenced or manually based on procedures, as a result of the AST adoption. The NRC staff finds this design acceptable.

The NRC staff requested that the licensee confirm that there are no changes to the environmental qualification program (EQ/10 CFR 50.49) as a result of the AST adoption. In its

April 29, 2010, letter, the licensee stated that no components were added to the PINGP 10 CFR 50.49 program and the analyses do not result in a change to the post accident temperature and pressure profiles in containment, auxiliary building or turbine building as a result of the AST adoption. In addition, the analyses do not result in a change in the chemical environmental parameters inside containment used for EQ.
The NRC staff also reviewed the EQ portion of the LAR. As stated in its April 29, 2010, letter, the licensee used the methodology contained in TID 14844 to determine the radiation doses in the existing environmental qualification analyses. As mentioned previously, the use of this methodology is consistent with the guidance contained in RG 1.183. Since the licensee will continue to use the TID 14844 methodology and no new equipment is added to its 10 CFR 50.49 program, the environmental qualification of equipment should remain bounding during full-scope implementation of an AST. The NRC staff finds this design acceptable.

2.3.3 Conclusion

Based on the information provided by the licensee and the evaluation above, the NRC staff finds the proposed revision to the PINGP licensing bases provides reasonable assurance that an AST can be implemented at PINGP, Units 1 and 2, in accordance with 10 CFR 50.67 and RG 1.183. The staff also concludes that the proposed changes are in accordance with 10 CFR 50.49, and the requirements of GDCs 17 and 18. 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 requires that structures, systems, and components (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 Part 100, 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 qualification and seismic interactions, of SSCs such as the ABSVS and portions of the ABNVS which are credited in the implementation of the AST at PINGP.

The licensee received construction permits prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. Section 1.5, "General Design Criteria," of the PINGP USAR states that PINGP was designed and constructed to comply with the licensee's understanding of the intent of the Atomic Energy Commission General Design Criteria (AEC GDC or proposed GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. As such, the NRC staff's evaluation within the purview of the mechanical and civil engineering aspects of this LAR considered 10 CFR 50.55a and AEC GDC 1 and 2. However, as noted in the PINGP USAR, the AEC staff's safety evaluation report (SER) acknowledges that the AEC staff assessed the PINGP final safety analysis report (FSAR) against the Appendix A GDC and "... [were] satisfied that the plant design generally conforms to the intent of these criteria."

The NRC staff's evaluation considered 10 CFR 50.55a, AEC GDC 1 and AEC GDC 2. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of affected SSCs under postulated accident conditions, as analyzed with the implementation of an AST at PINGP. As indicated above, the acceptance criterion are based on the continued conformance with the requirements of 10 CFR 50.55a, and AEC 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 AEC GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of 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, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." With respect to the mechanical and civil engineering aspects of the AST implementation, RG 1.183 indicates that licensees must evaluate the 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 at PINGP. As described in RG 1.183, a full scope AST implementation refers to the licensee's request to re-calculate the dose consequences of select 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 or NUREG-0800.

The NRC recently issued similar AST implementation license amendments for the St. Lucie Nuclear Plant, Unit 1, on November 26, 2008 (ADAMS AN ML082682060), Edwin I. Hatch Nuclear Plant, Units 1 and 2, on August 28, 2008 (ADAMS AN ML081770075), South Texas Project, Units 1 and 2, on March 6, 2008 (ADAMS AN ML080160013), and Salem Nuclear Generating Station, Units 1 and 2, on February 17, 2006, (ADAMS AN ML060040322).

2.4.2 Technical Evaluation

Section 3.0 of the enclosure to Reference 1 itemizes the proposed changes to the PINGP licensing basis and the SSCs affected by these changes as a result of the implementation of the AST methodology at PINGP. The SSCs affected by the proposed AST implementation include the reactor containment vessel (i.e., the primary containment system), the SB (i.e., the secondary containment system), the SBVS, the ABSVS, the ABNVS, the CRSVS, the spent fuel pool enclosure, the spent fuel pool special ventilation system (SFPSVS), and the ESFs alternating current and direct current electrical power systems. Attachment 4 to Reference 1 includes the licensee's conformance matrices (tables) documenting the comparison of the guidance found in RG 1.183 to the licensee's LAR contents for the proposed AST methodology implementation at PINGP. As indicated in Table A of Attachment 4, the licensee noted one exception to the guidance found in Section 5.1.2 of RG 1.183. This exception indicates that accident mitigation credit was taken for two, nonsafety-related dampers which are part of the ABNVS. As noted above, the NRC staff's review of the proposed LAR focused on verifying that the aforementioned dampers will continue to satisfy the applicable regulatory requirements related to the structural integrity and seismic adequacy of SSCs which are credited to perform accident mitigating functions.

2.4.2.1 Auxiliary Building Ventilation Systems

Sections 10.3.2 and 10.3.4 of the PINGP USAR describe the pertinent characteristics of the ABNVS and the ABSVS, respectively. The ABNVS provides ventilation for the Auxiliary Building under normal operating conditions. The ABSVS provides a sub-atmospheric pressure environment for the Auxiliary Building Category I Ventilation Zone (i.e., the Auxiliary Building Special Ventilation Zone or ABSVZ) in order to support the removal of fission products resulting from a LOCA. The ABSVZ contains those areas of the Auxiliary Building which have the potential to collect a significant amount of primary coolant leakage as a consequence of a LOCA. The establishment of the ABSVZ is accomplished, in part, by isolating the ABNVS air make-up and exhaust paths utilizing the two aforementioned, nonsafety-related dampers which are part of the ABNVS. As indicated in Reference 1, the ABNVS as a whole is not credited for accident mitigation as part of the proposed AST methodology implementation at PINGP. In order to demonstrate that the dampers would be able to perform their required functions following the implementation of the proposed AST, the licensee provided a detailed listing of assurances regarding the dampers in Reference 1.

In order to assess the merits of the licensee's AST LAR with respect to the applicable regulatory requirements, the NRC staff issued an RAI regarding these assurances due to the lack of discussion regarding the structural integrity and seismic qualification of these nonsafety-related dampers. In response to the NRC staff's RAI (Reference 2), the licensee indicated that these dampers were seismically qualified by one of two methods. The first follows the methodology described in Section 12.2 of the PINGP USAR for Category I SSCs. This section of the USAR notes that Category I SSCs at PINGP are evaluated for structural integrity and seismic adequacy using loading combinations which include SSE loads. The evaluation results include ensuring that the stresses in these SSCs remain below design code allowable stresses which are also described in the PINGP USAR. The second method described in the licensee's RAI response involves the use of the Electric Power Research Institute (EPRI) Technical Report 1014608, "Seismic Evaluation Guidelines for [Heating, Ventilation, and Air Conditioning] HVAC Duct and Damper Systems," to assess the seismic adequacy of the dampers under review. This EPRI Technical Report is a revision to EPRI Technical Report 1007896.

2.4.2.2 NRC Staff Evaluation

The NRC staff considers the licensee's efforts to demonstrate the seismic ruggedness of the nonsafety-related dampers acceptable. This acceptance is based on the fact that either of the two methodologies described in the licensee's response provides reasonable assurance that the dampers will maintain an adequate level of structural integrity following a design basis earthquake (i.e., DBE or SSE). This assurance is supported by the fact that the first methodology described above ensures that those SSCs analyzed in accordance with the methods found in Section 12.2 of the PINGP USAR are subjected to loading combinations which include SSE seismic loads. These SSCs are then evaluated against design codes described in Chapter 12 of the PINGP USAR to ensure that the maximum allowable stress levels prescribed by these codes are not breached under design basis loading conditions.

The NRC staff also considers the second methodology, utilizing EPRI Technical Report 1014608, acceptable for demonstrating the ability of the aforementioned dampers to maintain

their structural integrity under design basis seismic loading at PINGP. The methodology described in the subject EPRI report relies on the evaluation of seismic failure mechanisms for duct and damper systems from seismic experience data and includes methods to screen and identify seismic vulnerabilities and weaknesses in HVAC SSCs. In addition, the EPRI methodology relies on in-plant walkdowns and reviews to determine the seismic adequacy of HVAC SSCs, selecting representative review samples, performing analytical reviews and resolving outliers. The NRC staff has previously found this methodology acceptable for use on a case-by-case basis for demonstrating the seismic adequacy of HVAC components credited for the implementation of AST methodologies. Notably, the AST license amendment issued for the Hatch Nuclear Plant, discussed in SE Section 2.4.1 above, made extensive use of EPRI Technical Report 1007896, of which EPRI Technical Report 1014608 is a comparable revision. While the NRC staff considers this methodology acceptable for the demonstration of the seismic adequacy of the two nonsafety- related dampers at PINGP, the staff's acceptance of utilizing the seismic experience-based methodology of the above referenced EPRI report is limited to its application for the proposed PINGP AST methodology implementation and not an endorsement for its use in evaluating other SSCs at PINGP, or elsewhere, where the NRC staff has not yet evaluated its use.

2.4.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed LAR associated with the implementation of the full scope AST methodology at PINGP on the aforementioned SSCs, including the ABNVS nonsafety-related dampers. On the basis of the staff's review as described above, which demonstrates the structural adequacy of the SSCs credited in the implementation of the AST methodology at PINGP, the NRC staff finds the proposed AST implementation acceptable. This acceptance is based on the demonstration that SSCs credited to support the AST implementation meet the aforementioned regulatory requirements and provides reasonable assurance that these SSCs will be able to perform their intended functions under their associated design-basis loading conditions.

2.5 Containment and Ventilation Systems

The proposed changes will revise the TS requirements for the ABSVS, the SFPSVS, Containment Penetrations, the Ventilation Filter Testing Program, the Containment Leakage Rate Testing Program, and the Control Room Habitability Program. The proposed changes reviewed in this section of the SE are as follows:

A. TS 3.3.7, "Spent Fuel Pool Special Ventilation System Actuation Instrumentation"

The licensee stated that this TS and associated Bases will be removed from TSs. It provides requirements for instrumentation associated with the SFPSVS. The licensee stated that this change is acceptable because the SFPSVS is no longer credited for filtration or isolation. Thus, the SFPSVS is not required in order to meet dose consequence limits. Therefore, the actuation instrumentation for the SFPSVS is not required to meet dose consequence limits.

B. TS 3.7.12, "Auxiliary Building Special Ventilation System"

The LAR proposes to revise TS SR 3.7.12.3. This surveillance requirement verifies that

each train of the ABSVS can produce a negative pressure within a specified duration after initiation. The current duration is six minutes. The LAR proposes to revise this duration to 20 minutes.

C. TS 3.7.13, "Spent Fuel Pool Special Ventilation System"

TS 3.7.13 provide requirements for the SFPSVS. This LAR removes this TS and associated TS Bases.

D. TS 3.9.4, "Containment Penetrations"

This TS and associated Bases are being replaced with a TS and Bases on Decay Time, which requires that recently irradiated fuel (fuel that has occupied part of the critical reactor core within the previous 50 hours) cannot be handled. This change is requested because the FHA analysis does not credit containment closure when handling fuel more than 50 hours after shutdown.

E. TS 5.5.9, "Ventilation Filter Testing Program"

This LAR proposes to revise TS 5.5.9 by deleting the Spent Fuel Pool Special and Inservice Purge Ventilation System (SFPSIPVS) from the Ventilation Filter Testing program. This is acceptable because the FHA analysis does not credit the SFPSIPVS for filtration or isolation of the respective area. Therefore, the SFPSIPVS is not required in order to meet dose consequence limits, so testing is no longer necessary for the SFPSIPVS filters.

E1. TS 5.5.9b and TS 5.5.9d

This LAR proposes to revise TS 5.5.9b and TS 5.5.9d to reflect that the SBVS is no longer applicable to charcoal adsorber testing. This is requested because the LOCA analysis only credits the SBVS's HEPA filter and takes no credit for the charcoal adsorber.

E2. TS 5.5.9c

This LAR proposes to revise TS 5.5.9c to delete the methyl iodide penetration values for the SBVS and SFPSIPVS and to revise the methyl iodide penetration value for the ABSVS. Deletion of the methyl iodide penetration values for the SFPIPVS is requested because the SFPSIPVS is not required in order to meet dose consequence limits, so testing of the charcoal adsorbers for the SFPSIPVS is no longer necessary. Deletion of the methyl iodide penetration values for the SBVS is acceptable because the SBVS charcoal filter is not credited in the dose consequence analysis, so testing of the charcoal adsorber is not required. The revision of the methyl iodide penetration value for ABSVS is required in order to achieve acceptable dose consequences in the control room for the LOCA analysis. The ABSVS charcoal filter efficiency is increased from 70 percent to 80 percent, which will increase the additional 10 percent iodine loading on the charcoal and proportionately increase the filter shine dose.

E3. TS 5.5.9e

This LAR proposes to revise TS 5.5.9e to delete the test face velocity values for the SBVS and SFPSIPVS. This amendment is requested because the SFPSIPVS is not required in order to meet dose consequence limits, so testing of the charcoal adsorbers for the SFPSIPVS is no longer necessary. The SBVS charcoal filter is not credited in the dose consequence analysis, so testing of the charcoal adsorber is not required.

F. TS 5.5.14, "Containment Leakage Rate Testing Program"

This LAR proposes to revise values for the primary containment leakage rate. To achieve acceptable dose consequences in the CR and TSC for the LOCA, the licensee determined it is necessary to reduce the assumed containment leakage rates. TS 5.5.14 provides the containment leakage-rate testing program. TS 5.5.14c provides acceptance criteria values for primary containment leakage.

G. Proposed TS 5.5.16, "Control Room Habitability Program"

This LAR proposes to revise the proposed TS 5.5.16, to delete references to TID dose consequence of 5 rem whole body. By letter dated June 24, 2009 (ADAMS AN ML091760988), NSPM proposed a revision to this TS. The revised TS included references to both AST and TID dose consequences for a fuel handling accident. This current proposed revision to delete reference to TID is acceptable because reference to the TID methodology is no longer applicable at PINGP.

2.5.1 Regulatory Evaluation

This section of the SE addresses the acceptability of the requested changes to the TSs. The regulatory requirements applicable to this evaluation are 10 CFR 50 Appendix A, General Design Criterion 19, "Control Room" and Part 50.67 (10 CFR 50.67) "Accident Source Term."

2.5.2 Technical Evaluation

A. TS 3.3.7, "Spent Fuel Pool Special Ventilation System Actuation Instrumentation"

This TS provides requirements for instrumentation associated with the SFPSVS. The SFPSVS ensures that radioactive material in the fuel pool enclosure atmosphere following a fuel handling accident are filtered and adsorbed prior to exhausting to the environment. This action serves to reduce the radioactive content in the fuel pool enclosure exhaust following a fuel handling accident so that offsite doses remain within the limits specified in 10 CFR Part 100. The LCO requirements ensure that instrumentation necessary to initiate the SFPSVS is OPERABLE.

As stated below in the evaluation of the licensee's request to remove TS 3.7.13 and the associated Bases from TSs, the NRC staff agrees with the licensee's contention that TS 3.7.13 no longer meets any of the four criteria listed in 10 CFR 50.36(c)(2)(ii) for inclusion in TSs. Therefore, the staff also agrees that the actuation instrumentation for

this system, which is not required in order to meet dose consequences limits, may also be removed from the TSs. The NRC staff finds this request acceptable.

B. TS 3.7.12, "Auxiliary Building Special Ventilation"

The LAR proposes to revise TS SR 3.7.12.3. This surveillance requirement verifies that each train of the ABSVS can produce a negative pressure within specified duration after initiation. The current duration is 6 minutes. The LAR proposes to revise this duration to 20 minutes.

The licensee's calculation No. GEN-PI-079, Revision 0, assumes the drawdown time for the ABSVZ. Most of the potential leakage through containment penetrations will be collected and processed in the Shield Building or the ABSVZ. The ABSVS serves the ABSVZ.

The requested change is consistent with the analysis provided in Calculation No. GEN-PI-079. The ABSVZ drawdown time used in the calculation input (calculation design input parameter no. 5.3.2.5) is 20 minutes. The NRC staff finds the requested change to TS SR 3.7.12.3 for producing a negative pressure in the ABSVZ within 20 minutes instead of 6 minutes to be acceptable because this is within the value used as an input to the calculation of record for post-LOCA exclusion area boundary, low population zone, and CR doses.

C. TS 3.7.13, "Spent Fuel Pool Special Ventilation System"

TS 3.7.13 provides requirements for the SFPSVS. This LAR removes TS 3.7.13 and associated TSs Bases.

The licensee's calculation No. GEN-PI-077, Revision 0, was performed to determine the EAB, LPZ and CR doses due to a FHA occurring with the reactor being shutdown for at least 50 hours. The calculation assumes post-FHA activity is released to the atmosphere through the common area of the auxiliary building. This calculation does not take credit for the SFPSVS.

The NRC staff's assessment found the requested change to remove TS 3.7.13 and the associated Bases acceptable. The LAR shows that the current TS 3.7.13 no longer meets any of the four criteria listed in 10 CFR 50.36(c)(2)(ii) for inclusion in Technical Specifications.

D. TS 3.9.4, "Containment Penetrations"

This TS and associated Bases are being replaced with a TS and Bases on Decay Time, which requires that recently irradiated fuel (fuel that has occupied part of the critical reactor core within the previous 50 hours) cannot be handled. This change is requested because the FHA analysis does not credit containment closure when handling fuel more than 50 hours after shutdown.

The NRC previously issued Amendment No. 166 (Facility Operating License No. DPR-42) and Amendment No. 156 (Facility Operating License No. DPR-60) (ADAMS

AN ML042430504). These amendments revised TS 3.9.4 requiring containment penetrations to be in their required position (TS LCO 3.9.4) when moving fuel assemblies in containment that have undergone radioactive decay for less than 50 hours.
The NRC staff's assessment finds that the new TS 3.9.4, "Decay Time," provides a level of safety comparable to the TS it is replacing, TS 3.9.4, "Containment Penetrations." The old TS permitted movement of fuel in containment that had decayed less than 50 hours as long as containment penetrations were in their required positions. The new TS prohibits the movement of any fuel in containment that has undergone radioactive decay less than 50 hours. The NRC staff finds the proposed change acceptable.

E. TS 5.5.9, "Ventilation Filter Testing Program"

This LAR proposes to revise TS 5.5.9 by deleting the Spent Fuel Pool Special and Inservice Purge Ventilation System (SFPSIPVS) from the Ventilation Filter Testing program. This is acceptable because the FHA analysis does not credit the SFPSIPVS for filtration or isolation of the respective area. Therefore, the SFPSIPVS is not required in order to meet dose consequence limits, so testing is no longer necessary for the SFPSIPVS filters.

The NRC staff's review found the dose assessment for the radiological dose at the EAB, LPZ, and CR no longer credits the SFPSIPVS. Consequently, the staff concluded that the LAR to delete the SFPSIPVS from TS 5.5.9 is acceptable.

E1. TS 5.5.9b and TS 5.5.9d

This LAR proposes to revise TS 5.5.9b and TS 5.5.9d to reflect that the Shield Building Ventilation System (SBVS) is no longer applicable to charcoal adsorber testing. The basis for this request is the new LOCA analysis that only credits the SBVS HEPA filter and takes no credit for the charcoal adsorber.

The NRC staff requested additional information regarding periodic verification of inleakage into the SB annulus. The licensee's calculation No. GEN-PI-079, "Post-LOCA EAB, LPZ, and CR Doses – AST", assumes a specific Shield Building leakage rate as shown in Table 2 of the calculation. This leakage rate helps determine the recirculation and holdup time of any source term leaking into the SB annulus.

The licensee responded stating that when activated, the SBVS exhausts directly to the shield building exhaust stack to draw a negative pressure in the shield building annulus area outside of the reactor containment vessel. After a pre-set negative pressure is established, a recirculation damper opens to allow the system to recirculate a portion of the total airflow while exhausting to maintain a negative pressure. During a phone conference the licensee clarified that the discharge and the recirculation dampers are two-position dampers and do not modulate in response to any control signal. There is only one fan per train, providing both recirculation and exhaust functions.

The licensee indicated that they do not directly measure the SB inleakage or the SBVS exhaust flow. They use an indirect method to verify that they do not exceed the SBVS exhaust value used in the accident analysis. In 1983, the licensee requested and

received an amendment to delete the shield building leak-testing requirement. They stated that an annulus drawdown test would demonstrate acceptable performance.

Under steady state conditions, the mass flow rate of air entering the annulus equals the mass flow rate of air exhausted. Under the conditions considered, the density of the air entering is not significantly different from that of the exhausted air. Therefore, the volumetric flow of air entering is approximately equal to the volumetric flow exhausted.

For fluid flow, friction loss is proportional to the square of the velocity of the fluid. Low air inleakage into the annulus results in low exhaust airflow. Low exhaust flow results in a low discharge duct velocity. As inleakage increases, exhaust flow increases, increasing the pressure drop in the exhaust duct. In leaking air will also be subject to friction. Assuming the inleakage flow-area and length are constant, an increase in leakage will result in an increase in pressure drop along the length of the flow area.

The SBVS fan provides a pressure increase from its inlet to the outlet. The fan pressure differential is independent from the surrounding environment. The fan operation is not affected by the inlet pressure at ambient or the discharge pressure at ambient. The fan inlet and discharge pressures relative to ambient pressure vary as the inlet and discharge "conduit" pressure drops vary. (Conduit is used here as the collection of all items along the flow path.)

Under steady state conditions, the pressure drop from outside the shield building, through any leaks, through the inlet ductwork to the fan inlet is the same as the pressure drop from the fan discharge through the exhaust duct to the discharge outdoors. Inlet ductwork and discharge ductwork remain constant. The variables are the pressure drop through the filter assembly, the fan performance, and the pressure drop through shield building inleakage paths. Routine surveillance testing verify filter pressure drop and fan performance.

The differential pressure between the outdoors and the shield building annulus helps determine the inleakage. Increasing differential pressure increases the inleakage. At a steady state condition, the pressure drop through the shield building leakage paths, through the inlet duct, to the inlet of the fan is equal to the pressure drop from the fan discharge, through the exhaust duct, to atmosphere.

The licensee developed an algorithm to determine the inleakage based on measured annulus pressure. This algorithm was verified using startup data. The licensee uses this algorithm along with surveillance testing annulus-atmosphere pressure to verify the shield building inleakage (and SBVS exhaust). The licensee applies a safety factor of "2" to the calculated value to assure the leakage/exhaust rate is within the limits assumed in the control room and offsite dose analyses.

The NRC staff's assessments find the dose assessment for radiological dose at the EAB, for the LPZ, and the CR no longer credit the SBVS charcoal adsorber. The surveillance testing currently used to verify the shield building total leakage is sufficient (based on the above discussion) to assure the leakage value used in the dose analyses is not exceeded. The analysis of record for post-LOCA exclusion area boundary, low population zone, and CR doses credits the SBVS HEPA filter for aerosol removal only.

No credit is taken for elemental iodine or organic iodine removal by the SBVS charcoal adsorber. There is no longer any need perform a TS performance test of the SBVS charcoal adsorber. Therefore, the LAR to delete the SBVS charcoal adsorber from TS 5.5.9 is acceptable.

E2. TS 5.5.9c

This LAR proposes to revise TS 5.5.9c to delete the methyl iodide penetration values for the SBVS and SFPSIPVS and to revise the methyl iodide penetration value for the ABSVS. Deletion of the methyl iodide penetration values for the SFPIPVS is requested because the SFPSIPVS is not required in order to meet dose consequence limits, so testing of the charcoal adsorbers for the SFPSIPVS is no longer necessary. Deletion of the methyl iodide penetration values for the SBVS is acceptable because the SBVS charcoal filter is not credited in the dose consequence analysis, so testing of the charcoal adsorber is not required. The revision of the methyl iodide penetration value for ABSVS is required in order to achieve acceptable dose consequences in the control room for the LOCA analysis. The ABSVS charcoal filter efficiency is increased from 70 percent to 80 percent, which will increase the additional 10 percent iodine loading on the charcoal and proportionately increase the filter shine dose.

The NRC staff's assessment found the request to delete the methyl iodide penetration values for the SBVS and the SFPSIPVS to be acceptable. The accident analyses for radiological dose at the EAB, for the LPZ, and for the CR no longer credits the charcoal adsorbers for these systems.

The laboratory tests of charcoal adsorbers are performed in accordance with ASTM D3803-1989 at a temperature of 30 °C and 95 percent relative humidity. In RG 1.52 Revision 3, "Design, Inspection, and Testing, Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants," it provides a formula for determining allowable methyl iodide penetration. The penetration is equal to 100 percent minus the organic iodide efficiency credited in the accident analysis, the total divided by a safety factor. The minimum acceptable safety factor is two.

TS 5.5.9c is being changed to read in part "...methyl iodide penetration less than: 1) 10 % penetration for ABSVS ..." The licensee's calculation No. GEN-PI-079 assumes ABSVS charcoal adsorber efficiencies of 80 percent for elemental iodine and 80 percent for organic iodine. A safety factor of two yields an allowable penetration of 10 percent.

The NRC staff's assessment finds the proposed change to the allowable methyl iodide penetration for the ABSVS charcoal adsorber from 15 percent to 10 percent meets the acceptance criteria for the assumed adsorber efficiency. The proposed change is acceptable.

E3. TS 5.5.9e

This LAR proposes to revise TS 5.5.9e to delete the test face velocity values for the SBVS and SFPSIPVS. This amendment is requested because the SFPSIPVS is not required in order to meet dose consequence limits, so testing of the charcoal adsorbers

for the SFPSIPVS is no longer necessary. The SBVS charcoal adsorber is not credited in the dose consequence analysis, so testing of the charcoal adsorber is not required. The NRC staff's assessment found that the laboratory testing criteria for both the SFPSIPVS and the SBVS charcoal adsorbers is being removed from TS 5.5.9e. With the removal of the laboratory test criteria, there is no longer a need to maintain the minimum filter test velocities for the SFPSIPVS and SBVS charcoal adsorbers. Therefore, the proposed change to delete the minimum filter test velocities for the SFPSIPVS and SBVS charcoal adsorbers is acceptable.

F. TS 5.5.14, "Containment Leakage Rate Testing Program"

This LAR proposes to revise values for the primary containment leakage rate. To achieve acceptable dose consequences in the control room and technical support center for the LOCA, the licensee determined it is necessary to reduce the assumed containment leakage rates. TS 5.5.14 provides the containment leakage-rate testing program. TS 5.5.14c provides acceptable values for primary containment leakage.

To be consistent with the AST dose consequence analysis, the licensee proposed (1) a maximum allowable primary containment leak rate of 0.15 percent of containment weight per day, (2) for pipes connected to the ABSVZ a leak rate of 0.06 percent of containment weight per day, and (3) for pipes connected to systems that are exterior to both the shield building and the ABSVZ, a leak rate of 0.006 of containment weight per day.

The maximum allowable primary containment leakage rate, L_a, at pressure P_a (Pa is the peak calculated containment internal pressure related to the design basis loss-of-coolant accident as specified in the TSs) will be reduced from 0.25 percent to 0.15 percent of the primary containment air weight per day.

The NRC staff's assessment finds that the new 0.15 percent primary containment air weight per day leakage rate is consistent with the value assumed in the calculation of record, Calculation No. GEN-PI-079, "Post-LOCA EAB, LPZ, and CR Doses - AST." Therefore, the proposed change is acceptable.

G. Proposed TS 5.5.16, "Control Room Envelope Habitability Program"

This LAR proposes to revise TS 5.5.16, to delete references to TID dose consequence of 5 rem whole body. TS 5.5.16 was proposed by letter dated June 24, 2009 (ADAMS AN ML091760988). This TS included references to both AST and TID dose consequence because at the time, PINGP was both a TID and AST plant for FHA. On May 20, 2010, Amendment No. 195 to the PINGP Unit 1 operating license and Amendment No. 184 to the PINGP Unit 2 operating license were issued (ADAMS AN ML100890640) adding TS 5.5.16. The deletion of the reference to TID dose consequence is acceptable because the reference to the TID methodology will no longer be applicable to PINGP.

2.5.3 Conclusion

The NRC staff has reviewed the licensee's proposed changes to the containment and ventilations systems associated with implementing an alternative source term at PINGP. Based

on its review and the considerations discussed above, the NRC staff has concluded that the proposed revision to the PINGP licensing basis with respect to the containment and ventilation systems is acceptable.

2.6 Reactor Systems

This section of the SE documents the NRC staff's evaluation of the margin-to-overfill (MTO) analysis and mass release analysis of a steam generator tube rupture (SGTR) event. The MTO analysis is used to validate the assumption that the SG will not overfill and only steam releases will occur during an SGTR event. The mass releases analysis is used to calculate the steam releases through the SGs. The results of both analyses will be used as input in the dose analysis to show that the applicable dose limits are met.

2.6.1 Regulatory Evaluation

The following regulatory requirements and guidance documents are applicable:

10 CFR 50.34, "Contents of applications; technical information," requires that safety analysis reports be submitted that analyze the design and performance of 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 re-evaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In 10 CFR 50.67 it stipulates that the NRC may only issue an AST amendment if an applicant's analysis demonstrates with reasonable assurance that the dose limits specified in 10 CFR 50.67(b)(2) would be met.

Regulatory Position 1.3.2 specified in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states that an analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

Section 5.1.3 in RG 1.183 states that the numeric values that are chosen as inputs to the analyses should be selected with the objective of determining a conservative postulated dose.

The NRC staff's review covers:

(1) Postulated initial core and plant conditions,
(2) Method of thermal and hydraulic analysis,
(3) The sequence of events (assuming offsite power either available or unavailable),

(4) Assumed reactions of reactor system components,
(5) Functional and operational characteristics of the RPS,
(6) Operator actions consistent with the plant's emergency operating procedures (EOPs), and
(7) The results of the accident analysis.

2.6.2 Technical Evaluation

An SGTR event will transfer radioactive reactor coolant to the shell side of the SG as a result of the ruptured SG tube, and ultimately to the atmosphere. Therefore, the SGTR analyses for the proposed AST application were performed to show that the resulting doses will stay within allowable guidelines. The NRC staff's review of the SGTR in this section of the SE focused on the thermal-hydraulic analyses for the SGTR in order to: (1) confirm that the faulted SG does not experience an overfill, since preventing SG overfill is necessary in order to justify the assumption in the SGTR analysis that only steam releases will occur, and (2) verify that the maximum mass releases are calculated for use as input in the radiological dose calculations.

2.6.2.1 Steam Generator MTO Analysis for an SGTR Event

Section 3.7 of Reference 1 discussed the radiological dose analysis for an SGTR event. The analysis included a key assumption that the SG overfill will not occur and the ruptured SG can be isolated within 30 minutes of an SGTR initiation, and thus, only considered steam releases carrying radioactive material for the radiological dose analysis. Water releases have a significantly greater concentration of radioactive material when compared with that of steam releases and will result in worse radiological releases. If the SG overfill occurs, the secondary side of the SG will be filled and water may enter the steam lines, resulting in an unanalyzed condition. As a result, the radiological dose analysis for the SGTR event may become invalid and unacceptable for the AST application.

The MTO analysis is a method that can be used to validate the assumption that the SG will not overfill and only steam releases will occur during an SGTR event. Without an acceptable MTO analysis, the NRC staff cannot determine the validity of the key assumption for mass release, and thus, can accept neither the calculated steam mass releases used as an input to the radiological dose analysis, nor the associated SGTR radiological release analysis for supporting the AST application. Therefore, the NRC staff determines that the conclusions drawn from the MTO analysis are essential considerations in the NRC staff's determination of the acceptability of the SGTR dose releases analysis, which is used to support the AST application.

In response to an NRC staff's RAI, the licensee provided a discussion of the MTO analysis (Reference 3). The licensee indicated that it used the plant-specific simulator to perform the SG MTO analysis during an SGTR event for the AST application. Specifically, the simulator was used in combination with EOP E-0, "Reactor Trip or Safety Injection," and E-3, "Steam Generator Tube Rupture," to determine the response time for the break flow termination. Based on the response times (including the operator action times and the RCS thermal hydraulic response times), the flow rates for the break flow into the ruptured SG, and the volume available in the ruptured SG, the SG MTO was determined.

Based on its review of the MTO analysis, the NRC staff found that the assumptions used in the analysis for various initial plant conditions (such as the pressure difference between the RCS

primary side and secondary side in the ruptured SG) were conservative with respect to calculating the break flow rates into the ruptured SG. The NRC staff found that the operator action times and the RCS thermal-hydraulic response times would significantly affect the overall conservatism of the SGTR MTO analysis, as they were used as part of the input to calculate the total amount of the break flow into the ruptured SG, which would determine if the ruptured SG would be over-filled with water. Therefore, the NRC staff determined that the use of a method acceptable to the NRC for determining the operator action and RCS thermal hydraulic response times is equally as important as the assumptions of the initial conditions that maximize break flow rates into the ruptured SG in assuring the adequacy of the SGTR MTO analysis.

Based on its review, the NRC staff found that the computer code and RCS thermal-hydraulic models used in the simulator at PINGP had not been previously submitted to the NRC for review and approval for use in the transient and accident analyses in support of licensing applications. The NRC staff compared the results from the licensee's simulator exercises to the results from the NRC-approved MTO analyses for other comparable Westinghouse 2-loop plants (PINGP is a Westinghouse 2-loop plant). The comparison revealed that the total response time to terminate the break flow from the RCS primary to secondary side using the simulator at PINGP was significantly shorter (about 30 minutes vs. 50 minutes) than the response times of the other plants. The response time was used to determine the total amount of the break flow to the SG secondary side, which determined the SG MTO. The shorter response time calculated by the licensee's simulator exercises for PINGP is, therefore, not conservative with respect to the NRC-approved SG MTO analyses.

In a follow-up RAI, the NRC staff requested the licensee to provide information related to the computer code and thermal-hydraulic models used in the simulator for the NRC staff review and approval. Alternatively, the NRC staff indicated that the licensee could perform a MTO reanalysis for the SGTR event by using the previous NRC-approved computer codes and methods, and submit the results of the reanalysis to the NRC for review and approval. The NRC staff informed the licensee that an example of the NRC-approved SGTR MTO method for Westinghouse plants could be found in WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill."

2.6.2.1.1 Methodology and Computer Code Used in the MTO Analysis

In its RAI response, the licensee provided an SG MTO reanalysis in Section A of Enclosure 2 in Reference 4. The analysis was performed using the NRC-approved methodology documented in WCAP-10698-P-A (Reference 5). The licensee did not use the simulator for the MTO analysis, with exception of substantiating operator action times credited in the analysis. The analysis was based on a double-ended rupture of a single SG tube, which would result in the greatest credible break flow from the RCS primary-to-secondary system and thus, a minimum MTO. The analysis did not consider the effect of a single failure concurrent with the SGTR event. The NRC staff found that this approach was acceptable since the assumption of not considering the worst single failure was consistent with the current license basis.

The MTO analysis was performed using the LOFTTR2 code, which explicitly modeled operator actions consistent with the results of simulator exercises and emergency operating procedure EOP-3. The use of LOFTTR2 was a deviation from the WCAP 10698-P-A methodology, which used the LOFTTR1 code. The NRC staff found that the LOFTTR2 code used in an SGTR MTO

analysis was previously approved by the NRC for a Westinghouse plant (Reference 6). Therefore, the NRC staff concluded that the use of the LOFTTR2 code was acceptable.

2.6.2.1.2 Initial Conditions

The MTO analyses were performed based on the configuration of replacement SGs (Framatome ANP 56/19). Currently, Unit 1 is installed with Framatome ANP 56/19 SGs and Unit 2 is installed with Westinghouse Model 51 SGs, which will be replaced by Framatome ANP 56/19 SGs, currently scheduled for the fall of 2013. Therefore, to reflect the SG configuration assumed in the analysis, the licensee proposed to add, in Appendix B in Enclosure 1 of Reference 4, a license condition that requires the licensee to provide the NRC written notification when Unit 2 replacement SG (RSG) installation is complete and AST license amendment implementation has commenced. The written notification should be submitted to the NRC within 30 days after completion of the outage in which the Unit 2 RSGs are installed. The NRC staff determined that the added license condition was acceptable, since the license condition would preclude the implementation of the AST amendments until after the installation of the RSGs that were considered in the MTO analysis.

The analysis used conservative values of the plant initial conditions, which resulted in a minimum MTO. The values used in the analysis were listed in Tables 4 and 5 of Enclosure 2 in Reference 4. The NRC staff found that the values used were consistent with the methodology in the NRC-approved report, WCAP-10698-P-A, except for the following plant parameters: (1) the analyses were based on the power level of 1683 megawatt thermal (MWt), which is the current licensed reactor core power level of 1677 MWt, plus calorimetric uncertainties, (2) a maximum (a nominal value plus uncertainties) auxiliary feedwater (AFW) flow rate, and minimum (a nominal value minus uncertainties) AFW temperature and decay heat (ANS 1979 minus 2 standard deviations) were modeled in the analysis, and (3) the initiation of the charging flow was based on the operator actions well after reactor trip. Since (1) the power level was consistent with the current licensed power level with inclusion of uncertainties, and (2) the values for plant parameters were determined by the licensee's sensitivity study for the limiting case that resulted in a minimum MTO, the NRC staff determined that the values for the key plant parameters were conservative and acceptable for the PINGP MTO analysis.

One of the key plant parameters that would affect the results of the SG MTO analysis is the initial SG water level. In the analysis, the licensee used a 73 percent narrow range span (NRS) water level for each SG, because it represented the water level of 44 percent NRS corresponding to the nominal setpoint for SG level at the 100 percent power level with inclusion of the 10 percent increase in initial SG mass for uncertainties plus the mass added due to the turbine runback.

The SG initial water levels are controlled by the plant SG level control program as a function of the initial power level. During normal operation, the SG initial levels are controlled at a higher value when the power level is low and a lower value when the power level is high. In Figures 9 and 10 of reference 3, the licensee showed that the 100 percent power level would capture 95 percent of the operating time during a fuel cycle for Units 1 and 2. Based on the plant operating data, the NRC staff determined that the SG water level based on a 100 percent power level represented a significant portion of the operating time during fuel cycle, and therefore, the use of its corresponding initial SG water level in the analysis was acceptable. The NRC staff also found that the value of the initial SG water level used in the analysis had included

uncertainties and the effect of turbine runback, and therefore, determined that it was conservative, resulting in a minimum MTO, and was acceptable.

2.6.2.1.3 Operator Actions and Action Times

The operator actions and the associated action times modeled in the analysis were shown in Table 5 of Reference 3. Specifically, operator action to initiate RCS cooldown using the power-operated relief valve on the intact SG was credited within 19 minutes following the reactor trip. Additionally, LOFTTR2 was used to model operator actions to initiate RCS depressurization in four minutes following completion of the RCS initial cooldown, secure safety injection (SI) pumps in two minutes following the depressurization, and balance letdown and charging flow in 15 minutes following securing the SI pumps.

The licensee confirmed (RAI 2 response, Reference 7) that it will document the above discussed operator action times in its applicable plant procedures and validate the action times. The validation procedure requires all individuals and crew to meet the operator action times during the validation process. Whenever a required operator time is not met during a training scenario, the licensee will place the discrepancy into its plant-specific corrective action program. The cause of discrepancy will be determined and corrected.

Operator action delay times credited in this analysis are evaluated in Section 2.8 of this SE.

2.6.2.1.4 Nonsafety-Related Equipment for Consequence Mitigation

In Tables 1 and 2 of Reference 7, the licensee identified all nonsafety-related systems, components, or instruments (SCIs) used by the operator to complete the manual actions credited in the SGTR analysis. Table 1 addresses systems and components and Table 2 addresses instrumentation. Information provided in Tables 1 and 2 included (1) a discussion of function of each of the identified SCIs, (2) a listing of backup SCIs that are available to meet the functions of the SCIs, (3) description of testing and maintenance performed to ensure the reliability of the SCIs, and (4) additional details regarding power supply and other pertinent consideration. For each of the identified nonsafety-related SCIs, the licensee provided justification in Tables 1 and 2 of Reference 7 for its use in the SGTR analysis. The justification provided reasonable assurance that the nonsafety-related SCIs would be operable on demand, since it showed that each SCI has more than one of the following design and operating features: (1) it has redundant SCIs; (2) it has backup SCIs that can be used for the same functions; (3) its operability and testing requirements are specified in pertinent TSs, or the maintenance rule program; (4) its power supply is safety-related and will be available during a loss-of-offsite power event; and (5) its reliability is demonstrated through normal operation.

Also, the MTO analysis was only used as supplemental information to the analysis in the USAR in support of the assumption of steam-only atmospheric releases during an SGTR event, and it did not change the current design basis events. In addition, the use of limited nonsafety-related equipment in the MTO analysis was consistent with Chapter 14.5.4.5, "Recovery Procedure," of the current USAR. Therefore, the NRC staff determined that the use of the nonsafety-related equipment was acceptable for operator actions credited in the MTO analysis.

The instrumentation and controls credited in the SGTR analysis are evaluated in Section 2.7 of this SE.

2.6.2.1.5 Pressurizer Thermal Shock Evaluation for Cooldown during the SGTR Event

Observing the time-temperature plots (Figure 5 and 6) provided in the licensee's response to the RAI dated June 22, 2011 (Reference 4), the transient associated with the SGTR MTO analysis shows what could be a pressurized thermal shock (PTS) event with respect to the reactor vessel. The NRC staff requested the licensee to justify that the condition (RT_{PTS}) of the limiting materials in the PINGP, Units 1 and 2, RPVs is in compliance with the NRC's PTS rule, 10 CFR 50.61, and the RPVs are adequately protected from failure due to the transients shown in the RAI response for the operating life time of the facility.

In the RAI response dated August 9, 2011 (Reference 7), the licensee provided two critical pieces of information. First, the results of the PINGP, Units 1 and 2 SGTR MTO analysis were bounded by the transients considered in NRC's SECY-82-465, "Staff Report on Pressurized Thermal Shock," dated November 23, 1982. SECY-82-465 provides the technical basis for the NRC's Pressurized Thermal Shock Rule, 10 CFR 50.61. As the transients considered in the 10 CFR 50.61 technical basis represented more severe cooldown events than those projected to occur during the PINGP Units 1 and 2 SGTR event, the application of the requirements of 10 CFR 50.61 to the PINGP, Units 1 and 2 will ensure adequate protection of the units' RPVs from failure due to PTS. Second, the licensee demonstrated that the limiting (i.e., most highly embrittled) materials of the PINGP, Units 1 and 2 RPVs meet the screening limits established in 10 CFR 50.61 throughout the period of the units' operating licenses. Therefore, based on its review of the licensee's response, the NRC staff concludes that the licensee has provided information which demonstrates that the PINGP, Units 1 and 2, RPVs will be adequately protected from the potential for brittle failure during a PTS event for the licensed operating life of each unit.

2.6.2.1.6 Results and Conclusion

The licensee has performed sensitivity analyses, considering various initial plant conditions, to determine the limiting MTO case at 1683 MWt, which is the current licensed reactor core power level of 1677 MWt with calorimetric uncertainties. The analyses used the methodology documented in NRC-approved report, WCAP-10698-P-A. The results showed a MTO of 186 ft^3 in the ruptured SG for the limiting case.

Based on its review discussed above, the NRC staff found that (1) the licensee's MTO analysis had adequately accounted for operation of the plant at the current licensed thermal power (CLTP) conditions, (2) the analysis was performed with appropriately conservative methods and an NRC-approved computer code, (3) the assumptions used in the analysis were conservative, resulting in a minimum MTO, and (4) the results showed that the SGTR event would likely not result in an overfill of the rupture SG. Therefore, the NRC staff concluded that the MTO analysis was acceptable to show a MTO during an SGTR event for the AST application at CLTP conditions.

2.6.2.2 SG Mass Release Analyses for a SGTR Event

Information on page 116 of Enclosure of Reference 1 indicated that the results of a recent Westinghouse SGTR analysis were used to determine: (1) the primary coolant releases to the ruptured SG, (2) the steam mass releases from ruptured SG to the environment, and (3) the steam mass releases from the intact SG to the environment. During the course of the review, the NRC staff requested the licensee to provide a discussion of the Westinghouse SGTR analysis and confirm that the methods used in the analysis were previously approved by the NRC. The NRC staff's evaluation of the licensee's response is discussed below.

2.6.2.2.1 Licensing Basis SG Mass Release Analysis

In its response to the RAI, the licensee explained in Section B of Enclosure 2 in Reference 4 that the Westinghouse SGTR analysis for PINGP Units 1 and 2 was consistent with that described in the original final safety analysis report (FSAR): it did not consider a single failure, and did not use computer codes or model specific operator actions. The event analyzed was based on the double-ended rupture of a single SG tube, which was previously identified as the limiting break, resulting in maximum mass releases. The primary-to-secondary flow rate was calculated using the orifice equation, neglecting the frictional losses in the SG tube to maximize the break flow rate. The analysis assumed that a reactor trip and an SI actuation occurred simultaneously when the pressurizer pressure decreased to the SI actuation point, and loss-of-offsite-power (LOOP) occurred at reactor trip. An assumption of a LOOP resulted in the release of steam to the atmosphere via the SG safety valves. It also assumed that, immediately following reactor trip, the RCS pressure stabilized at the equilibrium point where the incoming SI flowrate equaled the outgoing break flow rate. The break flow termination was assumed at 30 minutes after the initiation of the event.

The SGTR analysis included eight cases: four cases for Unit 1 SGs and other four cases for Unit 2 SGs at the varying combination of 0 percent to 10 percent SG tube plugging levels, and high and low values for RCS average temperature. These eight cases were performed individually to determine the primary-to-secondary break flow and steam releases to the atmosphere for the dose analysis between 0 and 30 minutes. The limiting break flow rates from all of the different calculations along with the limiting steam released to the atmosphere were used in the dose calculation. A single calculation was performed to calculate the long term steam releases from the intact SG for the time intervals 0 to 2 hours, 2 to 8 hours and 8 to 14 hours. The analysis showed that the Unit 2 SG configuration provided the bounding break flow and steam releases.

The analysis included a break flow flashing fraction that represented a portion of the break flow that would flash directly to steam entering the secondary side of the ruptured SG. The licensee identified conditions that would maximize mass releases in consideration of pressure differences between the RCS and the main steam system, while considering that lower secondary pressures and temperatures would reduce the SG enthalpy, and thus, have a deleterious - somewhat non-conservative - effect on the flashing prediction. The licensee determined that the highest possible pre-trip flashing fraction, based on the range of operating conditions covered by this analysis, was for a case with a hot-leg temperature of 606.8°F, RCS pressure of the SI setpoint of 1845 pounds per square inch absolute (psia) and an initial secondary pressure of 740 psia. All cases considered the same post-trip RCS pressure of 2062 psia and the post-trip SG pressure of 1016 psia. The licensing basis analytical results, including

the integrated tube rupture break flow, flashed break flow, and integrated atmospheric steam releases, were provided in Table 8 of Enclosure 2 in Reference 4.

The licensing basis analysis did not explicitly model operator actions, although it was assumed that break flow into the ruptured SG secondary side was terminated within 30 minutes. To confirm that the licensing basis analysis was acceptably conservative, the licensee performed two analyses, MTO and supplemental mass release analyses, using more sophisticated methods. These analyses demonstrated that: (1) the assumption of a steam-only atmospheric release was appropriate, and (2) MTO the licensing basis flashed break flow and atmospheric steam releases were conservative.

The NRC staff accepted the licensee's licensing basis analysis and results because they were based on conservative assumptions, and because the MTO reanalysis and supplemental mass release analysis, performed using more sophisticated methods, demonstrated that the licensing basis calculation was conservative. The evaluation of the MTO analysis is discussed in Section 2.6.2.1 of this safety evaluation. The evaluation of the supplemental mass release analysis is discussed in Section 2.6.2.2.2.

2.6.2.2.2 Supplemental Mass Release Analysis

The licensing basis analysis discussed in above Section 2.6.2.2.1 assumed that the break flow into the ruptured SG was terminated within 30 minutes. The licensee noted that the present licensing basis does not necessarily require such action to be taken, and the reactor coolant may continue to blow down into the ruptured SG beyond 30 minutes. Therefore, the licensee performed a supplemental mass release analysis to demonstrate that, even assuming the break flow would not be terminated within 30 minutes, the licensing basis analysis retained sufficient conservatism and was thus valid.

This analysis was performed using the same acceptable LOFTTR2 code discussed in above Section 2.6.2.1.1. The LOFTTR2 analysis explicitly modeled operator actions leading to break flow termination based on the PINGP EOPs and simulator exercises specific to PINGP Units 1 and 2. Specifically, operator action to isolate the ruptured SG was credited immediately following auxiliary feedwater isolation, and operator action to initiate RCS cooldown using the SG power-operated relief valve on the intact SG was credited within 19 minutes. Additionally, LOFTTR2 was used to model an operator action to initiate RCS depressurization seven minutes following completion of the RCS cooldown, secure emergency core cooling system two minutes following the depressurization, and balance letdown and charging flow in 15 minutes following SI termination. The operator actions and associated times were consistent with that used in the MTO analysis, except for seven minutes for RCS depressurization initiation time. As discussed above in Section 2.6.2.1.3 of the SE, the RCS depressurization initiation time of four minutes will be validated in the licensee's training program. Therefore, the NRC staff determined that the use of seven minutes was conservative, resulting in a greater mass release, and is thus, acceptable. In addition, the supplemental analysis modeled the allowable reactor vessel temperature, SG tube plugging ranges, and the power level consistent with the hand calculation input discussed above in Section 2.6.2.2.1 to dose analysis.

The analysis credited a reactor trip on over-temperature delta-temperature and an SI actuation on low pressurizer pressure. It did not include single failure consideration, which was consistent with the licensing basis analysis and, thus, was acceptable. The licensee presented the results

of the analysis in Table 8 of Enclosure 2 to Reference 4 for the mass release input to the dose analysis. Compared to the licensing basis analysis, the supplemental analysis calculated a 18 percent higher total tube rupture break flow (164,800 pounds-mass (lbm) vs. 140,000 lbm), but a 77 percent reduction in flashed break flow (4,000 lbm vs. 17,680 lbm). The atmospheric release from the ruptured SG was also 52 percent less (38,700 lbm vs. 80,500 lbm). Flashed break flow was assumed to directly release from the RCS to the environment with no mitigation in the secondary side of the ruptured SG, and thus, would be expected to have a significant impact on the results of the SGTR dose analysis. In response to the NRC staff's request, the licensee estimated the net effect of the flashed break flow and total steam releases on the dose consequences. The results of the licensee's estimation provided in the RAI 6 response of Reference 7 showed that the supplemental mass release analysis would predict a dose consequence about a factor of 2 lower than the current licensing basis analysis. The licensee assumed nominal initial conditions for various plant parameters, including initial power level of 1811 MWt (a power level for the future extended power uprate application) and secondary mass without consideration of uncertainty. Although these assumptions of the plant initial conditions were evaluated in more detail in Section 2.6.2.1.1 for the MTO analysis, in the case of the supplemental mass release analysis, the results of this analysis were acceptable to the NRC staff because there remained such a large difference between the predicted flashing fractions and atmospheric steam releases. Therefore, the licensing basis results were conservative.

The key reason for the large differences in results between the licensing basis analysis and the supplemental mass release analysis was that the supplemental release analysis used a flashing fraction that was based on the thermal-hydraulic conditions calculated by the computer code, and they were time-dependent. The licensing basis analysis conservatively assumed one flashing fraction for the pre-trip condition, and another for the post-trip condition. The licensing basis analysis was performed to maximize the flashing fraction. Based on the considerations discussed above, the NRC staff found that it remained acceptable because it provided a credible demonstration that the licensing basis analysis was conservative.

2.6.3 Conclusion

Based on its review, the NRC staff found that: (1) the MTO analysis was performed with appropriately conservative methods and acceptable computer code, (2) the assumptions used in the analysis were conservative, resulting in a minimum MTO, and (3) the results showed that the SGTR event would not result in an overfill of the ruptured SG. Therefore, the NRC staff concluded that the MTO analysis was acceptable to show no-overfill to occur during an SGTR event, and the assumption of the steam-only releases in the dose analysis remained valid. Also, the NRC staff determined that the current license basis mass release analysis was acceptable because it was based on conservative assumptions, and because the acceptable MTO reanalysis and supplemental mass release analysis, performed using more sophisticated methods, demonstrated that the licensing basis calculation was conservative.

2.7 Instrumentation and Controls

By letter dated November 22, 2011 (ADAMS AN ML113250575), the NRC staff requested additional information on the adequacy of the Principal and Backup instrumentation used for STGR mitigation. By letter dated December 8, 2011 (ADAMS AN ML113430091), the licensee provided the necessary information concerning the adequacy of the instrumentation used for

STGR mitigation. This section of the SE addresses the adequacy of this instrumentation for SGTR mitigation.

2.7.1 Regulatory Evaluation

The NRC staff reviewed the proposed changes against the regulatory requirements and guidance listed below to ensure that there is reasonable assurance that the systems and components affected by the proposed changes will perform their safety functions.

2.7.1.1 Regulatory Requirements

The NRC staff considered the following regulatory requirements:

Part 50 of 10 CFR, "Domestic Licensing of Production and Utilization Facilities," establishes the fundamental regulatory requirements. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides criteria for the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

In 10 CFR 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36 states that "each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

In addition, 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

The NRC staff reviewed the adequacy of the instrumentation used for SGTR mitigation against these requirements to ensure that there is reasonable assurance that this instrumentation will perform their required safety functions.

2.7.1.2 Regulatory Guidance

The NRC staff considered the regulatory guidance provided in RG 1.97, "Instrumentation for Light–Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2. Clause 1.3.1 in this RG stipulates the Design and Qualification Criteria for Category 1 instrumentations and Clause 1.3.2 for Category 2 instrumentations. Table 2 in RG 1.97 lists the suggested Category for Pressurized Water Reactor plant instrumentation.

2.7.2 Technical Evaluation

In response to an NRC staff RAI dated November 22, 2011, by letter dated December 8, 2011, the licensee provided a list of the Principal Instrumentation utilized for SGTR mitigation in Table 1 and a list of Secondary Instrumentation in Table 2.

Table 1 includes: (1) Core Exit Thermo-couples, (2) RCS Subcooling Monitor, (3) SG Water Level Indication – Narrow Range (NR), (4) Pressurizer Water Level Indication, (5) RCS Pressure Indication, and (6) SG Pressure Indication. Furthermore, Table 1 indicates that all the Principal Instrumentation is a safety-related grade for the process loop part with the indicators complying with RG 1.97, Category 1 or 2, except for SG Water Level Indication process loop which is currently nonsafety-related. The licensee stated that because the indicators of the SG Water Level Indications are currently not qualified to RG 1.97, this deficiency was entered into the plant Corrective Action Program, which subsequently determined that the SG-NR instrumentation should be classified as a RG 1.97, Type A, instrument. However, the licensee intends to perform further evaluation to fully demonstrate and document the correct RG 1.97 type classification and to ensure that the RG 1.97 design and qualification criteria are adequately implemented. In addition to the initiation of the corrective action program, the licensee made the following commitment:

> "NSPM will revise the Prairie Island Nuclear Generating Plant design and licensing bases to indicate that the Steam Generator Water Level – Narrow Range instruments are required to meet Regulatory Guide 1.97, Revision 2 requirements. This commitment will be completed prior to implementation of the Alternative Source Term license amendment."

The licensee also stated that this commitment will ensure:

- Implementation of the correct design and qualification criteria for these instruments in accordance with RG 1.97, Revision 2,

- The PINGP USAR is updated to document the correct RG 1.97 classification based on the results of the evaluation of the PINGP EOPs in relation to the technical basis for the SG-NR instrumentation, and

- The Technical Specifications will be revised in accordance with 10 CFR 50.36(c)(ii), to change the licensing bases associated with the SG-NR instruments, if required.

The NRC staff finds that the licensee's corrective action program has already identified that the SG-NR instrumentation should be qualified to RG 1.97 and has initiated further evaluation to fully demonstrate and document that the SG-NR instrumentation will comply with the correct RG 1.97 classification.

The NRC staff has evaluated the functions of all the Principal Instruments listed in Table 1 of the licensee's December 8, 2011, letter and finds that after the licensee makes the SG-NR instrument fully qualified to comply with RG 1.97, Revision 2, all the Principal Instruments in Table 1 will be fully qualified for SGTR mitigation prior to the adoption of the proposed AST Methodology.

Table 2 of the December 8, 2011, letter includes: (1) Condenser Air Ejector Radiation Monitor, (2) Steam Generator Blowdown Liquid Radiation Monitor, and (3) Main Steam Line Monitor. The Condenser Air Ejector Radiation Monitor and the Main Steam Line Monitor loops and indicators comply with RG 1.97, Category 1 or 2. However, the Steam Generator Blowdown Liquid Radiation Monitor loops and indicators are not safety grade. The NRC staff finds that,

because the Table 2 instrumentation is only for Secondary instruments, they provide additional measures for SGTR mitigation and hence need not necessarily comply with RG 1.97, Revision 2. Even though the Steam Generator Blowdown Liquid Radiation Monitor instrumentations are not fully qualified, the plant instrumentation and control design is adequately qualified for SGTR mitigation.

2.7.3 Conclusion

The NRC staff evaluated the licensee's justifications for the proposed changes specified in this section of the SE. The staff finds that, with the exception of the SG-NR instrumentation, all other Principal instruments required for SGTR mitigation are fully qualified for SGTR mitigation. After the licensee qualifies SG-NR instrumentation to meet the criteria of RG 1.97, all the Primary instrumentation listed in Table 1 of the licensee's letter dated December 8, 2011, will be adequate for SGTR mitigation and will comply with the requirements of 10 CFR 50.36. Therefore, the proposed changes, with respect to the SGTR event instrumentation, are acceptable.

2.8 Human Factors

2.8.1 Regulatory Evaluation

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 alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guidance states that defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The area of human performance deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the AST methodology. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed AST methodology.

The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in "Standard Review Plan," Chapters 13.2.1, 13.2.2, 13.5.2.1, and 18.0. Chapters 13.2.1 and 13.2.2 provide review guidance concerning the necessary training for licensed reactor operators and non licensed plant staff as described by licensees regarding changes associated with this license amendment request. Chapter 13.5.2.1 provides guidance regarding the operating procedures that will be used by the operating organization. NRC staff uses the criteria in this chapter to verify that abnormal and emergency activities continue to be conducted in a safe manner considering the changes in this license amendment request. Chapter 18.0 addresses acceptable Human Factors Engineering practices and guidelines that are used to incorporate changes to the plant's design that may apply to this license amendment request. 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," and ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions."

2.8.2 Technical Evaluation

To support their request to implement an AST at PINGP, the licensee states in its submittal that no new operator actions are credited as part of the AST analyses for design-basis accidents. The submittal also identified the manual actions that credited the AST analysis. These manual actions are included in the analysis for the SGTR event. The current licensing basis regarding a SGTR credits operator action to terminate break flow within 30 minutes. This 30-minute action time includes identifying the ruptured steam generator, isolating the ruptured steam generator, cooling down the RCS using the intact steam generator, depressurizing the RCS, and terminating safety injection flow. The information included in the AST analysis is consistent with this 30 minute operator action time to terminate break flow in the current licensing basis analysis. The licensee also stated in its submittal that it reviewed the procedures regarding the initiation of the residual heat removal (RHR) system and the time frame at which credit is taken is 45.5 hours after the initiating event as a part of the AST analysis. The current licensing basis analysis for the main steam line break takes credit 8 hours after the initiating event.

The staff concludes that crediting the operator actions is acceptable, based on the following:

- There are no new manual actions credited as a part of the AST analysis.
- The credited manual actions are not changing from the current licensing basis.
- The change in the times associated with the credit taken for procedural actions associated with initiating RHR increase and is more conservative than the current licensing basis.
- There is no required additional training associated with the credited manual action in the AST analysis.

The licensee stated that the operator actions credited in the supplemental SGTR MTO analysis are consistent with the EOPs for mitigating a SGTR. The licensee also stated that these same EOPs are used for training and simulator time validation. The plant simulator was used to determine times that it takes operators to perform various steps in the EOPs for a SGTR. Time critical operation action requirements must be met by all individuals and crews during validation. Assessments have been performed to provide reasonable assurance that the ruptured SG would not be overfilled during this event.

2.8.3 Conclusion

The NRC staff has reviewed the licensee's credited manual operator actions associated with implementing an alternative source term at PINGP. Based on its review and the considerations discussed above, the NRC staff has concluded that the proposed revision to the PINGP licensing basis with respect to the proposed changes in operator manual actions is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on April 6, 2010 (75 FR 17466). 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 amendments.

5.0 CONCLUSION

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

6.0 REFERENCES

1. Letter from M. Schimmel (Xcel) to NRC, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, License Amendment Request (LAR) to Adopt Alternative Source Term Methodology," dated October 27, 2009 (ADAMS AN ML093160583).

2. Letter from M. Schimmel (Xcel) to NRC, "Prairie Island Nuclear Generating Plant Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DRP-42 and DRP-60, Response to Requests for Additional Information Re: License Amendment Request to Adopt Alternative Source Term Methodology," dated April 29, 2010 (ADAMS AN ML101200083).

3. Letter from M. Schimmel (Xcel) to NRC, "Response to Requests for Additional Information RE: License Amendment Request to Adopt the Alternative Source Term Methodology (TAC Nos. ME2609 and ME2610)," dated May 25, 2010 (ADAMS AN ML101460064).

4. Letter from M. Schimmel (Xcel) to NRC, "Response to Requests for Additional Information (RAI) Associated with Adoption of the Alternative Source Term (AST) Methodology (TAC Nos. ME2609 and ME2610)," dated June 22, 2011 (ADAMS AN ML111740145).

5. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," dated August 1987. (Proprietary information. Not publicly available.)
6. Letter from J. F. Stang (NRC) to R. P. Powers (Indiana Michigan Power Company), "Donald C. Cook Nuclear Plant, Unit 1 and 2 – Issuance of Amendments (TAC Nos. MB0739 and MB0740)," dated October 24, 2001 (ADAMS AN ML012690136).

7. Letter from M. Schimmel (Xcel) to NRC, "Response to Requests for Additional Information (RAI) Associated with Adoption of the Alternative Source Term (AST) Methodology (TAC Nos. ME2609 and ME2610)," dated August 9, 2011 (ADAMS AN ML112220098).

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Date of issuance: January 22, 2013

ATTACHMENT 1

Tables

Table 2.1-1
Prairie Island Nuclear Generating Plant, Units 1 and 2
Control Room Atmospheric Dispersion Factors
(χ/Q Values, sec/m^3)

Source	Intake	Accident	0–2 hours	2–8 hours	8-24 hours	1–4 days	4–30 days
Unit 2 Shield Bldg Vent Stack	122 CR	LOCA, CREA	$4\ 53 \times 10^{-3}$	3.93×10^{-3}	1.73×10^{-3}	1.22×10^{-3}	9.16×10^{-4}
Unit 2 Aux. Bldg. Normal Vent Make-Up Air Intake	122 CR	RWST	2.53×10^{-2}	2.13×10^{-2}	9.65×10^{-3}	7.14×10^{-3}	6.15×10^{-3}
Unit 2 Shield Bldg Vent Stack	TSC	LOCA	1.13×10^{-3}	7.51×10^{-4}	3.05×10^{-4}	1.91×10^{-4}	1.45×10^{-4}
Unit 2 Aux. Bldg. Normal Vent Make-Up Air Intake	TSC	RWST	2.15×10^{-3}	1.12×10^{-3}	4.33×10^{-4}	3.04×10^{-4}	2.40×10^{-4}
Common Area of Aux. Bldg.	121 CR	FHA, HLD	6.71×10^{-3}	2.89×10^{-3}	1.22×10^{-3}	9.21×10^{-4}	7.44×10^{-4}
Common Area of Aux. Bldg.	122 CR	MSLB, from faulted generator	4.79×10^{-3}	3.60×10^{-3}	1.60×10^{-3}	1.21×10^{-3}	9.55×10^{-4}
Unit 2 MSSVs/PORVs – Group 1	122 CR	MSLB, SGTR, LRA, CREA	3.07×10^{-2}	2.49×10^{-2}	1.12×10^{-2}	7.78×10^{-3}	6.17×10^{-3}
Unit 2 MSSVs/PORVs – Group 2	122 CR	SGTR, LRA, CREA	2.20×10^{-3}	1.81×10^{-3}	7.97×10^{-4}	5.16×10^{-4}	4.00×10^{-4}

Table 2.1-2
PINGP Units 1 and 2
EAB and LPZ Atmospheric Dispersion Factors
(χ/Q Values, sec/m^3)

EAB	715 m		
		0-2 hours	6.49×10^{-4}
LPZ	2414 m		
		0-8 hours	1.77×10^{-4}
		8-24 hours	3.99×10^{-5}
		1-4 days	7.12×10^{-6}
		4-30 days	1.04×10^{-6}

Table 2.1-3
PINGP Units 1 and 2
Parameters and Assumptions for the LOCA

Parameter	Value
Reactor power	1852 MWt
Iodine Chemical Form in Containment	
Elemental	4.85%
Organic	0.15%
Aerosol (cesium iodide)	95%
Containment Net Free Volume	1,320,000 ft^3
Shield Building Free Air Volume	374,000 ft^3
Containment Leak Rates	
0 - 24 hours	0.15 weight %/day
> 24 hours	0.075 weight %/day
Shield Building Drawdown Time	12 minutes
ABSVZ Drawdown Time	20 minutes
Initiation of Shield Building Recirculation	22 minutes
Shield Building Recirculation Flow Rate	3,600 cfm
Shield Building Exhaust Rates (50% mixing) (22 minutes – 720 hours)	2,000 cfm
ABSVS Charcoal Filter Efficiencies	80% for elemental iodine 80% for organic iodine
ABSVS and SBVS HEPA Filter Efficiency	99% for aerosol
Containment Wetted Surface Area	246,270 ft^2
Minimum Sump Water Volume	230,000 gallons 30,745 ft^3
ESF Leakage Rate	4 gallons/hour

Table 2.1-3
PINGP Units 1 and 2
Parameters and Assumptions for the LOCA

Parameter	Value
ESF Leakage Initiation Time	0.0 minutes
ESF Leakage Iodine Flashing Factors:	
0 - 5.56 hours	4.27%
5.56 - 8.33 hours	1.87%
> 8.33 hours	3%
Iodine Species ECCS Leakage Released to the Atmosphere	97%
Elemental	3%
Organic	
RWST Leakage Rate	10 gallons/hour
RWST Leakage Iodine Flashing Factors:	
0 - 5.56 hours	4.27%
5.56 - 8.33 hours	1.87%
> 8.33 hours	3%
RWST Capacity	275,000 gallons
RWST Liquid Temperature	
Minimum	60°F
Maximum	120°F
RWST Volume at Transfer to Recirculation	29,040 gallons = 3,882 ft^3
Minimum RWST Leakage Transit Time	35.0 hours
Atmospheric Dispersion Factors	See Tables 3.4-1 & 3.4-2
Control Room Parameters	See Table 2.1-10

Table 2.1-4
PINGP Units 1 and 2
Parameters and Assumptions for the FHA

Parameter	Value
Reactor power	1852 MWt
Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12
Number of Damaged Fuel Assembly	1
Number of Fuel Assemblies in Core	121
Irradiated Fuel Decay	50 hours
Radial Peaking Factor	1.9
Iodine Chemical Form Release from Fuel to Water	
Aerosol (cesium iodide)	95%
Elemental	4.85%
Organic	0.15%
Minimum Refueling Cavity and Pool Water Depths	23 feet
Overall Effective Decontamination Factor (DF) for Iodine	200
Chemical Form of Iodine Released from Pool Water	
Elemental	57%
Organic	43%
DF of Noble Gas	1
Duration of Relase	2 hours
Activity Release Rate	39 cfm
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2

Table 2.1-4
PINGP Units 1 and 2
Parameters and Assumptions for the FHA

Parameter	Value
Control Room Parameters	See Table 2.1-10

Table 2.1-5
PINGP Units 1 and 2
Parameters and Assumptions for the HLD

Parameter	Value
Reactor power	1852 MWt
Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12
Number of Damaged Fuel Assembly	2
Number of Fuel Assemblies in Core	121
Irradiated Fuel Decay	7 days = 168 hours
Radial Peaking Factor	1.9
Iodine Chemical Form Release from Fuel to Water	
Aerosol (cesium iodide)	95%
Elemental	4.85%
Organic	0.15%
Minimum Refueling Cavity and Pool Water Depths	23 feet
Overall Effective Decontamination Factor (DF) for Iodine	200
Chemical Form of Iodine Released from Pool Water	
Elemental	57%
Organic	43%
DF of Noble Gas	1
Duration of Relase	2 hours
Activity Release Rate	39 cfm
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2

Table 2.1-5
PINGP Units 1 and 2
Parameters and Assumptions for the HLD

Parameter	Value
Control Room Parameters	See Table 2.1-10

Table 2.1-6
PINGP Units 1 and 2
Parameters and Assumptions for the MSLB

Parameter	Value
Reactor power	1852 MWt
Reactor Coolant Activity (Initial)	
Pre-Accident Iodine Spike	30 µCi/gm DE I-131
Accident-Initiated Iodine Spike	0.5 µCi/gm DE I-131
Noble Gas	580 µCi/gm DE Xe-1 33
Concurrent Iodine Spiking Factor	500
Duration of Accident-Initiated Iodine Spike	8 hours
Secondary Coolant Iodine Specific Activity	0.1 µCi/gm DE I-131
Primary Containment Leakage to Faulted SG	150 gpd
Maximum Allowed Accident Induced SG Leak Rate	1 gpm
Termination of Release from Faulted SG (Time to Cool RCS Below 212°F)	75 hrs
Termination of Release from Intact SG	45.5 hours
Iodine Form (Atmospheric Release)	
Elemental	97%
Organic	3%

Table 2.1-6
PINGP Units 1 and 2
Parameters and Assumptions for the MSLB

Parameter	Value
Steam Releases from Intact SG to Environment	
0 - 2 hours	226,414 lbm
2 - 8 hours	406,952 lbm
8 - 24 hours	796,899 lbm
24 - 45.5 hours	863,053 lbm
Intact SG Liquid Iodine Partition Coefficient	10
Steam mass released from faulted SG to the Environment	107,100 lb (Unit 1) 107,420 lb (Unit 2)
Faulted SG Dryout Time	600 seconds
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2
Control Room Parameters	See Table 2.1-10

Table 2.1-6
PINGP Units 1 and 2
Parameters and Assumptions for the MSLB

Parameter	Value
Iodine Form (Atmospheric Release)	
Elemental	97%
Organic	3%
Steam Mass Released From Ruptured SG to the Environment	80,500 lbm
Termination of Release from Ruptured SG	30 minutes
Primary Containment Leakage to Intact SG	150 gpd
Intact SG Primary-to-Secondary Leak Duration	14 hours
Steam Releases from Intact SG to Environment	
0 - 2 hours	237,100 lbm
2 - 8 hours	569,000 lbm
8 - 14 hours	416,000 lbm
SG Liquid Iodine Partition Coefficient	100
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2
Control Room Parameters	See Table 2.1-10

Table 2.1-7
PINGP Units 1 and 2
Parameters and Assumptions for the SGTR

Parameter	Value
Reactor power	1852 MWt
Reactor Coolant Activity (Initial)	
Pre-Accident Iodine Spike	30 µCi/gm DE I-131
Accident-Initiated Iodine Spike	0.5 µCi/gm DE I-131
Noble Gas	580 µCi/gm DE Xe-1 33
Concurrent Iodine Spiking Factor	335
Duration of Accident-Initiated Iodine Spike	8 hours
Secondary Coolant Iodine Specific Activity	0.1 µCi/gm DE I-131

Table 2.1-8
PINGP Units 1 and 2
Parameters and Assumptions for the CREA

Parameter	Value
Reactor power	1852 MWt
Post-CREA failed fuel	10%
Percentage of Melted Fuel Release	
Containment Leakage	
Iodine	25%
Noble Gases	100%
Primary-to-Secondary Leakage	
Iodine	50%
Noble Gases	100%
Iodine Chemical Form Release to Containment	
Aerosol (cesium iodide)	95%
Elemental	4.85%
Organic	0.15%
Containment Leak Rates	
0 - 24 hours	0.15 weight %/day
> 24 hours	0.075 weight %/day
Primary-to-Secondary Leak Duration	45.5 hours
RCS Leakage	
One Intact SG	150 gpd
Two Intact SGs	300 gpd
SG Liquid Iodine Partition Coefficient	100
Iodine Release from SG	
Elemental	97%
Organic	3%

Table 2.1-8
PINGP Units 1 and 2
Parameters and Assumptions for the CREA

Parameter	Value
Steam Releases from Intact SG to Environment	
0 - 2 hours	226,414 lbm
2 - 8 hours	406,952 lbm
8 - 24 hours	796,899 lbm
24 - 45.5 hours	863,053 lbm
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2
Control Room Parameters	See Table 2.1-10

Table 2.1-9
PINGP Units 1 and 2
Parameters and Assumptions for the LRA

Parameter	Value
Reactor power	1852 MWt
RCS Specific Activity	
Iodine	0.5 µCi/gm DE I-131
Noble Gases	580 µCi/gm DE Xe-133
Secondary Coolant Iodine Specific Activity	0.1 µCi/gm DE I-131
Post-LRA Failed Fuel	20%
Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
RCS Leakage	
One Intact SG	150 gpd
Two Intact SGs	300 gpd
SG Liquid Iodine Partition Coefficient	100
Iodine Release from SG	
Elemental	97%
Organic	3%
Steam Releases from Intact SG to Environment	
0 - 2 hours	226,414 lbm
2 - 8 hours	406,952 lbm
8 - 24 hours	796,899 lbm
24 - 45.5 hours	863,053 lbm
Atmospheric Dispersion Factors	See Tables 2.1-1 & 2.1-2
• Control Room Parameters	See Table 2.1-10

Table 2.1-10
PINGP Units 1 and 2
Control Room Parameters

Parameter	Value
CR Volume	61,315 ft^3
CRSVS Normal Flow Rate	2,000 cfm < 5 minutes
CRSVS Makeup Rate	0.0 cfm > 5 minutes
CRSVS Recirculation Flow Rate	3,600 cfm > 5 minutes
CRSVS Charcoal Filter Efficiencies	95% for elemental iodine 95% for organic iodide
CRSVS HEPA Filter Efficiency	99%
Control Room Unfiltered Inleakage	
LOCA	250 cfm
Non-LOCA events	300 cfm
CR Breathing Rate	3.5E-04 m^3/sec
Occupancy Factors	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

Table 2.1-11
PINGP Units 1 and 2
Calculated Radiological Consequences TEDE [1] (rem)

Design Basis Accident	EAB [2]	LPZ[3]	CR	TSC
Loss of Coolant Accident Dose Criteria[2]	2.58 25	2.42 25	4.52 5	4.48 5
Fuel Handling Accident Dose Criteria[3]	2.28 6.3	0.621 6.3	3.64 5	
Heavy Load Drop Accident Dose Criteria[3]	2.77 6.3	0.756 6.3	4.42 5	
Main Steam Line Break pre-accident iodine spike Dose Criteria[2] concurrent iodine spike Dose Criteria[3]	 0.111 25 0.549 2.5	 0.047 25 0.258 2.5	 0.788 5 4.04 5	
Steam Generator Tube Rupture Accident pre-accident iodine spike Dose Criteria[2] accident initiated iodine spike Dose Criteria[3]	 1.09 25 0.961 2.5	 0.299 25 0.269 2.5	 4.67 5 3.45 5	
Control Rod Ejection Accident Dose Criteria[3]	0.669 6.3	0.387 6.3	3.91 6.3	
Locked Rotor Accident Dose Criteria[2]	0.487 25	0.271 25	4.33 5	

[1] Total effective dose equivalent
[2] From 10 CFR 50.67
[3] From SRP 15.0.1

ATTACHMENT 2

Summary of License Conditions

(Applicable to Units 1 and 2)

Additional Conditions	Implementation Date
The Alternative Source Term (AST) License Amendments 206/193 will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed
NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed
Implement a physical plant modification or procedure modification that will ensure the 121 Laundry Fan exhaust flow path is not a potential source of post-accident radioactive release through the Auxiliary Building Ventilation Exhaust stack.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed

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

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 206 to DPR-42
2. Amendment No. 193 to DPR-60
3. Safety Evaluation

cc w/encls: Distribution via ListServ

DISTRIBUTION:

PUBLIC
LPL3-1 r/f
RidsNrrPMPrairieIsland Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn3MailCenter Resource
RidsNrrDraAhpb Resource
RidsNrrDraAadb Resource
RidsOgcRp Resource
RidsNrrDeEicb Resource
RidsNrrDeEvib Resource
RidsNrrDeEmcb Resource
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RidsNrrDssStsb Resource
RidsNrrDssScvb Resource
BHeida, NRR
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WJessup, NRR
HWalker, NRR
SMazumdar, NRR
MMitchell, NRR

ADAMS Accession No.: ML112521289 *SE memo dated

OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	DSS/STSB/BC	DCI/CSGB/BC*	DE/EEEB/BC (A)*
NAME	TWengert	BTully/SRohrer	RElliott	RTaylor	RMathew
DATE	01/04/2013	01/03/2013	11/21/12	08/13/11	03/23/11
OFFICE	DE/EICB/BC*	DRA/AADB/BC*	DIRS/IHPB/BC	DSS/SRXB/BC*	DE/EMCB/BC*
NAME	GWilson /*RStattel for*	TTate	UShoop	AUlses	MKhanna
DATE	12/23/11	10/01/12	11/20/12	10/19/11	07/25/11
OFFICE	DSS/SCVB/BC*	OGC NLO	NRR/LPL3-1/BC	NRR/LPL3-1/PM	
NAME	RDennig	BMizuno	RCarlson	TWengert	
DATE	09/15/11	12/20/12	01/17/13	01/22/13	