October 20, 2016

Mr. Joel P. Gebbie
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF TSTF-490, REV. 0, "DELETION OF E-BAR DEFINITION AND REVISION TO REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY TECHNICAL SPECIFICATION" AND IMPLEMENTATION OF FULL-SCOPE ALTERNATIVE SOURCE TERM (CAC NOS. MF5184 AND MF5185)

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 332 to Renewed Facility Operating License No. DPR-58 and Amendment No. 314 to Renewed Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated November 14, 2014, as supplemented by letters dated February 12, July 17, August 24, August 28, November 16, December 17, 2015, and February 19, May 6, July 12, and September 15, 2016.

The amendments replace the current CNP Units 1 and 2 TS 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit is based on a new dose equivalent Xenon (Xe)-133 definition that replaces the current E - Average Disintegration Energy definition. The amendments also revise the current dose equivalent Iodine (I)-131 definition to allow the use of additional thyroid dose conversion factors. Additionally, the CNP, Units 1 and 2, licensing basis and TSs are revised to adopt the alternative source term as allowed in Title 10 of the *Code of Federal Regulations*, Section 50.67.

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

Sincerely,

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 332 to DPR-58
2. Amendment No. 314 to DPR-74
3. Safety Evaluation

cc w/encls: Distribution via ListServ

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 332
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 14, 2014, as supplemented by letters dated February 12, July 17, August 24, August 28, November 16, December 17, 2015, February 19, May 6, July 12, and September 15, 2016, 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-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 332, are hereby incorporated in this license. The licensee shall operate the

Enclosure 1

facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to Renewed Facility Operating License No. DPR-58 and Technical Specifications

Date of Issuance: October 20, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 332

DONALD C. COOK NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Renewed Facility Operating License No. DPR-58 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of 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
3.4.16-1	3.4.16-1
3.4.16-2	3.4.16-2
3.4.16-3	--
5.5-5	5.5-5
5.5-10	5.5-10
5.5-14	5.5-14

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, 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; and

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

C. This renewed 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, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3304 megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 332, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Less than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013,

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify 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. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine 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 thyroid dose conversion factors from 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."
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."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 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 ≤ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 60 μCi/gm <u>OR</u> DOSE EQUIVALENT XE-133 not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 215.1 μCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm for an individual SG, for a total leakage of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

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

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 314
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 14, 2014, as supplemented by letters dated February 12, July 17, August 24, August 28, November 16, December 17, 2015, February 19, May 6, July 12, and September 15, 2016, 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-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 314, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to Renewed Facility Operating License No. DPR-74 and Technical Specifications

Date of Issuance: October 20, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 314

DONALD C. COOK NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Renewed Facility Operating License No. DPR-74 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of 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
3.4.16-1	3.4.16-1
3.4.16-2	3.4.16-2
3.4.16-3	--
5.5-5	5.5-5
5.5-10	5.5-10
5.5-14	5.5-14

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, 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; and

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

C. This renewed 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, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3468 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to the renewed operating license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed operating license shall be completed. Attachment 1 is an integral part of this renewed operating license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 314, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) Leak Testing of Emergency Core Cooling System Valves

Indiana Michigan Power Company shall prior to completion of the first inservice testing interval leak test each of the two valves in series in the

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify 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. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine 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 thyroid dose conversion factors from 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."
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."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 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 ≤ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 60 μCi/gm. <u>OR</u> DOSE EQUIVALENT XE-133 not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 215.1 μCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm in an individual SG, for a total leakage rate of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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

5.5.15 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 332 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

AND

AMENDMENT NO. 314 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated November 14, 2014 (Reference 1), as supplemented by letters dated February 12, July 17, August 24, August 28, November 16, December 17, 2015, February 19, May 6, July 12, and September 15, 2016 (References 2 through 11), Indiana Michigan Power Company, LLC (I&M, the licensee) requested license amendments for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The proposed amendments would adopt Technical Specifications Task Force Traveler (TSTF)-490, Revision 0 (Reference 12). The proposed amendments would also implement an alternative source term (AST) radiological analysis methodology, as described in Regulatory Guide (RG) 1.183 (Reference 13).

These amendments would replace the current CNP, Units 1 and 2, technical specification (TS) 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xenon-133 (Xe-133) definition that would replace the current E - Average Disintegration Energy (E Bar) definition. In addition, the current dose equivalent Iodine-131 (I-131) definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs). Additionally, the CNP licensing basis and TSs would be revised to adopt the AST as allowed in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

The supplemental letters dated July 17, August 24, August 28, November 16, December 17, 2015, February 19, May 6, July 12, and September 15, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 31, 2015 (80 FR 17091).

2.0 REGULATORY EVALUATION

2.1 Background on Alternative Source Term Requirements

The evaluation of the release of fission products into containment (called "source term") is used for judging the acceptability of both the plant site and the effectiveness of engineered safety features (ESFs). In the past, power reactor licensees have typically used U.S. Atomic Energy Commission (AEC) Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference 14), as the basis for design basis accident (DBA) source terms. DBAs are based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events which would result in potential hazards not exceeded by those from any accident considered credible. The DBA offsite radiological dose consequences are evaluated against the guideline dose values, in terms of whole body and thyroid dose, given in 10 CFR 100.11, "Determination of Exclusion Area Boundary [EAB], Low Population Zone [LPZ], and Population Center Distance," which makes reference to TID-14844.

In December 1999, the U. S. Nuclear Regulatory Commission (NRC or Commission) issued the new regulation 10 CFR 50.67, "Accident source term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST.

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," whenever a holder of an operating license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. As stated in paragraph (a) of 10 CFR 50.92, "Issuance of amendment," in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. As stated in 10 CFR 50.57, "Issuance of operating license," an operating license may be issued upon finding, among other things, that there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and there is reasonable assurance that such activities will be conducted in compliance with the regulations in this chapter; and the issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

Pursuant to 10 CFR 50.67(b)(1), a licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

As stated in 10 CFR 50.67(b)(2), the NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sieverts] (25 rem [roentgen equivalent man])[1] TEDE.

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

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

Regulatory guidance for the implementation of the AST is provided in RG 1.183 and 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 RG provides guidance on an acceptable AST application 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.

Regulatory Position 1.2 of RG 1.183, indicates that a complete implementation of an AST would upgrade all existing radiological analyses and address all characteristics of the AST; namely, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the total effective dose equivalent (TEDE) as the new acceptance criteria.

The TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11, to determine the EAB, LPZ, and population center distance. In addition, holders of operating licenses using an AST under 10 CFR 50.67 shall meet the requirements of Appendix A to 10 CFR Part 50, General Design Criteria (GDC), Criterion 19, "Control Room," regarding control room (CR) access and occupancy. This applies not only to the analyses performed in the application, which may only include a subset of the plant analyses, but also to all future design basis analyses. As a minimum for full implementation, the maximum credible dose consequence loss-of-coolant accident (LOCA) must be determined and then analyzed using the guidance in RG 1.183, Appendix A.

A design basis radiological consequence LOCA analysis is intended to be based upon a major accident, or possible event, resulting in dose consequences not exceeded by those from any

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1 The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in 10 CFR 50.67 as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

accident considered credible. Historically, this accident analysis, which is performed to show compliance with the dose criteria specified in 10 CFR 50.67, is referred to as the maximum hypothetical accident (MHA). It should be noted that the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," ensure that the emergency core cooling system (ECCS) will prevent significant core damage during a design basis LOCA. Notwithstanding, the requirements of 10 CFR 50.46, the MHA for dose consequence determinations deterministically assumes a substantial core melt with an appreciable release of fission products into the containment. Therefore, the MHA is a conservative surrogate to enable a deterministic evaluation of the response of a facility's ESFs such as containment systems. While the maximum hypothetical dose consequences from the DBA LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider the dose consequences from other DBAs such as the fuel handling accident (FHA). All design basis dose consequence accident analyses are performed in an intentionally conservative manner in order to compensate for known uncertainties in accident progression, airborne activity product transport, and atmospheric dispersion.

As stated in Regulatory Position 5.2 of RG 1.183, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. The inclusion or exclusion of a particular dose consequence 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.

Section C.1.3.5 of RG 1.183 states that the licensees may use the AST or TID-14844 (Reference 14) assumptions for performing the required environmental qualification (EQ) analyses to show that the equipment remains bounding. RG 1.183, Section C.1.3.5, further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID-14844) on EQ doses, pending the outcome of the evaluation of the generic issue. According to RG 1.183, maintaining pH basic will minimize re-evolution of iodine from the suppression pool water.

Regulatory Position 1.3.2 of RG 1.183 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. Additionally, Section 5.1.3 of 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.

Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved and documented in a memo dated April 30, 2001, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" (Reference 15) and in the NUREG-0933, Supplement 25, "A Prioritization of Generic Safety Issues," (Reference 16).

NUREG-0933, Issue 187, indicated that for equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses. It was concluded that there was no clear basis for back fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement.

The conclusion to Generic Issue 187 states the following:

> The staff concluded that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary. Therefore, in consideration of the cited references, the staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ considerations.

2.2 Background on TSTF-490

In this LAR, the licensee proposed to implement TSTF change traveler TSTF-490 (Reference 12). The Notice of Availability for TSTF-490 was published in the *Federal Register* on March 19, 2007 (72 FR 12838) (Reference 17). TSTF-490 was announced for availability as part of the consolidated line item improvement process. Since 2007, numerous licensees have adopted the Traveler. During the LAR reviews, some technical issues were identified, which were addressed through requests for additional information (RAIs). In order to streamline the process and avoid the issuance of RAIs, the TSTF submitted a revision to the Traveler (Reference 18). The NRC staff accepted Revision 1 of TSTF-490 for review. However, during the review of the revision, some additional technical concerns were identified. In lieu of continuing the review of Revision 1, the TSTF decided to withdraw it. By internal NRC memorandum dated March 14, 2012 (Reference 19), the NRC staff indicated that LARs related to TSTF-490, Revision 0, can be accepted for review, but will be handled through the normal LAR review process.

The proposed changes would replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on dose equivalent Xe-133 and would take into account only the noble gas activity in the primary coolant. 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. Since this submittal also contains the request to implement AST methodology, the definition of dose equivalent I-131 is revised using DCFs from the Environmental Protection Agency (EPA) Federal Guidance Report (FGR)-11 rather than from TID-14844, which is currently listed in the dose equivalent I-131 definition. The use of DCFs from FGR-11 is consistent with RG 1.183 guidance. As such, the NRC staff used the regulatory guidance provided in NUREG-0800, "Standard Review Plan" (SRP), Section 15.0.1, and the methodology and assumptions stated in RG 1.183. 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 generally 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 it is considered to be insignificant in relation to the dose consequence resulting from the release of fission products from the damaged fuel.

2.3 Applicable Regulations and Guidance

The following explains the applicability of 10 CFR Part 50, Appendix A, "GDC for Nuclear Power Plants," for CNP. CNP was designed and constructed to meet the intent of proposed GDC, published in the *Federal Register* on July 11, 1967. The Final Safety Analysis Report had been filed with the Commission when revisions of the GDC were published in February 1971, and July 7, 1971. In 1973, the AEC reviewed the plant design against the most recent GDC and concluded that the design meets these criteria. Therefore, the extent to which the Appendix A GDC have been invoked can be found in specific sections of the CNP Updated Final Safety Analysis Report (UFSAR) and in other CNP licensing basis documentation, such as license amendments.

Based on a review of UFSAR, Section 1.4, "Plant Specific Design Criteria (PSDC)," the NRC staff identified the following PSDCs as being applicable to the proposed amendment:

- Criterion 11, Control Room, which states in part:

> The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shutdown and maintain safe control of the facility without excessive radiation exposures of personnel.

- Criterion 39, Emergency Power, which states in part:

> An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

The following additional NRC requirements and guidance documents are also applicable to the NRC staff's review of the license amendment request (LAR):

- Pursuant to 10 CFR 50.34(a)(4), each application for a construction permit shall include a preliminary safety analysis report providing, among other things, a preliminary analysis and evaluation of the design and performance of Structures, Systems, and Components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

- Pursuant to 10 CFR 50.34(b), each application for an operating license shall include a final safety analysis report (FSAR). The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. Under 10 CFR 50.34(b)(2), the FSAR shall include a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- 10 CFR 50.36, "Technical Specifications," establishes the regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. Surveillance requirements are defined as 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 the LCO will be met.

- Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," whenever a holder of an operating license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Accordingly, as part of the license amendment request, licensees perform evaluations to ensure that their safety analyses remain bounding or continue to meet the applicable acceptance criteria. Licensees confirm that key inputs to the safety analyses, such as neutronic and thermal hydraulic (TH) parameters, are bounded, or if key safety analysis parameters are not bounded, licensees perform reanalyses or re-evaluations of the affected transients or accidents to ensure that the applicable acceptance criteria are satisfied.

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires, in paragraph (a), that the licensee establish a program for qualifying the electrical equipment important to safety. Pursuant to 10 CFR 50.49(b), electric equipment important to safety covered by 10 CFR 50.49 is (1) safety-related electric equipment relied upon to remain functional during and following design basis events to ensure (A) the integrity of the reactor coolant pressure boundary; (B) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (C) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11 of the Commission's regulations, as applicable. Also covered is nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in the Commission's regulations. Last, certain post-accident monitoring equipment is covered. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

- 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

- Safety Guide 23, "Onsite Meteorological Programs" (Reference 20), represents the guidance on which the onsite meteorological monitoring program, as described for the current licensing basis in the UFSAR, was established.

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1 (Reference 21), presents criteria for an acceptable onsite meteorological monitoring program and the resulting meteorological database that may be used as input to atmospheric dispersion estimates.

- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units in Post-Accident Engineered-Safety Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Reference 22), presents methods acceptable to the NRC for meeting CR occupancy protection requirements. RG 1.52 also provides information on acceptable maximum allowable methyl iodide penetration and filter efficiency for the CR emergency ventilation charcoal absorber.

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

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Reference 24), provides guidance on methods acceptable to the NRC staff for determining atmospheric relative concentration (χ/Q) values in support of design basis CR radiological habitability assessments at nuclear power plants performed in support of applications for licenses and LARs.

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors" (Reference 25), provides guidance on establishing the licensing bases for the CR, its associated ventilation systems, and those located in, traversing or serving areas adjacent to the CR.

- RG 1.203, "Transient and Accident Analysis Methods," (Reference 26) provides guidance to the industry for the analysis of transient behavior. In particular, licensees must include a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

- NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Reference 27), is an acceptable method for implementing the agency's emergency planning regulations.

- NUREG-0696, "Functional Criteria for Emergency Response Facilities" (Reference 28), indicates that the technical support center should have the same habitability as the CR.

- NUREG–0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" (Reference 29), provides methodologies for evaluating sequential hourly meteorological data used to characterize onsite meteorological conditions including input to atmospheric dispersion modeling analyses.

- NUREG/CR–2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-basis Accidental Releases of Radioactive Materials from Nuclear Power Stations" (Reference 30).

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 31), provides estimates of AST that were more physically based and that could be applied to a pressurized water reactor (PWR).

- NUREG/CR-5950, "Iodine Evolution and pH Control" (Reference 32).

- NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays" (Reference 33).

- NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels" (Reference 34).

- NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Reference 35), is incorporated into the analysis code RADionuclide Transport and Removal and Dose Estimation (RADTRAD).

- NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) (Reference 36), provides user's guidance for the ARCON96 dispersion model.

- NUREG/CR-7220, "SNAP/RADTRAD 4.0: Description of Models and Methods," (Reference 37), provides a description of the models and methods exercised by the NRC code, Symbolic Nuclear Analysis Package/RADionuclide Transport, Removal and Dose Estimation (SNAP/RADTRAD), Version 4.0, to perform confirmatory DBA calculations.

The following sections of the SRP are also applicable:

- Section 2.3.3, "Onsite Meteorological Measurements Program" (Reference 38), provides the acceptance criteria for the review of the onsite meteorological monitoring program and the resulting meteorological database that may be used as input to atmospheric dispersion estimates.

- Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases" (Reference 39), provides the acceptance criteria for reviewing estimates of atmospheric dispersion factors at the EAB, outer boundary of the LPZ, and at the CR for, among other things, postulated design-basis accidental radioactive airborne releases;
- Section 6.4, "Control Room Habitability System" (Reference 40).
- Section 6.5.2, "Containment Spray as a Fission Product Cleanup System" (Reference 41), provides the acceptance criteria regarding the systems used to minimize iodine re-evolution as presented in the licensee's re-analysis of the radiological consequences for the LOCA.
- Section 11.1, "Coolant Source Terms" (Reference 42), as it relates to design basis source terms for the RCS, which is typically based on 0.25 – 1 percent fuel defects;
- Section 13.3, "Emergency Planning" (Reference 43).
- Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 44), provides guidance to the NRC staff for the review of the models, assumptions, and parameter inputs used by the licensee for the calculation of the AST radiological consequences.
- Section 15.0.2, "Review of Transient and Accident Analysis Methods" (Reference 45), describes the NRC staff's review process and acceptance criteria for analytical models and computer codes used by licensees to analyze accident and transient behavior. The purpose of the NRC staff review for this SRP section is to verify that the evaluation model is adequate to simulate the accident under consideration.

As stated in RG 1.183, there may be applications of the accident source term identified in various licensee commitments, such as plant-specific licensing commitments made in response to NUREG-0737, "Clarification of [Three Mile Island (TMI)] Action Plan Requirements" (Reference 46). Applicable sections of NUREG-0737 include the following:

- NUREG-0737 II.B.2, "Post-accident Access Shielding," as it relates to post-accident radiation exposure incurred while performing necessary plant operations outside of the CR.
- NUREG-0737 II.B.3, "Post-accident Sampling Capability," as it relates to post-accident radiation exposure during sampling operations.
- NUREG-0737 II.F.1, "Additional Accident-Monitoring Equipment," as it relates to the ability of the monitors to operate during and following an accident and perform the intended function in the accident environment.
- NUREG-0737 III.D.1.1, "Leakage Control," as it relates to post-accident radiation exposure.
- NUREG-0737 III.A.1.2, "Emergency Response Facilities," as it relates to maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.
- NUREG-0737 III.D.3.4, "Control Room Habitability," as it relates to maintaining the CR in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

2.4 Technical Specification (TS) Changes

2.4.1 Proposed TS Changes for Adoption of AST and TSTF-490

The licensee has proposed the following changes to the TSs.

- Revise the definition of DOSE EQUIVALENT I-131.
- Delete the definition E-AVERAGE DISINTEGRATION ENERGY.
- Add a new TS definition, DOSE EQUIVALENT XE-133.
- Revise limiting condition for operation (LCO) 3.4.16 to require "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits."
- Revise LCO 3.4.16, "Applicability," to specify the LCO is applicable in MODES 1, 2, 3 and 4.
- Modify the TS 3.4.16 ACTIONS Table as follows:
  - Condition A is modified to delete the reference to Figure 3.4.16-1, and define an upper limit that is applicable to all power levels.
  - Condition B is modified to state "Required Action and associated Completion Time of Condition A not met, OR DOSE EQUIVALENT I-131 > 60 [microcuries per gram (μCi/gm)], OR DOSE EQUIVALENT XE-133 not within limit."
  - The proposed ACTION and COMPLETION TIME for the new Condition B has been revised with the requirement to be in MODE 3 within 6 hours and MODE 5 within 36 hours.
- Revise Surveillance Requirement 3.4.16.1 to verify the limit for DOSE EQUIVALENT Xe-133 is less than or equal to 215.1 μCi/gm.
- Revise Surveillance Requirement 3.4.16.2 to delete the note that the surveillance is only required to be performed in MODE 1.
- Delete Surveillance Requirement 3.4.16.3 on gross specific activity.
- Delete Figure 3.4.16-1.
- Revise Section 5.5.7.b.2 to clarify that steam generator (SG) leakage is limited to 0.25 gallons per minute (gpm) per SG.
- Revise Section 5.5.9.c to increase the maximum allowable methyl iodide penetration for the CR Emergency Ventilation charcoal adsorber to 2.5 percent.
- Revise Section 5.5.14.c to reduce the maximum allowable containment leakage rate to 0.18 percent per day.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed changes. The staff performed independent calculations to confirm the conservatism of the licensee's analyses. The findings of this safety evaluation (SE) are based on the descriptions of the analyses and other supporting information submitted by the licensee.

3.1 Design Basis Accidents (DBAs)

The licensee proposed a full implementation of the AST, in accordance with the guidance in RG 1.183 and SRP, Section 15.0.1. The scope of the licensee AST analyses included the PWR

DBAs identified in the CNP UFSAR, Chapter 14, "Safety Analysis" by applying the guidance described in RG 1.183 for PWR DBAs that could potentially result in significant CR and offsite doses. AST analyses were performed for the following accidents and events:

- LOCA
- FHA
- MSLB
- SGTR
- Locked Rotor Accident (LRA)
- Control Rod Ejection (CRE)
- Waste Gas Decay Tank (WGDT) Rupture
- Volume Control Tank (VCT) Rupture.

By letter dated November 14, 2002 (Reference 47), the NRC approved a selective implementation of the AST for CNP through the issuance of Amendment Nos. 271 and 252, for Units 1 and 2, respectively. Selective implementation is a modification of the facility design basis that either is based on one or more of the characteristics of the AST, or entails re-evaluation of a limited subset of the design basis radiological analyses. The amendment replaced the TID-14844 accident source term used in the DBA analyses only for CR habitability with the AST pursuant to 10 CFR 50.67 by applying the guidance described in RG 1.183. Other relevant license amendments considered by the staff include:

- By letter dated October 24, 2001, the NRC issued Amendment Nos. 256 for CNP, Unit 1 and 239 for Unit 2 (Reference 48). The amendments approved the October 24, 2000, LAR to incorporate a supplemental methodology for the analysis of SG overfill following a SGTR.

- By letter dated October 24, 2001, the NRC issued Amendment Nos. 257 for CNP, Unit 1, and 240 for Unit 2 (Reference 49). The amendments approved a portion of the June 12, 2000, LAR which dealt with Generic Letter 99-02, "Laboratory Testing of Nuclear-grade Activated Charcoal."

- By letter dated November 13, 2001, the NRC issued Amendment Nos. 258 for CNP, Unit 1, and 241 for Unit 2 (Reference 50). The amendments approved a portion of the June 12, 2000, LAR to revise the FHA with an AST pursuant to 10 CFR 50.67 using the methodology described in RG 1.183.

- By letter dated November 14, 2002, the NRC issued Amendment Nos. 271 for CNP, Unit 1, and 252 for Unit 2 (Reference 47). The amendments approved a portion of the June 12, 2000, LAR that replaced the TID-14844 accident source term used in design-basis radiological analyses for CR habitability with an AST pursuant to 10 CFR 50.67 using the methodology described in RG 1.183. The licensee retained many aspects of the licensing basis from this initial AST application in the current AST LOCA evaluation.

- By letters dated December 20, 2002 (Reference 51), and May 2, 2003 (Reference 52), the NRC issued Amendment Nos. 273 for CNP, Unit 1, and 259 for Unit 2, respectively. Amendment No. 273 approved the June 28, 2002, LAR that requested an increase of the licensed reactor core power level by 1.66 percent from 3250 megawatts thermal

(MWt) to 3304 MWt for CNP, Unit 1. Amendment No. 259 approved the November 15, 2002, LAR that requested an increase of the licensed reactor core power level by 1.66 percent from 3411 MWt to 3468 MWt for Unit 2.

This SE addresses the impact of the proposed changes on the previously analyzed DBA radiological consequences for the CR as well as the off-site locations at the EAB and LPZ. While reviewing the applicant's analysis to determine if the analysis demonstrated with reasonable assurance that the criteria in 10 CFR 50.67(b)(2)(i)-(iii) are met, the NRC considered the accident-specific criteria in Chapter 15 of the SRP, and PSDC 11, as supplemented by Section 6.4 of the SRP.

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the LPZ and the integrated dose to a CNP CR operator 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 3.0, computer code. The RADTRAD radiological consequence computer code was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performs independent confirmatory dose evaluations, as needed, using the models and methods exercised in the NRC code, the Symbolic Nuclear Analysis Package/RADionuclide Transport, Removal and Dose Estimation (SNAP/RADTRAD), Version 4.0. The code consists of the SNAP Model Editor graphical user interface, the SNAP/RADTRAD plugin and the RADionuclide Transport, Removal and Dose Estimation analytical code. A description of the models and methods exercised by SNAP/RADTRAD 4.0, can and be found in NUREG/CR-7220 (Reference 37). The SNAP/RADTRAD code was developed for the NRC to estimate transport and removal of radionuclides and dose at selected receptors. The SNAP/RADTRAD code can be used to estimate the radionuclide release from the containment using either the NRC TID-14844 or NUREG-1465 (Reference 31) source terms and assumptions, or a user-specified source term and assumptions. In addition, the code can account for a reduction in the quantity of radioactive material released due to containment spray (CTS), natural deposition, filters, and other ESFs. The code uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario.

3.2 Design Basis Accident (DBA) Source Terms

The licensee developed a new source term for the CNP DBA reactor core source term using the ORIGEN-ARP code. The FHA source term is derived from the core source term, modifying the value based on the number of fuel assemblies and radial peaking factor. The RCS source term is established with input from the reactor core source term and modeling in the GOTHIC code. The CNP source terms were provided in the letter dated November 16, 2015 (Reference 6).

3.2.1 Reactor Core Source Term

The licensee analyzed the design basis events involving postulated fuel failures using a source term for Westinghouse 17 x 17 VANTAGE 5 fuel. RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states the following:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN2 or ORIGEN-ARP.

The equilibrium core inventory source term used in the CNP AST analysis was determined using the ORIGEN2 computer code and is based on a core power level of 3480 MWt. This core power level is equal to the CNP, Unit 2, licensed rated thermal power level of 3468 MWt plus 0.34 percent to account for measurement uncertainty. This Unit 2 core power level bounds the Unit 1 core power level. Therefore, in accordance with RG 1.183, the dose consequence analyses are based on a reactor power level of 3480 MWt, based on 100.34 percent of the rated thermal power of 3468 MWt.

3.2.2 Loss-of-Coolant Accident (LOCA) Source Term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position 3, regarding the reactor core inventory, release fractions, and timing for the evaluation of its dose consequence LOCA. The radioactivity released into the containment is assumed to terminate at the end of the early in-vessel phase, which occurs at 2 hours after the onset of a LOCA.

For this analysis, it is assumed that the core inventory is released to the containment atmosphere according to the release fractions and timing of release phases given in RG 1.183 for the gap release and early in-vessel damage phases. During the progression of the LOCA, some fission products released from the fuel will be carried to the containment sump via spillage from the RCS or by the CTS and natural processes. With the exception of noble gases, the fission products released from the fuel are assumed to instantaneously and homogeneously mix in the containment sump water. Noble gases are assumed to remain in the containment atmosphere. Plate-out on internal surfaces is not credited. Of the radioiodine released to the containment in a postulated accident, 95 percent is assumed to be cesium iodide, 4.85 percent is assumed to be elemental iodine, and 0.15 percent is assumed to be organic iodide, consistent with RG 1.183. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form. The post-LOCA containment sump pH is designed to be a minimum of 7.0, therefore, re-evolution of particulate iodine into elemental iodine is not considered and is consistent with RG 1.183. The post-LOCA containment sump pH is discussed further in Section 3.3.3 of this SE.

3.2.3 Non-LOCA Gap Release Source Term

To determine the amount of isotopes released during non-LOCA events, gap fractions and the number of failed fuel rods are used to determine the source term of the accident. Gap fractions are the fraction of a given isotope residing in the gap between the fuel pellet and the fuel cladding. When fuel damage is assumed to occur, the gas in the gap is released. For non-LOCA events, such as the FHA, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in RG 1.183 Table 3, Non-LOCA Fraction of Fission Product Inventory in Gap.

The release fractions from RG 1.183, Table 3, are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. As stated in RG 1.183, the release fractions associated with the core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved fuel with a peak burnup of 62 gigawatt days per metric ton of uranium (GWD/MTU) provided that the maximum linear heat generation rate (LHGR) does not exceed the LHGR limit of 6.3 kilowatt per foot (kW/ft) peak rod average power for burnups exceeding 54 GWD/MTU.

To account for future core design margin, the licensee proposed to increase the non-LOCA gap release fractions intended for high-burnup fuel rods (i.e., 54 GWD/MTU that exceed the 6.3 kW/ft LHGR limit stated in RG 1.183, Table 3). Footnote 11 of Table 3 of RG 1.183 states:

> The release fractions listed here have been determined to be acceptable for use with currently approved [light-water reactor (LWR)] fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

The licensee stated in the submittal that 150 rods are assumed to fail with burnups above the 6.3 kW/ft limit to allow for future core design margin. The licensee then stated in the submittal that the gap fractions will be doubled for 100 percent of the rods in the affected assemblies, which was a method that had been previously approved by the NRC (Reference 53). The NRC staff sent RAI-ARCB-2 to confirm that CNP was within the assumptions of the referenced NRC-approved analysis. The licensee responded in its supplement submitted on December 17, 2015 (Reference 7), that to stay within the assumptions of the analysis, the licensee will confirm the LHGR stays below 6.3 kW/ft on a per cycle basis during the reload safety analysis checklist. In this way, the licensee will ensure the power history assumed stays bounding. The licensee also stated in response to RAI-ARCB-2 that if the LHGR exceeds the limit, the fuel vendor will be requested to perform additional analyses to ensure the assumptions in Reference 53 remain valid.

To calculate the non-LOCA gap release fraction for an AST, the NRC staff considers acceptable the use of approved methodologies and bounding power histories. The NRC staff endorses the American National Standards Institute/American Nuclear Society-5.4-1982 model, "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, licensees can perform new gap release fraction calculations that would be acceptable to the NRC staff.

The gap fractions for non-LOCA events are acceptable for assessing the dose consequences because the peak LHGR of 6.3 kW/ft ensures the original values in RG 1.183, Revision 0, are met for all burnups. Further conservatism is added by doubling the gap fractions, and the reload design process confirms that the peak LHGR remains bounding.

3.2.4 Reactor Coolant System (RCS) Source Term

The equilibrium nuclide concentration in the RCS is calculated based on the core inventory. The rate of nuclide release from the core to the reactor coolant for applicable isotopes is calculated from fission product escape rate coefficients and assumes that 1 percent of fuel rods have defects. With this isotopic production rate in the coolant established, RCS concentrations are calculated with a hydraulic model of the RCS purification system using GOTHIC. The GOTHIC code is used to simulate the RCS purification system to determine the relative concentrations of nuclides in the reactor coolant, and is also used to calculate the time-dependent refueling water storage tank (RWST) temperature due to back leakage from the containment sump. This model accounts for; radioactive decay and daughter production, removal of nuclides by the demineralizers, degassing in the VCT, and dilution of the nuclide concentration by normal makeup for RCS boron control. The GOTHIC model is run until the radionuclide concentrations reach equilibrium values. The GOTHIC output provides the relative distribution of isotopic concentrations in the RCS. These values are then manually scaled such that the iodine activities match the dose equivalent I-131 limit of 1.0 μCi/gm specified in the TS.

3.2.5 DBA Source Terms Conclusion

The NRC staff reviewed the licensee's determinations of the reactor core source term, LOCA source term, non-LOCA gap release source term, and RCS source term. The NRC staff finds that the analyses were performed in accordance with RG 1.183, using NRC-approved methods, and are therefore acceptable.

3.3 LOCA

3.3.1 LOCA Overview

The radiological consequence design basis LOCA is a surrogate for the MHA, required by regulation and postulated from considerations of possible accidental events that would result in potential hazards (dose consequences) not exceeded by those from any accident considered credible. The AST based on NUREG-1465, like its predecessor the TID-14844 source term, is used to evaluate the ESFs used to protect public health and safety in the unlikely event of a nuclear accident that results in substantial meltdown of the core with subsequent release of appreciable quantities of fission products. The MHA source term described by TID-14844 or NUREG-1465 is derived from a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping referred to as a LOCA. The AST LOCA accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling, which results in a significant amount of core damage as specified in RG 1.183. Unlike the DBA LOCA used to evaluate the ECCS requirements of 10 CFR 50.46, this general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis ECCS or design basis transient analyses.

The CNP LOCA accident is described in the UFSAR, Section 14.3.1, "Large Break LOCA Analysis." The licensee's AST LOCA analysis assumes a double-ended rupture of the largest reactor coolant pipe. The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously into the RCS and then throughout the free air volume of the containment. When evaluating an AST LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. I&M used the AST release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183, Tables 2, 4, and 5.

3.3.2 LOCA Transport

As described in Section 5 of the CNP UFSAR, "Containment Systems," the containment has the capability to maintain functional integrity during and following peak transient pressures and temperatures which would occur following any postulated LOCA. The CNP containment has an ice condenser reactor containment design. The ice condenser design concept is to rapidly absorb the energy release in the containment during a LOCA by condensing the steam in a low temperature heat sink. This heat sink, located inside the containment, consists of a suitable quantity of borated ice in a cold storage compartment. The containment design internal pressure is 12 pounds per square inch gage. The effects of pipe rupture in the primary coolant system, up to and including a double-ended rupture of the largest pipe as well as a rupture of the main steam line, are considered in determining the peak accident pressure. The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended severance.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dumping may occur. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves. Emergency feedwater flow via the auxiliary feedwater pumps is initiated on the reactor trip signal. The secondary flow aids in the reduction of RCS pressure. When the RCS depressurizes to the accumulator gas cover pressure, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initiation of the accident and effects of pump coast down are included in the blowdown analyses.

In its evaluation of the proposed licensing basis AST LOCA, the licensee considered dose contributions from the following potential radioactive material release pathways:

- Containment purge
- Containment leakage
- ESF leakage to the auxiliary building
- ESF leakage to the RWST

For all four cases, the CR ventilation system is automatically placed into the pressurization mode upon receipt of the safety injection signal.

For the purposes of this analysis, modeling of the ice condenser containment design is effectively divided into sub-compartments that are fully or partially enclosed volumes within the containment. If a pipe break accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures. Each sub-compartment is designed to limit the adverse effects of a postulated high-energy pipe rupture within them. These sub-compartments include the SG enclosure, fan accumulator room, pressurizer enclosure, loop sub compartment and upper and lower reactor cavity, and a "dead-ended region."

The containment air recirculation system circulates air from the upper compartment through the fan rooms to the lower compartment and then from the lower compartment through the ice condenser back to the upper compartment, thus establishing a circulation pattern. Relative to sprayed and unsprayed containment regions, for the containment equalization function, the suction source for the containment air recirculation fan is the sprayed upper compartment. The discharge flow from the containment air recirculation fan is assumed to be proportionately distributed by volume between the sprayed lower compartment and unsprayed lower compartment. Similarly, the flow through the ice condenser is assumed to be proportionately distributed by volume between the sprayed upper compartment and unsprayed upper compartment. The mixing rate between sprayed and unsprayed portions of the same containment region, i.e., the upper compartment or lower compartment, is assumed to reflect natural convection and to be two turnovers of the unsprayed region per hour. In addition, since the dead-ended region is wide open to the lower compartment, the natural convection mixing rate is assumed to be two turnovers of the dead-ended region per hour and is assumed to occur only between the dead-ended region and unsprayed lower compartment.

For the purpose of evaluating the CTS, some of the containment regions are not assumed to be directly sprayed, including the ice condenser bed and inside the SG and pressurizer enclosures. Spray removal of elemental and aerosol iodine is credited using the guidance of the SRP, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System." The unsprayed volume conservatively represents 313,028 cubic feet (ft^3) of the total minimum 1,066,352 ft^3 of free volume. The compartment volumes applied by the licensee are considered best-estimate values used in the licensee's LOCA TH models, and do not necessary reflect the rounded values stated in the UFSAR. In order to account for unsprayed regions, the removal of iodine takes place only in the sprayed regions, while mass transfer of iodine from unsprayed to sprayed regions accounts for the decrease in the iodine concentration in the unsprayed volumes. The containment includes the following three sprayed regions; upper compartment, lower compartment, and fan rooms. The licensee minimized CTS volumetric flow rates for the purpose of determining iodine removal capabilities, they are 1466 gpm in the upper compartment, 660 gpm in the lower compartment, and 201 gpm in the fan rooms. These flow rates are based on an assumed single failure. The conservatively minimized sprayed containment free volumes are 621,968 ft^3 in the upper compartment, 103,770 ft^3 in the lower compartment, and 48,913 ft^3 in the fan rooms. The licensee assumed compartment volumes that are generally biased low due to the net free volumes of containment. Smaller volumes will tend to produce higher radionuclide concentrations and higher activity release rates from containment. In addition, RG 1.183, Appendix A, Section 3.3, directs calculating the natural convection mixing rate between sprayed and unsprayed regions based upon the size of the unsprayed volumes where smaller compartment volumes conservatively minimizes the containment internal mixing. However, larger volumes reduce spray effectiveness and result in lower iodine removal coefficients. For modeling purposes, the licensee applied the most limiting (conservative) volumes for each compartment to increase the estimated doses in the CR and at the EAB and LPZ. Removal of elemental iodine in each of the sprayed regions is terminated when the elemental decontamination factor in the region reaches a value of 200. Similarly, the removal rate of the aerosol iodine is reduced by a factor of 10 when the aerosol decontamination factor reaches 50. Natural deposition of only the aerosol iodine is considered in the analysis, and deposition is credited only in unsprayed regions and in sprayed regions after the CTS has been secured.

3.3.2.1 Containment Purge Pathway

The containment purge release pathway represents releases through the containment purge supply and exhaust system prior to containment isolation. Since the purge system is isolated within 15 seconds following the initiation of the event, the release is secured before the onset of the gap release at 30 seconds, which is consistent with RG 1.183, Table 4. Therefore, only those isotopes initially contained in the RCS are available for release from containment which are assumed to be instantaneously and homogeneously mixed throughout the containment atmosphere at the initiation of the event. For this pathway, the containment is modeled as a single compartment without credit for isotope removal by sprays or deposition. Radionuclides are released from containment directly to the environment without mitigation until the containment purge system is isolated. In addition, there is no radionuclide reduction by the containment purge ventilation system, which exhausts to the plant vent.

3.3.2.2 Containment Leakage Pathway

The licensee's AST LOCA analysis assumes that the radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The CNP containment building is modeled as seven separate regions. The entire containment building is served by the safety-related containment ventilation system. The containment ventilation system does not have any filtration capability; however, it does provide some additional level of mixing between the regions. Three of the regions are capable of being sprayed by the CTS. Spray induced mixing is also credited between adjacent sprayed and unsprayed regions based on two turnovers of the unsprayed volume per hour. The CTS system is assumed to be secured after 24 hours.

Iodine is released into the containment with a chemical composition of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic. The containment sump pH is maintained greater than 7.0 following the onset of CTS; therefore, re-evolution of particulate iodine into elemental iodine is not considered. Spray removal of elemental and aerosol iodine is credited and is consistent with the guidance of SRP, Section 6.5.2. Removal of elemental iodine in each of the sprayed regions is terminated when the elemental decontamination factor in the region reaches a value of 200. Similarly, the removal rate of the aerosol iodine is reduced by a factor of 10 when the aerosol decontamination factor reaches 50, consistent with RG 1.183. Natural deposition of only the aerosol iodine is considered in the analysis, and deposition is credited only in unsprayed regions and in sprayed regions after the CTS system has been secured.

Unfiltered leakage from the containment to the environment is assumed to occur uniformly from all seven regions at an initial rate of 0.18 percent/day. This leakage rate is reduced by 50 percent to 0.09 percent/day after 24 hours and is consistent with RG 1.183. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse source from the containment surface.

3.3.2.3 ESF Leakage to the Auxiliary Building Pathway

Subsequent to the emptying of the RWST during the initial phase of emergency core cooling, water from the containment sump is recirculated by the residual heat removal pumps and spray pumps, cooled via the residual heat exchangers and spray heat exchangers, and then returned to the RCS and containment. Since the containment sump water contains the radioactivity of the spilled reactor coolant, the potential off-site and CR dose due to operation of these external recirculation paths is evaluated.

Per the guidance of RG 1.183, the ESF systems that recirculate 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. This release source may also include leakage through valves isolating interfacing systems. The licensee's analysis of operation of the ECCS takes suction from the containment sump and allows system fluid to be released into the auxiliary building through pump seals, valve packing glands, and flanged connections. For this case, a portion of the core source term is deposited into liquid in the containment sump according to the release fraction and timing shown in RG 1.183, Tables 1 and 4. With the exception of noble gases, the fission products released from the fuel are assumed to instantaneously and homogeneously mix in the containment sump water. Leakage from the ECCS systems begins at the onset of switchover to recirculation, and the leak rate into the building is taken as 2 times the allowable limit established by the leak rate monitoring program. The analysis considers the equivalent of 0.2 gpm unfiltered ECCS leakage starting at the onset of the LOCA. Site administrative controls limits the leakage to 0.1 gpm. This value applies to all sources of ESF leakage into the auxiliary building, both inside and outside the ESF ventilation system envelope. This parameter is required to be doubled per the guidance in RG 1.183, Section 5.2 of Appendix A. Once the sump fluid exits the system, particulate nuclides are assumed to be retained in the liquid phase, which limits the release to iodine isotopes only. Ten percent of the iodines in the sump fluid are then assumed to become airborne and are released directly to the environment. No credit is taken for holdup or dilution in the auxiliary building, or for filtration removal by the ESF ventilation system. All releases from the auxiliary building occur from the plant vent.

3.3.2.4 ESF Leakage to the Refueling Water Storage Tank Pathway

The evaluation of ESF leakage through valves that isolate interfacing systems from the RWST is performed in a separate case. The sump activity for this pathway is identical to that applied in the case of ESF leakage to the auxiliary building. For this pathway, flow from the sump into the RWST is assumed to begin immediately upon switchover to recirculation at a rate of 1 gpm (2 times 0.5 gpm). This parameter is required to be doubled per the guidance in RG 1.183, Section 5.2 of Appendix A. These values serve as a limit on the total ESF seat leakage past isolation valves on lines which recirculate sump fluid back to the RWST. The modeling of the release of volatile iodine from the tank to the atmosphere is based upon the guidance of

NUREG/CR-5950 (Reference 32). Based upon sump pH controls, the iodine in the sump is considered to be nonvolatile. However, when introduced into the acidic solution of the RWST, a portion of the particulate iodine is converted to elemental iodine. The methodology of NUREG/CR-5950 accounts for two iodine transport/release mechanisms. The first is the fraction of the total iodine in the tank that is released in the form elemental iodine, and the second is the partitioning of elemental iodine between the liquid and vapor phases of the tank. The fraction of the total iodine that becomes elemental is a function of both the RWST pH and the total iodine concentration in the tank. The licensee's analysis determined a time-dependent RWST pH profile by modeling the sump fluid entering the tank with a constant pH of 7.0 which mixes with the remaining inventory following switchover to recirculation. Similarly, the total RWST iodine concentration that is calculated as sump iodine is transported into the tank from the sump. Calculation of the time-dependent iodine concentration in the RWST liquid for purposes of determining the elemental iodine release fraction conservatively does not model the reduction in concentration due to the release of iodine into the vapor phase. These two parameters combine to produce the elemental iodine fraction, which increases from a value of 0.0 at the beginning of the event to a maximum of 0.1914.

The ratio of the elemental iodine concentrations between the liquid and vapor phases of the tank is determined by a partition coefficient that is a function of the RWST liquid temperature. The analysis using the computer code GOTHIC calculates a bounding envelope of the high RWST temperature profile by modeling hot sump fluid entering the tank without credit for heat removal in the piping between the sump and the tank or heat losses through the tank walls. This results in a time-dependent partition coefficient that decreases from 45.41 at the beginning of the event to 31.92 after 30 days. The elemental iodine fraction and partition coefficient are applied to the leakage flow rate from the sump to the RWST to obtain an adjusted elemental iodine release rate from the tank. A similar approach is taken with the organic iodine, using a release fraction of 0.15 percent, and is consistent with RG 1.183, Position 2 of Appendix A. The release location for this pathway is from the RWST vent.

3.3.2.5 LOCA Transport Conclusion

The NRC staff reviewed the licensee's analyses of the containment purge pathway, containment leakage pathway, ESF leakage to the auxiliary building, and ESF leakage to the RWST. The NRC staff finds that the analyses were performed in accordance with RG 1.183, using NRC-approved methods, and are therefore acceptable.

3.3.3 Post-LOCA Containment Sump pH

According to NUREG-1465 (Reference 31), iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide, which is a highly ionized salt soluble in water. Iodine in this form does not present any radiological problems since it remains dissolved in the sump water and does not enter the containment atmosphere. However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can therefore be released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is very important. Maintaining a basic pH in the containment sump water will ensure that this conversion will be minimized. The pH of the sump water at CNP is controlled by addition of sodium hydroxide (NaOH) from the spray additive tank (SAT), and sodium tetraborate (NaTB) from the ice condenser, to the boric acid

(H_3BO_3) dissolved in the sump water after a LOCA. After a LOCA, several acids are generated in the containment. Relative amounts of these acids, and those of NaOH and NaTB, determine the pH reached by the containment sump water. The licensee performed a plant-specific containment sump pH evaluation using a system of equations to determine the effect of the post-LOCA acid generation. The calculations included assumptions for temperature, boron concentration, sodium concentration, and strong acid generation. The licensee indicated that the current limiting case for the lowest sump pH occurs with a large break LOCA with one eductor operating at the surveillance test minimum flow criteria. The calculation takes into account H_3BO_3 from the RCS, safety injection accumulators, and RWST; NaOH from the SAT; and NaTB from the ice condenser. The CNP TS SAT minimum volume for NaOH solution is 4,000 gallons. The licensee conservatively assumed the end of melt-time of the ice in the ice condenser to occur at 30 days. The boron and sodium concentrations in the analysis were derived from the H_3BO_3, NaOH, and NaTB concentrations. The parameters the licensee used in the containment sump pH analysis are found in Table 1.

Table 1. Design Parameters Used in Minimum Containment Sump pH Analyses

Component	**Volume (gallons)**	**Concentration**
RWST	375,500	2600 parts per million (ppm) H_3BO_3
RCS	72,904	2600 ppm H_3BO_3
Safety Injection Accumulators	29,054	2600 ppm H_3BO_3
SAT	4,000	30 weight percent NaOH
Ice Condenser	287,582	1800 ppm NaTB

Based on the information provided in Table 1, the NRC staff calculated the mass quantities provided in Table 2.

Table 2. Mass Quantities Based on Design Parameters

Component	**Compound**	**Pounds-mass (lb_m)**
RWST	H_3BO_3	8,148
RCS	H_3BO_3	1,582
SI Accumulators	H_3BO_3	630

The licensee stated that the maximum containment sump volume is 50,955 ft^3 or $1.44x10^{06}$ liters (L). Using the maximum sump volume and the information provided in Tables 1 and 2, the NRC staff determined that the total mass for H_3BO_3 to be approximately 10,360 pound-mass (lb_m) or $5.28x10^{-02}$ gram-moles per liter (g-mol/L).

In addition, the licensee considered the effects of strong acid generation on the post-LOCA containment sump pH. Per guidance NUREG/CR-5950 (Reference 32), the licensee calculated the mass of nitric acid (HNO_3) generated by the radiolysis of air and water inside containment and the mass of hydrochloric acid (HCl) generated by the radiolysis of Hypalon® electrical cable insulation inside containment.

HCl is formed from decomposition of chlorinated polymer cable insulation by radiation. The licensee estimated the cable insulation in containment to weigh approximately 100,000 lb_m. Based on this weight and in accordance with NUREG/CR-5950, the amount of HCl produced by the irradiation of electrical cable is estimated as $4.6x10^{-04}$ gram-moles of HCl per pound-mass of insulation per Megarad (g-mol/Mrad-lbm). The NRC staff determined that the total amount of HCl produced per Mrad, based on the mass of cable insulation determined by the licensee, is 4.60×10^{01} g-mol/Mrad. In accordance with guidance identified in NUREG/CR-5175 (Reference 54), the NRC staff obtained an estimated airborne post-LOCA containment radiation dose. Radiation is commonly used for evaluating EQ of electrical equipment in PWR containments. Utilizing the plant-specific information and applying the values derived from the stated NUREG/CRs, the staff determined a total amount of HCl produced, which confirms the values determined by the licensee. The amount of HNO_3 produced is proportional to the time-integrated dose rate for gamma and beta radiation. The licensee provided the concentrations of HCl and HNO_3 at various times of interest, as shown in Table 3.

Table 3. Time-Dependent Hydrochloric Acid and Nitric Acid Concentrations (g-mol/L)

End of Injection		**End of SAT Eduction**		**8 Hours**		**30 Days**	
HCl	HNO_3	HCl	HNO_3	HCl	HNO_3	HCl	HNO_3
1.53×10^{-04}	4.50×10^{-06}	1.47×10^{-04}	4.50×10^{-06}	8.14×10^{-04}	3.45×10^{-05}	3.00×10^{-03}	2.29×10^{-04}

The licensee determined that the amounts of strong acids generated in the containment post-LOCA for 30 days decreases the sump water pH by less than 0.1 pH unit. The amount of sump water pH decrease due to strong acid generation at CNP is consistent with what other similarly operating power plants have determined.

In order to neutralize the H_3BO_3, HCl, and HNO_3, the licensee chose to buffer the sump pool water by using NaOH and NaTB buffers. Such buffering action is intended to maintain basic pH in the sump pool despite the presence of the acids. The licensee has calculated that by adding approximately 4,000 gallons of 30 weight percent NaOH from the SAT and NaTB from the ice condenser, the pH in the sump water will remain basic for 30 days.

Per the analysis, the licensee determined the containment sump pH under post-accident conditions to be 7.23 at the end of the injection phase, and 7.87, at 30 days, using the more limiting system alignment. The licensee-provided pH values at various times of interest are shown in Table 4.

Table 4.
Limiting Containment Sump pH Results (Large Break LOCA with Eductor at Minimum Flow)

End of Injection	**End of SAT Eduction**	**8 Hours**	**30 Days**
7.23	7.26	7.33	7.87

The NRC staff has reviewed the information the licensee provided and finds that by using NaOH and NaTB as the buffers in the quantity specified, the pH of the containment sump will remain above 7.0 for 30 days post-LOCA.

3.3.3.1 Post-LOCA Containment Sump pH Conclusion

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of containment sump water and the corresponding fraction of the dissolved iodine in the sump water that is converted into the elemental form. The methodology relies on using buffering actions of NaOH and NaTB. The assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment sump pH. The calculations were made for the 30-day period following a LOCA. The NRC staff verified that the post-accident containment sump pH will be maintained above 7.0 for 30 days following a LOCA. Since a basic pH will be maintained in the containment sump water, the conversion of iodine to the elemental form will be minimized.

3.3.4 Control Room (CR) Habitability

Regulatory position 4.2 of RG 1.183 states that the TEDE analysis should consider all sources of radiation that will cause exposure to the CR personnel. The licensee's modeling of the CR habitability included sources from:

- Contamination of the CR atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility;
- Radiation shine from the external radioactive plume released from the facility;
- Radiation shine from radioactive material in the reactor containment; and,
- Radiation shine from radioactive material in systems and components inside or external to the CR envelope, e.g., radioactive material buildup in recirculation filters.

3.3.4.1 CR Modeling

CNP has a separate CR for each unit. Each CR has its own redundant ventilation system. Each CR has two outside air intakes. One outside air intake at each unit supplies both of the air handlers for that unit. The other outside air intake at each unit is used during emergency conditions for pressurization. Motor-operated dampers isolate all four outside air intakes. The CNP CRs operate in the CR envelope within the filtered pressurization mode. Since the operation of the CRs is identical, and since the analyses performed in support of this amendment were based on the limiting parameters for either unit, radiological consequences only need to be assessed for one CR.

During normal operation the fresh-air intake provides 880 cubic feet per minute (cfm) of unfiltered air to the each CR envelope through the normal outside air intake. Also during this period, it is assumed that 40 cfm of unfiltered inleakage enters each CR. This value was determined by tracer gas leakage testing. The 100 percent flow pressurization fans are fitted with two motor-operated isolation dampers installed in parallel, and have a dedicated emergency power bus and air intake. The pressurization/cleanup air fans do not normally operate.

Following a safety injection signal, the CR ventilation system is automatically placed into recirculation after applicable delays for signal processing, emergency power restoration, and damper repositioning. In this configuration, the CR pressurization/cleanup fans circulate 5400 cfm of air through the CR filters, with 880 cfm of this flow supplied by fresh air from the emergency outdoor air intake and the remaining 4520 cfm taken from the CR envelope. At this flow rate, the licensee assumes the filter efficiency to be 94.05 percent for elemental and organic species, 98.01 percent for particulates. The licensee assumes a delay of 70 seconds from the time of safety injection signal actuation, which varies from accident to accident. When the CR ventilation system is aligned in the pressurization/cleanup mode, the CR envelope is at a positive pressure with respect to the surrounding areas and leakage is predominantly out of the CR. However, this flow configuration creates a negative pressure in the system ducting downstream of the isolated normal intake dampers. Therefore, the CR unfiltered inleakage is assumed to enter the CR at the location of the normal intakes. Also during this period, it is assumed that 40 cfm of unfiltered inleakage enters the CR.

Regulatory Position 4.2.4 of RG 1.183 states that credit for ESFs that mitigate airborne radioactive material within the CR may be assumed. Regulatory Position 5.1.2 of RG 1.183 states that credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operation procedures. Section 3 of Regulatory Issue Summary (RIS) 2006-04 (Reference 55), elaborates that some licensees have proposed that certain ESF ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal non-ESFs ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. The licensee did not credit non-ESF ventilation systems. The NRC staff finds that the licensee's assumption not to credit non-ESFs is consistent with Regulatory Position 4.2.4 and 5.1.2, and is therefore acceptable.

3.3.4.2 CR Direct Shine Dose

CR operators, as well as other plant personnel, are protected from radiation sources associated with normal plant operation by a combination of shielding and distance. To a large extent, the same radiation shielding also provides protection from DBA radiation sources. Regulatory Position 4.2.1 in RG 1.183 states that the TEDE analysis should consider all sources of radiation that will cause exposure to CR personnel. This includes direct shine contributions from the external radioactive plume released from the facility, radiation shine from radioactive material in the reactor containment, radiation shine from radioactive material in systems, and components inside or external to the CR envelope, e.g., radioactive material buildup in recirculation filters. Consistent with RG 1.183, Regulatory Position 4.2.1, the licensee considered the following DBA LOCA direct shine dose contributors in the CR habitability envelope analysis:

- CR filters shine;
- Airborne cloud shine external to the CR; and,
- Containment shine to the CR.

The filter shine dose is calculated by first determining the maximum activity loading on the CR ventilation system filters during the LOCA event. This is done by considering the CR ventilation maximum fan capacity flow rate along with filter efficiencies of 100 percent. The activities from the recirculation filter edit of the RADTRAD output files are used as the input for a radiation shielding calculation using the computer code MicroShield Version 8.03, which models the geometry and composition of the CR filter housing and the recirculation air handler unit position with respect to the CR. Credit is taken for shielding by structural materials and attenuation in air. An integrated 30-day dose is calculated for CR personnel.

The CR dose due to direct radiation streaming through the equipment hatch following a LOCA assumes that the CR receptor is positioned directly in front, such that the exposure from containment is due to a direct line-of-sight through the entire area of the hatch. The result is a

30-day dose to an individual in the CR due to direct shine through the containment equipment hatch.

The dose contribution from the external cloud is assessed qualitatively using the guidelines of NUREG-0800, Section 6.4, which states that 18 inches of concrete is generally adequate to attenuate the external DBA radiation to negligible levels. The licensee reviewed site drawings which described the minimum thickness of the CR walls to be 18 inches, which is subsequently surrounded auxiliary building and turbine building walls. Similarly, the thickness of the concrete ceiling immediately above the CR is 18 inches, and there is an additional concrete roof above the CR. As such, the shielding against external radiation sources well exceeds the amount identified as adequate in the guidance, and is considered acceptable to the NRC staff.

The total LOCA shine dose is presented in Table 5. The filter shine dose following the LOCA event is applied to all other events and is considered conservative since the LOCA event results in the highest activity source term release to the environment.

Table 5: LOCA Direct Shine Dose

Source	Dose roentgen equivalent man (rem)
Containment	0.246
Control Room Filters	0.139
External Cloud	Negligible
Total	0.385

3.3.4.3 CR Emergency Ventilation and Containment Leakage Rate

There are two changes related to this LAR that would affect emergency ventilation and containment leakage. The first change is to TS Section 5.5.9.c. This changes the CR emergency ventilation charcoal adsorber maximum allowable methyl iodide penetration from 1 percent to 2.5 percent. The dose analyses prompted the change of the penetration value. The change is acceptable because using a filter efficiency of 95 percent, there remains a factor of safety of 2 with this change (2.5 percent times 2). The change is also acceptable because it follows the recommendations in RG 1.52. RG 1.52 lists the methyl iodide penetration test acceptance criterion as less than or equal to 2.5 percent to ensure a filter efficiency of 95 percent. The NRC staff finds that this change is in accordance with the applicable regulatory requirements.

The second change is a change to TS Section 5.5.14.c. The change reduces the maximum allowable leakage rate specified by the containment leakage rate program from 0.25 percent per day to 0.18 percent per day. The maximum allowable containment leakage value is requested to be lowered to provide analysis margin to the dose limits. As this is a more restrictive value, the NRC staff finds this TS change to be acceptable because the maximum allowable leakage rate will continue to provide reasonable assurance that the activities authorized by the amended technical specification can be conducted without endangering health and safety.

Based on the regulatory and technical evaluations above, the NRC staff finds that the licensee has adequately addressed the request to change the CR emergency ventilation charcoal adsorber maximum allowable methyl iodide penetration and the maximum allowable leakage rate specified by the containment leakage rate testing program. The staff finds that licensee has adequately addressed the impact of this change.

The proposed change is acceptable because the change to the maximum allowable leakage rate is in a conservative direction. There have been no additional changes to the leak testing program. The licensee is in accordance with the recommendations of RG 1.52 with regard to the methyl iodide penetration acceptance criterion. Therefore, the proposed license amendment is acceptable with respect to CR emergency ventilation and containment leakage rate.

3.3.4.4 CR Habitability Conclusion

The NRC staff reviewed the licensee's assessment of CR habitability, including CR modeling, direct shine dose, and the CR emergency ventilation and containment leakage rate. The NRC staff finds that the assessment was performed in accordance with RG 1.183, considering all sources of radiation that would cause exposure to CR personnel, and is therefore acceptable.

3.3.5 LOCA Conclusion

The licensee's evaluation of a postulated LOCA, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of the SRP, PSDC 11, and Section 6.4 of the SRP, "Control Room Habitability Systems." The NRC staff verified that the source terms and transport pathways were assessed in accordance with approved guidance, and that the post-accident containment sump pH will be maintained above 7 for 30 days following a LOCA. Therefore, The NRC staff found that in the event of a LOCA, the EAB, LPZ, and CR doses would meet the applicable criteria and therefore are acceptable.

3.4 Fuel Handling Accident (FHA)

3.4.1 FHA Analysis Summary

The FHA analysis postulates that a spent fuel assembly, or a bundle of fuel assemblies, is damaged during fuel handling. The CNP FHA accident is described in UFSAR, Section 14.2.1, "Radiological Consequences of Fuel Handling Accident." The analysis considers both a drop in the containment building without established containment integrity, and a drop in the auxiliary building with the fuel handling area exhaust ventilation (FHAEV) in service. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad during normal power operations. All of the fuel rods in the dropped assembly are conservatively assumed to experience fuel cladding damage, releasing the radionuclides within the fuel rod gap to the fuel pool or reactor cavity water. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool, depending on their physical and chemical form. The remaining activity not decontaminated by passage through the overlaying water is assumed to be released directly to the outside atmosphere at a constant rate over a 2-hour period.

3.4.2 FHA Source Term

As discussed in Section 3.2.3 of this SE, "Non-LOCA Gap Release Source Term," the FHA source term is developed from the core source term and follows the guidance of RG 1.183, Position 3.1, which states that "the fission product inventory of each damaged fuel rod for DBA events that do not involve the entire core is determined by dividing the total core inventory by the number of rods in the core." Therefore, the affected assemblies are assumed to be those with the highest inventory of fission products of the 193 assemblies in the core. Consistent with RG 1.183, Position 3.2, to account for differences in power level across the core, a radial peaking factor of 1.65 is applied to the inventory of the damaged rods. The analysis of the FHA in both locations is modeled as a fuel assembly drop which occurs 120 hours after reactor shutdown. This is consistent with the guidance of RG 1.183, Regulatory Position 3.1. As discussed in RG 1.183, Regulatory Position 2, of Appendix B, if the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively. This gives an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85 percent) and organic iodine (0.15 percent) species results in the iodine above the water being composed of 57 percent elemental and 43 percent organic species. As further discussed in RIS 2006-04, an overall decontamination factor of 200 is achieved when the decontamination factor for elemental iodine is 285, not 500. Therefore, the licensee credited a decontamination factor of 285 for elemental iodine and 1.0 for organic iodines for both FHA scenarios. The pool water is assumed to retain 100 percent of the alkali metals. It is assumed by the licensee that no more than 150 fuel rods per assembly could exceed the new non-LOCA maximum LHGR of 6.3 kW/ft for burnups exceeding 54 GWD/MTU. Based on the licensee's non-LOCA gap release analysis, the gap release fractions found in RG 1.183, Table 3, for rods that exceed the 6.3 kW/ft LHGR limit above 54 GWD/MTU are doubled.

3.4.3 FHA Transport within the Containment

For the FHA occurring inside containment, the licensee assumed that the containment integrity is not established. The release location in containment for the CR dose is assumed to be a point on the external containment surface closest to the CR intakes. The CR is assumed to be manually placed into the pressurization mode 20 minutes after the start of the event by plant operators. Consistent with the guidance in RG 1.183, the licensee assumed that the release of all fission products to the environment from an FHA occurring within the containment occurs over a 2-hour period. Consistent with the guidance in RG 1.183, the licensee did not credit holdup or dilution of the released activity within the containment. The NRC staff finds that the licensee's assumptions regarding the release of fission products from an FHA within the containment are consistent with the guidance in RG 1.183 and are therefore acceptable.

3.4.4 FHA Transport within the Auxiliary Building

The licensee evaluated the FHA within the auxiliary building taking credit for filtration of the FHAEV being in service. The FHAEV system discharges to the plant vent. Consistent with the guidance in RG 1.183, the licensee assumed that the release of all fission products to the environment from an FHA occurring within the auxiliary building occurs over a 2-hour period. Consistent with the guidance in RG 1.183, the licensee did not credit holdup or dilution of the released activity within the auxiliary building. The NRC staff finds that the licensee's assumptions regarding the release of fission products from an FHA within the auxiliary building are consistent with the guidance in RG 1.183, and are therefore acceptable.

3.4.5 FHA Conclusion

The licensee's evaluation of the two postulated FHAs, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of NUREG-0800, PSDC 11, and Section 6.4 of the SRP. The NRC staff found that in the event of an FHA, the EAB, LPZ, and CR doses would meet the applicable criteria, and are therefore acceptable.

3.5 Main Steam Line Break (MSLB) Accident

3.5.1 MSLB Accident Analysis Summary

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in the nuclear system process barrier. The CNP MSLB accident is described in the UFSAR, Section 14.2.5, "Rupture of a Steam Pipe." CNP is designed to immediately detect such an occurrence, initiate isolation of the broken line and actuate the necessary protective features.

In order to evaluate the possible effects of this event, the licensee postulated a complete circumferential break on one of the main steam lines at a location outside of containment, resulting in the release of steam from the affected steam line. The faulted SG would rapidly depressurize, and release its entire liquid inventory and dissolved radioiodines through the faulted steam line to the environment. The rapid secondary depressurization would cause a reactor power transient, resulting in a reactor trip. Since a loss of offsite power is assumed to occur with the reactor trip, the main condenser would not be available as a heat sink. The unaffected SGs would be used to cool down the plant by dumping steam to the environment. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The radiological consequences of a break outside containment will bound those results from a break inside containment. Thus, only the break outside containment is considered with regard to CR dose.

3.5.2 MSLB Accident Source Term

An MSLB would not result in fuel damage. Appendix E of RG 1.183, Regulatory Position 2, states the following regarding MSLB accident analyses:

> If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.

Thus, consistent with RG 1.183, the licensee evaluated the MSLB based on the two iodine spiking cases. The first case, the "pre-accident iodine spike case," assumes that a reactor transient has occurred prior to the postulated MSLB in which the primary coolant iodine concentration has increased to the maximum TS value of 60 µCi/gm. The second case, the

2 The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

"concurrent iodine spike case," a spiking model is used that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value of 1.0 μCi/gm specified in TSs. This concurrent iodine spike is assumed to have a duration of 8 hours.

The chemical form of radioiodine released from the SGs to the environment is assumed to be 97 percent elemental and 3 percent organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking. In both cases, the remaining nuclides in the RCS are also available for release from the fuel and are assumed to be released instantaneously and homogenously through the primary coolant. Prior to the accident, the secondary coolant activities and the primary-to-secondary leakage rate is modeled to correspond with the proposed TS 5.5.7.b.2 program limit of 0.25 gpm per SG.

The NRC staff finds the licensee's assumptions regarding iodine spiking and chemical form to be consistent with Appendix E of RG 1.183, since they maximize the radiological consequences of the event.

3.5.3 MSLB Accident Transport

The licensee analyzed the MSLB by assuming that the leakage from the RCS into all of the SGs, and the steam release from the intact SGs, continues until the RCS is cooled to 212 degree Fahrenheit (° F) after 24 hours. All of the noble gases and all of the nuclides which leak into the faulted SG are released directly to the environment without mitigation. During this period, leakage into the intact SGs mixes with the bulk fluid where a portion of the activity is released based upon the steaming rate and a partition coefficient. A partition coefficient of 100 is assumed for the iodine nuclides, and retention of particulate radionuclides in the SGs is limited by moisture carryover from the SGs.

Leakage that immediately flashes to vapor will rise through the bulk water of the SG and enter the steam space. The licensee recognized that early in the transient, the water level in the intact SG secondary may be below the top of the tube bundle, and the bulk water partitioning may not apply. In this case, a flashing fraction is calculated based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary-to-secondary leakage which flashes to vapor is assumed to be released directly to the environment without mixing. The iodine and particulate partition coefficients are applied to the unflashed portion. The tube bundles in the intact SGs are assumed to be fully covered after 40 minutes.

The release locations from the faulted SG are selected to maximize the CR and offsite doses without regard to the location of the break with respect to the main steam isolation valves and without credit for the main steam isolation valve closure. Releases from the intact SGs occur from the pressure operated relief valves (PORVs) and Main Steam Safety Valves (MSSVs). The CR is automatically realigned into the pressurization mode upon receipt of a safety injection signal.

3.5.4 MSLB Accident Conclusion

The licensee's evaluation of a postulated MSLB, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of NUREG-0800, PSDC 11, and Section 6.4 of the SRP. The NRC staff found that in the event of a MSLB, the EAB, LPZ, and CR doses would meet the applicable criteria, and are therefore acceptable.

3.6 <u>Steam Generator Tube Rupture (SGTR) Accident</u>

3.6.1 SGTR Analysis Summary

The CNP SGTR accident is described in the UFSAR Section 14.2.4, "Steam Generator Tube Rupture." This DBA postulates a rupture in a tube in one of the four SGs resulting in the transfer of reactor coolant water to the ruptured SG. The primary-to-secondary flow through the ruptured tube ("break flow") following a SGTR results in a depressurization of the RCS, a reactor trip, and actuation of safety injection. Since a loss of offsite power is assumed to occur when the reactor trips, the main condenser is not available as a heat sink, and contaminated steam is released to the environment through the SG power operated relief valves (and safety valves if their set-point is reached). After safety injection actuates, it is assumed that the RCS pressure will stabilize at a value at which the safety injection and break flows are equal. The break flow is assumed to continue at this equilibrium value until plant operators have taken action to reduce RCS pressure. When RCS pressure is less than the SG pressure, the pressure differential and the flow reverses direction, terminating the break flow. The licensee assumes that this occurs within 30 minutes from safety injection actuation. The unaffected SGs are used to cool down the plant by dumping steam to the environment. The released steam maybe contaminated due to leakage of reactor coolant into the SGs.

3.6.2 SGTR Accident Source Term

This event would not result in fuel damage. Therefore, prior to the accident, the primary and secondary coolant activities and the primary-to-secondary leakage is modeled corresponding to the limits defined in TSs. Appendix G of RG 1.183, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the TSs, where two cases of iodine spiking should be assumed. Therefore, consistent with RG 1.183, the licensee evaluated the SGTR based on the two iodine spiking cases. The first case, the "pre-accident iodine spike case," assumes that a reactor transient has occurred prior to the postulated SGTR in which the primary coolant iodine concentration has increased to the maximum TS value of 60 µCi/gm. In the second case, the "concurrent iodine spike case," a spiking model is used that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value of 1.0 µCi/gm specified in TSs. This concurrent iodine spike is assumed to have a duration of 8 hours. The chemical form of radioiodine released from the SGs to the environment is assumed to be 97 percent elemental and 3 percent organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking. In both cases, the remaining nuclides in the RCS are also available for release from the fuel and are assumed to be released instantaneously and homogenously through the primary coolant. Prior to the accident, the secondary coolant activities and the primary-to-secondary leakage rate is modeled to correspond with the proposed TS 5.5.7.b.2 program limit of 0.25 gpm per SG.

The NRC staff finds that the licensee's assumptions regarding iodine spiking and chemical form are consistent with Appendix E of RG 1.183, since they maximize the radiological consequences of the event.

3.6.3 SGTR Transport

Prior to the reactor trip, the activity is assumed to be released from the steam jet air ejector in the turbine building. Following the reactor trip, the release location shifts to the PORVs and MSSVs. The CR is automatically realigned into the pressurization mode upon receipt of a safety injection signal.

3.6.4 SGTR Accident Conclusion

The licensee's evaluation of a postulated SGTR, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of NUREG-0800, PSDC 11, and Section 6.4 of the SRP. The NRC staff found that in the event of a SGTR, the EAB, LPZ, and CR doses would meet the applicable criteria and are therefore acceptable.

3.7 Reactor Coolant Pump (RCP) Locked-Rotor Accident (LRA)

3.7.1 RCP LRA Analysis Summary

The CNP LRA is described in the UFSAR, Section 14.1.6.4, "Locked Rotor Accident." For this DBA, a reactor coolant pump rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage.

3.7.2 RCP LRA Source Term

The LRA dose analysis is defined by the 11 percent of the fuel rods which become damaged by the event. Radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the primary coolant. Noble gases are released directly to the environment, and the remaining isotopes are transported to the SGs at a rate of 1 gpm. The core source term described in Section 3.2.1 of this SE is applicable to this event, and the fraction of these activities available for release into the coolant are based upon the gap inventory fractions shown in Table 3 of RG 1.183 for non-LOCA gap inventory fractions and the assembly radial peaking factor of 1.65. To account for fuel rods contained in two of the fuel assemblies which exceed the burnup limits of Footnote 11 of Table 3 of RG 1.183, the gap inventory of all of the rods in these two assemblies are assumed to be doubled. As such, with 193 fuel assemblies in the core, the effective core-wide multiplier on the gap inventory fractions is 1 + (2/193) = 1.0104. Since the fuel failure fraction is applied to the entire core source term, 11 percent of the rods in both the standard and high burnup assemblies are assumed to fail, and the assembly peaking factor is conservatively applied to all of the failed rods in the core, the analysis is considered acceptable by the NRC staff.

3.7.3 RCP LRA Transport

The radiological consequences are due to leakage of the contaminated reactor coolant to the SG and from there, the environment. The primary-to-secondary leak rate in the SGs is 1 gpm to all SGs. The NRC staff finds that the licensee's assumption regarding the primary-to-secondary leak rate is consistent with the leakage performance criteria of the SG program described in the TSs and is therefore acceptable. A loss of offsite power is conservatively assumed to occur when the reactor trips, rendering the main condenser unavailable for steam dump. With the main condenser unavailable, the plant is cooled down by steam released to the environment. Eight hours after the accident, the residual heat removal system is assumed to start operating to cool down the plant, and steam and activity are no longer assumed to be released to the environment.

During the first 40 minutes of the event, the water level on the secondary side of the SGs is assumed to be below the top of the tube bundles. During this time, a portion of the primary-to-secondary leakage flashes to vapor based upon the thermodynamic conditions of the reactor and secondary coolant. Nuclides contained in the flashed tube leakage are released to the environment without mitigation. The unflashed leakage mixes with the bulk water in the SGs and is released as a function of the steaming rate and the partition coefficients. After 40 minutes, all of the leakage is treated as unflashed, which continues until 24 hours when the RCS temperature is cooled to 212 ° F. The NRC staff finds that the licensee's assumptions regarding the duration of the primary-to-secondary leakage are consistent with Appendix E of RG 1.183, and are therefore acceptable.

Since the quantity of the fission products released from the failed fuel dominates the RCS activity during the event, the initial nuclide concentration in the RCS prior to the event is not considered. However, the analysis does include the dose contribution from the release of iodine initially present in the SG secondary side. All releases occur from the PORVs and MSSVs, which are located in the main steam enclosures. For this event, the CR ventilation system remains in the normal alignment without filtration or recirculation until manually placed into the pressurization mode after 20 minutes.

3.7.4 RCP LRA Conclusion

The licensee's evaluation of a postulated reactor coolant pump LRA, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of NUREG-0800, PSDC 11, and Section 6.4 of the SRP. The NRC staff finds that in the event of a reactor coolant pump LRA, the EAB, LPZ, and CR doses would meet the applicable criteria and are therefore acceptable.

3.8 Control Rod Ejection (CRE) Accident

3.8.1 CRE Accident Analysis Summary

The CNP CRE accident is described in the UFSAR Section 14.2.6, "Rupture of Control Rod Drive Mechanism Housing." This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. This short positive reactivity insertion, together with an adverse core power distribution, produces a rapid core power level increase which results in fuel rod damage and localized melting. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high neutron flux signals. This failure breeches the reactor pressure vessel head resulting in a minor LOCA to the containment.

3.8.2 CRE Accident Source Term

Consistent with RG 1.183, Appendix H, Regulatory Position 1, the CRE accident analysis assumes that 10 percent of the fuel rods in the core experience a departure from nucleate boiling and cladding failure to such an extent that the entire fission product inventory in the cladding gaps of these rods is released. The gap activity consists of 10 percent of the core inventory of noble gases, iodine, and alkali metals. In addition, 50 percent of the fuel rods experiencing a departure from nucleate boiling are conservatively assumed to experience fuel melting. Of the fuel rods experiencing fuel melting, melting is assumed to occur over 50 percent of their axial length and 10 percent of their radial volume. Therefore, the fraction of fuel melting is 0.25 percent of the core. In addition, the gap inventory fractions are increased by a factor of 1.0104 to account for high burnup fuel.

3.8.3 CRE Accident Transport

Two fission product release pathways to the environment are analyzed independently; a containment leakage pathway and a secondary system pathway. For the containment leakage pathway, no credit is taken for the CTS. For the secondary system pathway, main steam condensers are assumed to be unavailable for steam dump, since offsite power is assumed to be lost. Eight hours after the accident, the residual heat removal system is assumed to start operating to cool down the plant, and steam and activity are no longer assumed to be released to the environment.

For the containment leakage pathway, of the fuel rods experiencing fuel melting, 100 percent of the noble gases and 25 percent of the iodines in the melted fuel are assumed to be available for release. The fission product inventory in the gap of the damaged fuel rods and available for release from the melted fuel is released throughout the containment atmosphere. For the secondary system pathway, of the fuel rods experiencing fuel melting, 100 percent of the noble gases and 50 percent of the iodines in the melted fuel are assumed to be available for release. The fission product inventory in the gap of the damaged fuel rods and available for release from the melted fuel is dissolved in the reactor coolant and available for release from the secondary system due to primary-to secondary leakage. Both release pathways assume 100 percent of the activity released from the fuel is released instantaneously and homogeneously through the containment atmosphere (first pathway) or completely dissolved in the primary coolant and available for release to the secondary system (second pathway).

For the release from containment, no credit is taken for removal by CTS or for deposition of elemental iodine on containment surfaces. Natural deposition of aerosols in containment is assumed to occur beginning 24 hours after the start of the event. Activity is released from containment at the proposed TS leak rate. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse release from the containment surface. For conservatism, the release of iodine initially present in the SG secondary side is also considered to address any supplemental cooldown by the SGs for this event. Consistent with RG 1.183, Appendix H, Regulatory Position 4, the chemical form of radioiodine released to the containment atmosphere is assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide, and is therefore considered acceptable by the NRC staff.

For the release from the secondary system, the noble gases are assumed to be released directly to the environment, and the remaining fission products are transported to the SGs at the TS SG program leakage limit of 1 gpm. At the beginning of the event, a portion of the primary-to-secondary leakage is assumed to flash to vapor based upon the thermodynamic conditions of the reactor and secondary coolant, and the flashed leakage is released directly to the environment without mitigation. The unflashed portion of the tube leakage mixes with the bulk fluid in the SG secondary and becomes vapor at a rate that is a function of the steaming rate and the partition coefficients. After 40 minutes, the water level in the SG is assumed to fully cover the tube bundles, and all of the primary-to-secondary leakage is treated as unflashed. The leakage continues until steam releases are terminated when the RCS temperature is cooled to 212 °F at 24 hours. With the large amount of fission products introduced into the reactor coolant by failed fuel, the initial activity of the RCS prior to the event is not considered. However, the dose contribution from the iodine activity initially present in the SG secondary is included in the analysis. All releases from the secondary system occur from the PORVs and MSSVs. Consistent with RG 1.183, Appendix H, Regulatory Position 5, iodine releases from the SGs to the environment is assumed to be 97 percent elemental and 3 percent organic, and is therefore considered acceptable by the NRC staff.

The CR ventilation system is automatically realigned into the pressurization mode following receipt of a safety injection signal.

3.8.4 CRE Accident Conclusion

The licensee's evaluation of a postulated CRE accident, using the AST, concluded that the radiological consequences at the EAB, LPZ, and CR are within the accident dose criteria stated in 10 CFR 50.67, as well as accident specific criteria in Chapter 15 of NUREG-0800, PSDC 11, and Section 6.4 of the SRP. The NRC staff found that in the event of a CRE accident, the EAB, LPZ, and CR doses would meet the applicable criteria and are therefore acceptable.

3.9 Waste Gas Decay Tank (WGDT) and Volume Control Tank (VCT) Rupture

The CNP WGDT and VCT rupture analyses are described in the UFSAR, Section 14.2.3, "Accidental Waste Gas Release," and UFSAR Section 14.1.5, "Chemical and Volume Control System Malfunction," respectively.

RG 1.183 does not provide guidance relative to either the WGDT or VCT rupture analyses. Guidelines for the WGDT and VCT rupture analyses are given in Branch Technical Position (BTP) 11-5 of the SRP, with additional instruction available from RIS 2006-04. As discussed in RIS 2006-04, as part of a full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For these accidents, they have proposed acceptance criteria of 500 millirem (mrem) TEDE. The acceptance criteria for these events are that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised Part 20 of 10 CFR to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses full implementation of an AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. However, licensees may also choose not to implement AST for this accident-type and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

By letter dated November 14, 2002 (Reference 47), the NRC approved a selective implementation of the AST for CNP through the issuance of Amendment Nos. 271 and 252 to CNP, Units 1 and 2. These amendments approved the use of an AST at CNP for CR habitability for the DBA WGDT and VCT analyses. Since the source term is being updated to reflect the use of high-burnup fuel in future core designs, the licensee revised the dose analysis for the CR. In addition, the licensee chose not to implement the AST for these accidents and will continue with the existing analysis and the acceptance criteria of 500 mrem whole body. Pursuant to the guidance of RIS 2006-04, the NRC staff finds this acceptable.

3.9.1 WGDT and VCT Rupture Analysis Summary

During normal operation of a nuclear power plant, the reactor generates radioactive fission and activation gases resulting from the radiolytic decomposition of water and escapement of fission products from defects in and contamination on the zirconium fuel cladding. These gases are continuously removed from the RCS. The processing of the reactor coolant water (known as letdown) by auxiliary systems results in the accumulation of radioactive gases in gas decay tanks and the VCT. The WGDT is used to store processed radioactive gases to allow for radioactive decay before controlled releases to the environment. The VCT tank is a component in the plants' chemical and volume control systems that serves as a surge volume to balance differences in processed letdown and makeup flow rates while maintaining reactor coolant inventory. As purified letdown water is sprayed into the VCT, which is normally less than 25 percent full, radioactive gases collect in the top of the VCT. Both of these tanks are located in the lower elevations of the auxiliary building and are seismically designed to withstand a design basis seismic event without failure. For the purposes of analysis, it is assumed that either tank ruptures by an unspecified mechanism after the reactor has been operating for one core cycle.

3.9.2 WGDT and VCT Rupture Source Terms

BTP 11-5 of NUREG-0800, states that the radiological consequences of a single failure of an active component in the waste gas system should use a system design-basis source term for light-water-cooled nuclear power plants. The NRC staff method of calculation for this analysis is based on conservative assumptions to maximize the design capacity source term. Only the radioactive noble gases (xenon and krypton) are considered since the assumed transit time is long enough to permit major radioactive decay of oxygen and nitrogen isotopes. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. Therefore, consistent with BTP 11-5, the licensee evaluated the WGDT source term to be equal to the noble gas content of the RCS during normal operation with 1.0 percent failed fuel. The total activity released is determined to be equal to 59,256.4 curies dose equivalent Xe-133, which exceeds the single WGDT licensing limit of 43,800 curies. For the VCT rupture case, a failure occurs just prior to venting, which releases the accumulated noble gases in the liquid and vapor phases of the tank. In addition, the noble gases within the fluid of the letdown line entering the tank continue to be released for an additional 15 minutes following the tank rupture. The tank and letdown line activities are calculated from RCS equilibrium noble gas concentrations based upon 1 percent failed fuel.

3.9.3 WGDT and VCT Rupture Transport

BTP 11-5 considers the release to the environment occur through a pathway not normally used for planned releases and will require a reasonable time to detect and take remedial action to terminate the release. Consistent with BTP 11-5, the licensee assumed that normal auxiliary building ventilation is not in service. As such, discharges to the plant vent are not ensured, and gases escaping from either ruptured tank are released through building openings with the highest atmospheric dispersion factors. For the offsite dose, the most limiting atmospheric dispersion factor is from the north auxiliary building normal ventilation intake, and for the CR dose, the limiting release point is the south normal ventilation intake. The CR ventilation system is assumed to remain in the normal alignment since a safety injection signal, which is required to automatically place the system in the pressurization mode, is not received for this event. No credit for hold-up, dilution, or decay of the gases are assumed in the auxiliary building. Similarly, while the CR ventilation system filters would have no impact on this event, the CR ventilation system remains in the normal system alignment for the duration of the event.

3.9.4 WGDT and VCT Rupture Conclusion

The licensee's evaluation of the postulated WGDT and VCT accidents using the AST, concluded that the radiological consequences for the CR are within the accident dose criteria stated in 10 CFR 50.67. The licensee chose not to implement the AST for these accidents regarding the EAB and LPZ and will continue with their existing analysis and the acceptance criteria of 500 mrem whole body. The NRC staff found that in the event of WGDT or VCT rupture, the EAB, LPZ, and CR doses would meet the applicable criteria and are therefore acceptable.

3.10 Atmospheric Dispersion Estimates

3.10.1 Meteorological Data

In support of the atmospheric dispersion analyses presented in the November 14, 2014, LAR, the licensee provided meteorological data for calendar years 2002, 2004, 2005, 2007, and 2010. The licensee provided files for each year that contained hourly data on wind speed, wind direction, dry bulb temperature, and precipitation taken from the 10 meter level location on the shoreline meteorological tower, located slightly northwest of CNP on the Lake Michigan shoreline, and the 10 meter and 60 meter levels from the primary meteorological tower, located approximately one mile east of CNP. Meteorological data recovery for the 5 years was greater than 90 percent in total, as specified in RG 1.23 (Reference 21). The licensee provided data files for each year formatted for use as input files to the ARCON96 atmospheric dispersion computer code (Reference 36) to calculate the updated χ/Q values for the CR. The licensee developed a joint wind speed, wind direction, and atmospheric stability joint frequency distribution for the same 5 years as an input to the PAVAN atmospheric dispersion computer code (Reference 30) to calculate the updated χ/Q values for the EAB and outer boundary of the LPZ. The NRC staff performed a screening of meteorological data and developed RAIs related to the location of the meteorological towers used at the plant, data substitution used in the data sets, the effects of missing data, and the categorization of various wind speeds. In response to the NRC RAIs, the licensee submitted the February 19, 2016, RAI response (Reference 8), which included a revised meteorological data set for the years 2002, 2004, 2005, 2007, and 2010. After reviewing the revised data set and clarifications provided in subsequent RAI responses, NRC staff found that the meteorological data were consistent with guidance outlined in RG 1.23 and therefore acceptable for use in the calculation of atmospheric dispersion estimates.

3.10.2 Onsite Control Room Atmospheric Dispersion Estimates

In support of the LAR, the licensee calculated CR χ/Q values using onsite meteorological data from calendar years 2002, 2004, 2005, 2007, and 2010, and guidance provided in RG 1.194 (Reference 24). RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in design basis accident radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there were no unusual siting, building arrangements, release characterizations, source-receptor configurations, meteorological regimes, or terrain conditions that precluded the use of this model in support of the LAR.

In response to NRC RAIs, the licensee developed new CR χ/Q values using ARCON96 and updated meteorological data (10 meter level wind data from the shoreline tower, 60 meter level wind data from the primary tower, and atmospheric stability data calculated from the temperature difference between the 60 meter and 10 meter levels on the primary tower) for the 2002, 2004, 2005, 2007, and 2010 calendar years. The data were formatted for use with ARCON96, and were provided in February 19, 2016, RAI response. The new CR χ/Q values were listed in tables provided in the licensee's May 6, 2016, RAI response (Reference 9).

The NRC staff reviewed the data, inputs, and assumptions used for the licensee's assessment and found them generally consistent with site configuration drawings. The staff also reviewed the licensee's assessments of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling for the extended accident duration. On the basis of this review and the staff's confirmatory calculations using ARCON96, the NRC staff concluded that the licensee's estimated CR χ/Q values are acceptable for use in the proposed radiological consequence assessments in support of this LAR.

3.10.3 Offsite EAB and LPZ Atmospheric Dispersion Estimates

In support of the November 14, 2014, LAR, the licensee calculated EAB and LPZ χ/Q values using meteorological data from calendar years 2002, 2004, 2005, 2007, and 2010, and guidance provided in RG 1.145 (Reference 23).

In response to NRC RAIs, the licensee developed new EAB and LPZ χ/Q values using PAVAN and the joint frequency distributions developed from updated meteorological data for the 2002, 2004, 2005, 2007, and 2010 calendar years that were provided in February 19, 2016, supplement. The new EAB and LPZ χ/Q values were listed in tables provided in the licensee's May 6, 2016, supplement. Wind speed values from either the shoreline tower or the primary tower were used, depending on the compass sector of the exposed population. Atmospheric stability was always based on temperatures measured on the primary tower.

The NRC staff reviewed the inputs and assumptions used for the licensee's assessment and found them consistent with site configuration drawings and staff practice. In addition, staff reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this review and the staff's confirmatory calculations using PAVAN, NRC staff concluded that these χ/Q values were acceptable for use in the proposed radiological consequence assessments in support of this LAR.

3.10.4 Atmospheric Dispersion Estimates Conclusion

The NRC staff performed a screening of meteorological data and developed RAIs related to the location of the meteorological towers, data substitution, the effects of missing data, and wind speed categorization. In response to the NRC RAIs, the licensee submitted a revised meteorological data set in the February 19, 2016, RAI response. NRC staff reviewed the licensee's atmospheric dispersion analysis and requested additional information on several issues regarding use of meteorological data and how these data were input to atmospheric dispersion calculations. In the February 19, 2016, RAI response, the licensee also provided clarifications for the onsite and offsite analysis; in the May 6, 2016, RAI response, the licensee provided updated χ/Q values for each analysis. NRC staff reviewed the responses from the licensee and performed confirmatory analyses. On the basis of this review and the staff's confirmatory analyses, the staff concluded that the meteorological data and the onsite and offsite χ/Q values were acceptable for use in the proposed radiological consequence assessments in support of this LAR.

3.11 Thermal Hydraulic (TH) Parameters

Enclosure 12 of the LAR lists the various new AST input parameter values, including those based on RCS performance, for offsite and CR habitability doses for each DBA analysis. Enclosure 12 also provides a comparison between the design input values used in the current licensing basis (CLB) dose consequence analyses supporting CNP, Units 1 and 2, and those utilized in the AST analyses supporting this LAR. Additionally, Enclosure 12 provides a description of the change for each DBA analysis listed above.

In reviewing the LAR, the NRC staff determined that additional information about the TH analysis applied as the basis for the values was necessary. The licensee's supplemental letter dated February 12, 2015 (Reference 2), stated the following:

> The current licensing basis (CLB) TH calculations were used to provide input to most of the new dose analyses, although some of those inputs are different from previous inputs that were derived from the same TH calculations.

The supplemental letter also states that the majority of the input parameters originate from calculations performed for previous submittals, such as CNP Units 1 and 2 license amendment Nos. 271 and 252 for implementation of AST for CR habitability, and license amendment Nos. 256 and 239 to address SGTR overfill. Other inputs were obtained from:

- Projects implemented under 10 CFR 50.59, such as the Unit 1 replacement SG modification,
- The CR habitability and offsite dose consequence analyses revised in 2011 and implemented under 10 CFR 50.59,

- Information obtained from actual plant post-trip data recorded by the CNP, Units 1 and 2, Plant Process Computer (PPC), and
- Information obtained from simulator data representing a Unit 1 SGTR transient including operator actions.

The NRC staff requested that the licensee provide the necessary additional information to connect each of the input parameter values provided in Enclosure 12 of the LAR to its respective source documentation. Additionally, the NRC staff requested that the licensee provide for inspection the source documentation and verification which produced the input parameter values being applied in each accident analysis. I&M provided its RAI response in a submittal dated August 24, 2015 (Reference 4). This RAI response gave additional information as to the sources for the RCS input parameter values; however, it was not known whether the proprietary source documents had been reviewed by the NRC staff in prior LARs. To verify the authenticity of the RCS input parameter values, NRC staff conducted an audit to review the supporting TH documentation and TH calculation files for this LAR. The NRC staff held an audit during the week of September 21, 2015, at the offices of I&M to review the supporting documentation and calculation files for this LAR. Audit results are documented in a report dated January 20, 2016 (Reference 56).

The NRC staff reviewed the documents listed in Table 1 of the audit report to determine the connections from the principal calculation files for the current full-implementation AST LAR input parameter values, through the common parameter reference document, to the source calculation files. For example, the NRC staff found that two documents in particular were relied upon for the full-implementation AST LAR from the previously approved partial implementation license amendment. An additional observation at the audit was that the licensee relied on certain TH input parameter values that were derived from plant operator simulator data. The NRC staff determined that this TH data used by the licensee did not conform to NRC guidance in RG 1.203 and SRP Section 15.0.2.

I&M provided a supplement to the LAR on May 6, 2016 (Reference 9). This supplement provided additional information regarding the sources for the TH parameter values concerning SG tube uncovery time based on CNP Units 1 and 2 PPC data, and flashing fractions derived from a previously NRC-reviewed LOFTRN2 safety calculation. To verify the veracity and authenticity of the information and its sources for these two sets of TH parameter values, the NRC staff conducted a second audit to review the supporting documentation to the May 6, 2016, RAI response. The results of the audit are documented in an audit summary dated July 25, 2016 (Reference 57).

The NRC staff reviewed the technical information provided in the two supporting documents to ensure that the evaluations were reasonable and applicable for their use in the CNP AST radiological analyses. For the SG tube uncovery time, the licensee applied an alternative approach based on actual plant data as the basis of the SG tube uncovery time. The NRC staff reviewed prior AST LARs and determined there were other NRC reviewed and approved AST safety analyses with similar results as the CNP SG tube uncovery time. The licensee derived flashing fractions from primary and secondary system TH information obtained from TH calculations that were previously reviewed by NRC and used NRC-approved methodologies.

Therefore, the second NRC staff audit verified the proper application of TH input parameter values for SG tube uncovery time and flashing fractions in the licensee's CNP AST radiological analyses. The staff concluded that the licensee's method of determining these values was reasonable as solely applied in support of the CNP AST LAR.

3.11.1 TH Parameters Conclusion

The NRC staff reviewed the LAR, RAI responses, and the LAR references provided during the two audits to determine if the associated calculations and assumptions conformed to RG 1.183. The NRC staff found the licensee's final set of AST TH inputs, assumptions, and results were appropriate for use in the dose calculations for the AST radiological analysis. The NRC staff has reviewed the licensee's TH parameter values which support the proposed AST LAR and finds the TH parameter values support the proposed change in accordance with the regulatory requirements discussed in Section 2.0 of this SE. The NRC staff concludes that the licensee's TH analysis for the AST conforms within the NRC guidance and complies with the regulatory requirements of 10 CFR 50.67, 10 CFR 100.11, and PSDC 11.

3.12 Environmental Qualification (EQ) of Equipment

The licensee proposed the implementation of the AST methodology for CNP radiological dose consequence analyses for off-site dose analyses and to update the dose analyses for CR habitability. The licensee requested to revise UFSAR, Chapter 14, DBA analyses to fully implement the AST methodology for off-site dose consequences and to update the CR analyses using the AST methodology outlined in RG 1.183. As stated in RG 1.183, Regulatory Position 6, "Assumptions for Evaluating the Radiation Doses for Equipment Qualification," the licensee may use either the AST or the TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. This generic issue has been resolved in an internal NRC memo dated April 30, 2001 (Reference 58) and in Supplement 25 to NUREG-0933 (Reference 16). The NRC staff concluded in the memo and NUREG-0933 that there was no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST, and there would be no discernable risk reduction associated with such a requirement. Therefore, in view of the cited references, the staff finds that it is acceptable for the TID-based assumptions to remain the licensing basis for equipment EQ analyses for CNP. The NRC staff reviewed the LAR to determine the impact the proposed changes may have on the safety-related electrical systems and EQ of electrical equipment.

In the LAR submittal, under Enclosure 10, "D.C. Cook AST RG 1.183 Compliance Matrix," the licensee stated that an exception for RG 1.183, Section 4.3 is taken to state that the CNP EQ analyses will continue to be based on TID-14844 in the EQ program, other than CR habitability. In an RAI, the NRC staff requested the licensee to provide a summary of any changes made to the EQ analyses that will demonstrate that the EQ electrical equipment will continue to meet its intended safety function. In letter dated August 28, 2015 (Reference 5), the licensee submitted a letter confirming that no changes will be made to the facility or fuel as a result of the AST implementation, and that it will not impact the current EQ analyses. The LAR states that the AST implementation is only to revise the dose consequences at the site boundaries and to the CR personal onsite. Additionally, the staff requested the licensee to provide a list and description of components that may be added to its 10 CFR 50.49 program due to this LAR and, if applicable, to confirm that these components are qualified for the environmental conditions to which they are expected to be exposed. The licensee stated in the letter dated August 28, 2015, that no new plant components are credited in the analysis of the AST, and therefore no components are being added to the 10 CFR 50.49 program. In addition, the licensee has stated in its response that all components credited in the analysis are already part of, and maintained under, the CNP EQ program. Therefore, CNP EQ electrical components will continue to meet their safety functions under 10 CFR 50.49.

In the LAR submittal, under Enclosure 10, "D.C. Cook AST RG 1.183 Compliance Matrix," the licensee stated in part, that for RG 1.183, Section 5.1.2: "assumptions regarding the occurrence and timing of a loss of offsite power (LOOP) are made with the objective of maximizing the impact on dose." In an RAI, the NRC staff requested that the licensee explain how the assumptions used for the LOOP analysis will affect the current EQ analysis.

The licensee stated in the letter dated August 28, 2015, that the assumptions made in the dose analysis regarding a LOOP event relates to the number of trains of safety-related equipment available to mitigate the event, and to the equipment response times due to the emergency diesel generator (EDG) start and sequencer operation. In addition, the licensee stated in its response that since the EQ analysis is not being changed due to AST implementation, the assumptions used for LOOP analysis will have no effect on the current EQ analysis. Based on this information, the staff finds that the current LOOP analysis remains unchanged and has no impact on the EQ analysis of electrical equipment at CNP.

The NRC staff requested additional information regarding whether any non-safety related systems and components are credited in the AST analyses. This is to ensure that the selected systems that have been credited in this analysis have been qualified in their respective environments and continue to be in compliance with the 10 CFR 50.49 requirements. The licensee has stated in RAI response letter dated August 28, 2015, that there are no non-safety related systems or components credited in the AST analyses.

The NRC staff further questioned whether any loads were being added to the CNP EDGs and if so, how the loads being added to the EDGs affect the capability and capacity of the EDGs. The licensee has stated in its RAI response letter dated August 28, 2015, that no new loads are being added to the CNP Class-IE Diesel Generators and no changes to the CNP EDG loading sequence have been made to support the revised dose analyses. Therefore, there are no changes made to the CNP EDG loads due to this LAR. Based on the above information, the staff finds that the EDGs have sufficient capacity and capability to perform its safety function and continues to meet the PSDC 39 requirements.

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 EQ of equipment will remain bounding during implementation of the proposed TS change.

3.12.1 EQ of Equipment Conclusion

Based on the above evaluation, the NRC staff finds the proposed changes to the CNP licensing basis and TS changes at CNP acceptable due to no changes to the loading on EDGs and EQ of electrical equipment. The licensee continues to comply with 10 CFR 50.49, 10 CFR 50.67, and PSDC 39. Therefore, the staff finds the proposed changes acceptable with regard to electrical systems.

3.13 TSTF-490 TS Changes

3.13.1 Revision to the Definition of Dose Equivalent I-131

The licensee proposes to revise the definition of dose equivalent I-131 to state:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine 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 thyroid dose conversion factors from Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

TSTF-490 lists the acceptable thyroid DCFs for use in the determination of dose equivalent I-131 as the following:

- Table III of TID-14844
- Table E-7 of RG 1.109
- International Commission on Radiological Protection (ICRP) 30
- CDE or CEDE DCFs from Table 2.1 of EPA FGR-11
- Table 2.1 of EPA FGR-11

The licensee proposed the use of the DCFs from CDE or CEDE DCFs from Table 2.1 of EPA FGR-11. Therefore, the NRC staff concludes that the licensee's proposed revision of the dose equivalent I-131 definition is consistent with the NRC staff-approved TSTF-490, and is therefore acceptable.

3.13.2 Deletion of the Definition of E Bar and the Addition of a New Definition for Dose Equivalent Xe-133

The NRC staff uses source terms for a variety of purposes. As described in SRP Section 11.1, "Coolant Source Terms," a design basis source term for the RCS is typically based on 0.25 to 1 percent fuel defects. This RCS source term is then generally applied to those DBAs which do not result in fuel damage. The licensee's analysis assumes an initial condition of the RCS source term equivalent to 1 percent fuel defects with no further damage occurring as a result of the accident. The assumption of an initial condition of a source term equivalent to 1 percent failed fuel for these accidents is the basis for the established CNP TS 3.4.16 limit on dose equivalent Xe-133 activity.

The new definition for dose equivalent Xe-133 is similar to the definition for dose equivalent I-131. The determination of dose equivalent Xe-133 will be performed in a similar manner to that currently used in determining dose equivalent I-131, except that the calculation of dose equivalent Xe-133 is based on the acute dose to the whole body and considers the noble gases 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 dose equivalent Xe-133 would use the effective DCFs from Table III.1 of EPA FGR-12. 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 E Bar. 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 dose equivalent Xe-133.

When E Bar 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 E Bar 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 or activation products, resulting in the determination of a value of E Bar 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 LCO becomes essentially meaningless. It also results in a TS limit that can vary during operation as different values for E Bar are determined.

This change will implement a 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 E Bar definition. LCO 3.4.16 specifies the limit for primary coolant gross specific activity as 100/E Bar μCi/gm. The current E Bar definition includes radioisotopes that decay by the emission of both gamma and beta radiation. The current Condition B of LCO 3.4.16 would rarely, if ever, be entered for exceeding 100/E Bar since the calculated value is very high (the denominator is very low) if beta emitters such as tritium (H-3) are included in the determination, as required by the E Bar definition.

The TS Section 1.1 definition for E - AVERAGE DISINTEGRATION ENERGY is deleted and replaced with a new definition for DOSE EQUIVALENT XE-133 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 dose equivalent Xe-133 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.

3.13.3 LCO 3.4.16, "RCS Specific Activity"

LCO 3.4.16 is modified to specify that iodine specific activity in terms of dose equivalent I-131 and noble gas specific activity in terms of dose equivalent Xe-133 shall be within limits. Currently, the limiting indicators are not explicitly identified in the LCO, but are instead defined in current Condition C and Surveillance Requirement 3.4.16.1 for gross non-iodine specific activity and in current Condition A and Surveillance Requirement 3.4.16.2 for iodine-specific activity. The change states "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133 specific activity shall be within limits." The NRC staff concludes that the proposed change is acceptable because it is consistent with the NRC-approved TSTF-490.

3.13.4 TS 3.4.16 Applicability

TS 3.4.16 Applicability is modified to include all of Modes 3 and 4. 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. In Mode 5 with the RCS loops filled, the SGs 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 in Mode 6, the SGs are not used for decay heat removal, the RCS and SGs are depressurized and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required. The change to modify the TS 3.4.16 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 from a radiological dose perspective.

3.13.5 TS 3.4.16 Condition A

TS 3.4.16 Condition A is revised by replacing the dose equivalent I-131 site specific limit "> 1.0 μCi/gm" with the words "not within limit" to be consistent with the revised TS 3.4.16 LCO format. The site specific dose equivalent I-131 limit of 1.0 μCi/gm is contained in surveillance requirement 3.4.16.2. This proposed format change will not alter current standard technical specification requirements and is acceptable from a radiological dose perspective.

TS 3.4.16 Required Action A.1 is revised to remove the reference to Figure 3.4.16-1, "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 dose equivalent I-131 spiking limit. The curve contained in Figure 3.4.16-1 was provided by the AEC in a June 12, 1974, letter from the AEC on the subject, "Proposed Standard Technical Specifications for Primary Coolant Activity." Radiological dose consequence analyses for SGTR and MSLB accidents that take into account the pre-accident iodine spike do not consider the elevated RCS iodine specific activities permitted by Figure 3.4.16-1 for operation at power levels below 80 percent rated thermal power. Instead, the pre-accident iodine spike analyses assume a dose equivalent I-131 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 operation. It is acceptable that TS 3.4.16 Required Action A.1 should be based on the short term site specific dose equivalent I-131 spiking limit, in order to be consistent with the assumptions contained in the radiological consequence analyses.

3.13.6 TS 3.4.16 Condition B Revision to include Action for Dose Equivalent Xe-133 Limit

TS 3.4.16 Condition B is revised to include dose equivalent Xe-133 not within limits. This change is made to be consistent with the change to the TS 3.4.16 LCO, which requires the dose equivalent Xe-133 specific activity to be within limits. The dose equivalent Xe-133 limit is site specific and the numerical value in units of μCi/gm is contained in revised Surveillance

Requirement 3.4.16.1. The site specific limit of dose equivalent Xe-133 in μCi/gm is established 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 DCFs. The primary purpose of the TS 3.4.16 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 E Bar definition.

In the original LAR, the licensee requested to modify Condition B to be consistent with TSTF-490, Revision 0, to have a Completion Time of 48 hours to restore dose equivalent Xe-133 to within limits. A Note was also added allowing the applicability of LCO 3.0.4.c.

The justification for this change as described in TSTF-490, Revision 0, is as follows:

> The Completion Time for revised TS 3.4.16 Required Action B.1 will require restoration of DOSE EQUIVALENT XE-133 to within limit in 48 hours. This is consistent with the Completion Time for current Required Action A.2 for DOSE EQUIVALENT I-131. The Completion Time of 48 hours for revised Required Action B.1 is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period.

While it is a correct statement that the proposed change makes the Completion Times of TS 3.4.16 Required Action A.2 and B.1 consistent, it is not clear why the Completion Times should be consistent generically since specific plant Conditions for these Required Actions are typically different.

The licensee's current TS 3.4.16, Condition A, of the ACTIONS table specifies that when dose equivalent I-131 is greater than 1.0 μCi/gm, the required action is to restore dose equivalent I-131 to within limit, with a Completion Time of 48 hours. During this Condition, LCO 3.0.4.c is applicable. If the dose equivalent I-131 is not restored within limit in 48 hours, the required action is to be in Mode 3, with average RCS temperature less than 500 ° F, within 6 hours.

The CNP TS 3.4.16 Required Action A.2 is required when the plant is in a condition analyzed in the DBA analyses where the reactor coolant dose equivalent I-131 is bound between 1 and 60 μCi/gm. The TSTF-490, Revision 0, TS 3.4.16 Required Action B.1 is required when the plant is in a condition not analyzed in the DBA analyses in which the dose equivalent Xe-133 is greater than 215.1 μCi/gm. Typically, the Required Action for a condition not analyzed requires the plant to take immediate actions to begin shutdown of the plant. The proposed TSTF-490, Revision 0, changes do not require immediate actions to begin shutdown of the plant, but allows 48 hours before the plant is required to begin shutting down.

In response to an RAI in a letter dated July 12, 2016 (Reference 10), a departure from the approved TSTF-490, Revision 0, was requested by the licensee. The licensee conducted additional reviews of the information related to the establishment of the Completion Time for TS 3.4.16, Condition B. As described in the proposed TS Bases B 3.4.16, "Applicable Safety Analyses," the analyses for the MSLB and SGTR DBAs establish the acceptance limits for RCS specific activity. For those accidents, the analysis assumptions include an initial condition of a source term equivalent to 1 percent failed fuel, but with no further fuel damage occurring as a result of the accident. The assumption of an initial condition of a source term equivalent to 1 percent failed fuel for these accidents is the basis for the established CNP TS 3.4.16 limit on dose equivalent Xe-133 activity.

As explained by the licensee, any spike in the RCS dose equivalent Xe-133 activity levels that exceed the limit specified in TS 3.4.16 would be expected to be caused by failed fuel elements, and dose equivalent Xe-133 activity increases caused by failed fuel cannot be returned to acceptable levels within 48 hours. Since there is not a current analysis that provides a value for dose equivalent Xe-133 that is greater than 215.1 µCi/gm, but below which the radiological consequences of the increased activity would not exceed the 10 CFR 50.67 dose guidelines, the licensee concluded that the allowed Completion Time for Condition B of TS 3.4.16 should not be changed to reflect the guidance of TSTF-490, Revision 0. Therefore, based on this conclusion, the licensee revised the LAR so that the Completion Time is consistent with the Completion Time for other unanalyzed conditions, which require the unit to be in Mode 3 within six hours, and Mode 5 within 36 hours, if the dose equivalent Xe-133 activity limit is exceeded.

3.13.7 Surveillance Requirement 3.4.16.1 Dose Equivalent Xe-133 Surveillance

The change replaces the current Surveillance Requirement 3.4.16.1 surveillance for RCS gross specific activity with a surveillance to verify that the site specific reactor coolant dose equivalent Xe-133 specific activity is ≤ 215.1 µCi/gm. This change provides a surveillance for the new LCO limit added to TS 3.4.16 for dose equivalent Xe-133. The revised surveillance requirement 3.4.16.1 surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days, which is the same frequency required under the current Surveillance Requirement 3.4.16.1 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 dose equivalent Xe-133 allow proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

3.13.8 Surveillance Requirement 3.4.16.3 Deletion

The current surveillance requirement 3.4.16.3 which required the determination of E Bar is deleted. TS 3.4.16 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 E Bar definition. With the elimination of the limit for RCS gross specific activity, and the addition of the new LCO limit for noble gas specific activity, this Surveillance Requirement to determine E Bar is no longer required.

3.13.9 TSTF-490 TS Changes Conclusion

The NRC staff has reviewed the proposed TS changes related to TSTF-490, to include revising the definition of dose equivalent I-131, deleting the definition of E Bar, adding a new definition for dose equivalent Xe-133, modifying TS Section 3.4.16, and deleting Figure 3.4.16-1. In addition, the NRC staff has evaluated the consistency of site-specific limits and DCFs for dose equivalent I-131 and dose equivalent Xe-133. The NRC staff concludes that the proposed changes are consistent with the NRC-approved TSTF-490, and that all deviations have been assessed appropriately in accordance with approved guidance and regulations. Therefore, the proposed TS changes related to TSTF-490 are acceptable.

3.14 Technical Evaluation Conclusion

The NRC staff has reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs for full implementation of an AST and implementation of TSTF-490 at the CNP, Units 1 and 2. The NRC staff concludes that the licensee used methods of analysis and assumptions consistent with the conservative regulatory requirements and guidance described in Section 2.0 of this SE. The NRC staff compared the doses estimated by the licensee to the applicable dose guidelines and criteria referenced in Section 2.0. Based on that comparison, the NRC staff concludes that the licensee's estimates of the EAB, LPZ, and CR doses comply with the regulatory requirements. The proposed changes will not impact the dose consequences of the applicable DBAs because the proposed changes will limit the RCS iodine and noble gas specific activity to ensure consistency with the values assumed in the site-specific DBA radiological consequence analyses. The NRC staff also concludes that there is reasonable assurance that CNP Units 1 and 2, as modified by the requested license amendment, will continue to provide sufficient safety margins and adequate defense-in-depth, under conditions of unanticipated events, and in the presence of the uncertainties in accident progression, assumptions, parameters, and analyses outlined above. Therefore, the proposed changes to the licensing basis are acceptable with respect to the radiological consequence of the DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the CNP Units 1 and 2 licensing basis is superseded by the revised licensing basis, incorporating the AST as 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 guidelines and criteria of 10 CFR 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the CNP, Units 1 and 2, design basis, and modified by the present amendment.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the 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 as published in the *Federal Register* on March 31, 2015 (80 FR 17091). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

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15. Memo from J. Rosenthal to A. Thodani, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" April 30, 2001 (ADAMS Accession No. ML011210348).

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Date of issuance: October 20, 2016

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

Sincerely,

/RA/

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 332 to DPR-58
2. Amendment No. 314 to DPR-74
3. Safety Evaluation

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