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

June 19, 2009

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO LOSS-OF-COOLANT (LOCA) ACCIDENT AND MAIN STEAM LINE BREAK (MSLB) ACCIDENT RADIOLOGICAL DOSE CONSEQUENCES ANALYSES AND AFFECTED TECHNICAL SPECIFICATIONS (TAC NOS. MD9140 AND MD9141)

Dear Mr. Wadley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. DPR-42 and Amendment No. 180 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 26, 2008, as supplemented by letters dated March 16 and May 1, 2009.

The amendments revise the Facility Operating Licenses by revising the licensing basis LOCA and MSLB accident radiological dose consequences as currently described in the PINGP Updated Safety Analysis Report Sections 14.5 and 14.9. The amendments also revise PINGP TSs 3.3.5, "Containment Ventilation Isolation Instrumentation", 3.4.17, "RCS Specific Activity", and 3.6.3, "Containment Isolation Valves", which are necessary to implement the revised analyses.

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

Sincerely,

Thomas J. Wengert

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

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 191 to DPR-42
2. Amendment No. 180 to DPR-60
3. Safety Evaluation

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

NORTHERN STATES POWER COMPANY*

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. DPR-42

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

   A. The application for amendment by Nuclear Management Company, LLC* (the licensee), dated June 26, 2008, as supplemented by letters dated March 16 and May 1, 2009, 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 Facility Operating License No. DPR-42 is hereby amended to read as follows:

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* On September 22, 2008, Nuclear Management Company, LLC (NMC), transferred its operating authority to Northern States Power Company, a Minnesota corporation (NSPM). By letter dated September 3, 2008, NSPM stated that it would assume responsibility for actions and commitments submitted by NMC.

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 191 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois M. James

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: June 19, 2009

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

NORTHERN STATES POWER COMPANY*

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-60

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

   A. The application for amendment by Nuclear Management Company, LLC* (the licensee), dated June 26, 2008, as supplemented by letters dated March 16 and May 1, 2009, 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 Facility Operating License No. DPR-60 is hereby amended to read as follows:

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* On September 22, 2008, Nuclear Management Company, LLC (NMC), transferred its operating authority to Northern States Power Company, a Minnesota corporation (NSPM). By letter dated September 3, 2008, NSPM stated that it would assume responsibility for actions and commitments submitted by NMC.

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois M. James

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: June 19, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 191 AND 180

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Facility Operating License No. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by a marginal line.

REMOVE	INSERT
DPR-42, License Page 3	DPR-42, License Page 3
DPR-60, License Page 3	DPR-60, License Page 3

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

REMOVE	INSERT
3.3.5-1	3.3.5-1
3.3.5-2	---
3.3.5-3	---
3.3.5-4	---
3.4.17-1	3.4.17-1
3.4.17-3	3.4.17-3
3.4.17-4	3.4.17-4
3.6.3-1	3.6.3-1
3.6.3-5	3.6.3-5
3.6.3-6	3.6.3-6
3.6.3-7	3.6.3-7

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

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

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

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

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 191 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

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

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

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

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

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

3.3 INSTRUMENTATION

3.3.5 Not used

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Specific Activity

LCO 3.4.17 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.5 μCi/gm.	-----------------NOTE--------------- LCO 3.0.4.c is applicable. ---------------------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.17-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.17.2	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.5 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.17.3	-----NOTE----- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ----- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

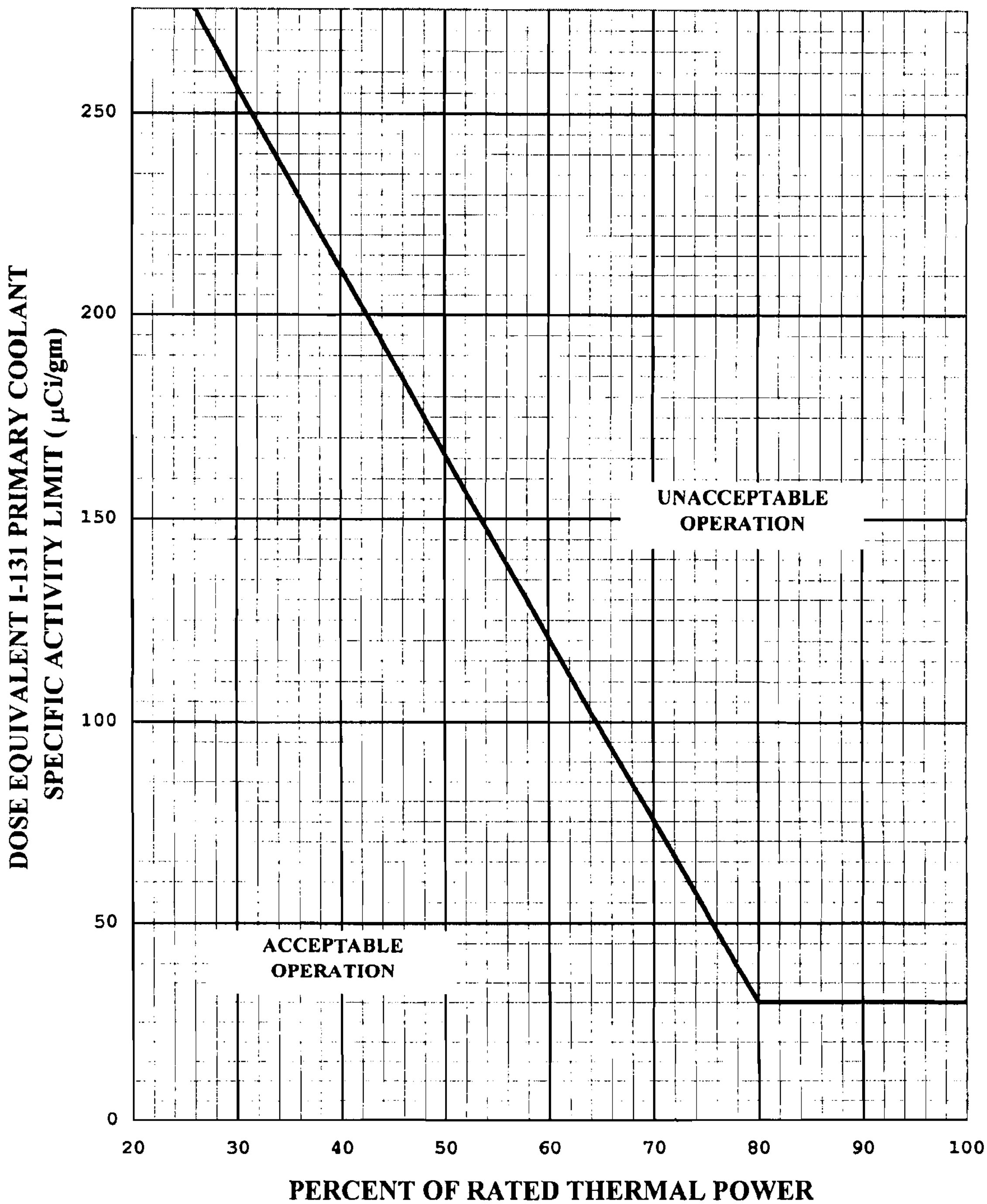

Figure 3.4.17-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

--NOTES--

1. Penetration flow path(s) except for 36-inch containment purge and 18-inch inservice purge system flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more secondary containment bypass leakage not within limit.	D.1 Restore leakage within limit.	4 hours
E. Containment purge blind flange or inservice purge blind flange leakage not within limit.	E.1 Restore leakage within limit.	1 hour
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 36-inch containment purge penetration blind flange is installed.	Prior to entering MODE 4 from MODE 5
SR 3.6.3.2	Verify each 18-inch containment inservice purge penetration blind flange is installed.	Prior to entering MODE 4 from MODE 5
SR 3.6.3.3	---------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	92 days
SR 3.6.3.4	---------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.3.5 Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6 Not Used	
SR 3.6.3.7 Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.6.3.8 Verify the combined leakage rate for all secondary containment bypass leakage paths is in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 180 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated June 26, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081790439), as supplemented by letters dated March 16 (ADAMS Accession No. ML090890180), and May 1 (ADAMS Accession No. ML091210703), 2009, Nuclear Management Company, LLC, a predecessor license holder to Northern States Power Company, a Minnesota corporation (NSPM, the licensee), requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP). The proposed changes would revise the Facility Operating Licenses by revising the licensing basis loss-of-coolant accident (LOCA) and main steam line break (MSLB) accident radiological dose consequences as currently described in the PINGP Updated Safety Analysis Report (USAR) Section 14.5 and Section 14.9.

In an effort to support future facility changes at PINGP, the licensee performed a validation of the current design and licensing basis for radiological dose consequence analyses of USAR Chapter 14 accidents. The proposed amendment includes a dose consequence reanalysis of the design-basis LOCA and MSLB accident to address the nonconformance issues identified in the validation effort. The licensee proposed that the new dose estimates replace the existing dose values as calculated using the accident radiological source term described in Technical Information Document (TID)-14844.

The licensee found that each of the issues that were identified fell into one of the following categories of nonconformance:

- some of the plant specific design parameters used in selected dose consequence analyses were not current (e.g., the core and coolant inventory, control room (CR) volume and unfiltered inleakage);

- in some cases, the atmospheric dispersion factors utilized did not address the actual release points/receptor locations; and

- there were some technical deficiencies in selected analyses (e.g., the LOCA analysis did not address all of the radioactivity release sources, and the activity transport model utilized for the dose assessment in the CR was found to be erroneous; the MSLB did not address the allowable primary coolant leakage when determining the equilibrium iodine appearance rates).

The revised dose consequence analyses submitted by the licensee in the subject license amendment request (LAR) are the result of the corrective actions taken, associated with the findings as categorized above. For this revision effort, the licensee corrected and updated design-basis accident (DBA) model parameters, inputs, and results of the specified dose consequence analyses. The result of this revision effort is evaluated by the U.S. Nuclear Regulatory Commission (NRC) staff herein.

The licensee also proposed concomitant TS amendments, which are necessary to implement the revised analyses. Specifically, the licensee proposed changes to:

1. TS 3.3.5, "Containment Ventilation Isolation Instrumentation," by deleting this section in its entirety.

2. TS 3.4.17, "RCS Specific Activity," by revising the action limit in Condition A and the acceptance limit in Surveillance Requirement (SR) 3.4.17.2 from 1.0 μCi/gm dose equivalent I-131 (DEI) to 0.5 μCi/gm and revising Figure 3.4.17-1 by reducing the Acceptable Operation DEI from 60 μCi/gm to 30 μCi/gm for plant operations 80 to 100 percent of rated thermal power.

3. TS 3.6.3, "Containment Isolation Valves," by revising Action Statements and SRs to require the Containment Inservice Purge System (CIPS) to be blind flanged during the plant operating Modes and removing CIPS operational requirements.

The supplemental information dated March 16 and May 1, 2009, contained clarifying information, did not change the scope of the June 26, 2008, application or the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

This safety evaluation (SE) describes the impact of the proposed changes on analyzed DBA radiological consequences. This evaluation has been conducted to verify that the results of the licensee's affected DBA radiological dose consequence analyses continue to meet the values referenced in 10 CFR Section 100.11, "Determination of exclusion area, low population zone, and population center distance," for offsite doses, and 10 CFR Part 50, Appendix A, GDC 19, "Control room," as dose acceptance criteria for the PINGP DBAs. For this evaluation, the applicable acceptance criteria are less than 5 rem Whole Body to CR personnel, and 25 rem Whole Body and 300 rem Thyroid at both the exclusion area boundary (EAB) and outer boundary of the low population zone (LPZ).

The licensee addressed the regulatory requirements applicable to the proposed amendment in Section 4.1 of Enclosure 1 to the application dated June 26, 2008. As described in this enclosure and confirmed by the PINGP USAR, Section 1.2 "Principal Design Criteria", PINGP is designed and constructed to meet the intent of the U. S. Atomic Energy Commission (AEC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as originally proposed in July 1967. Plant construction was significantly completed prior to the issuance of the February 20, 1971, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A GDC. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the Final Safety Analysis Report, against the Appendix A design criteria and "... are satisfied that the plant design generally conforms to the intent of these criteria." The regulatory requirements, criteria, and guidance applied by the NRC staff in the review of the proposed changes are as follows:

- Criterion 16, "Containment design," insofar as it requires that the containment and its associated systems (e.g., penetrations) be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- Criterion 19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the CR under accident conditions.

- Criterion 50, "Containment design-basis," insofar as it requires that the containment and its penetrations accommodate without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- Criterion 53, "Provisions for containment testing and inspection," insofar as it requires that provisions shall be provided for inspection and testing of containment penetrations which have resilient seals and expansion bellows.

- Criterion 56, "Primary containment isolation," insofar as it requires that containment penetrations shall be provided with redundant isolation valves.

Additionally, except where the licensee proposes a suitable alternative, the NRC staff used the regulatory guidance provided in applicable sections of the following guidance documents in performing this review:

- Regulatory Guide (RG) 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"

- RG 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Revision 2 (March 1978)

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

- NUREG-0800, Standard Review Plan (SRP), Chapter 15, for design-basis accidents, and SRP Chapter 6.4, for CR habitability

- NRC Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal" (June 1999)

- American Society for Testing Materials (ASTM) Standard D3803-89 "Standard Test Method for Nuclear-Grade Activated Carbon"

The NRC staff also referenced the PINGP TS and USAR for verification of design basis input, as the licensee relied on these documents for input to their analyses.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Enclosure 2 of the June 26, 2008, submittal. The findings of this SE are based solely on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

3.1 Technical Evaluation of Meteorological Data and Atmospheric Dispersion

3.1.1 Meteorological Data

The licensee used five years of onsite meteorological data collected during calendar years 1993 through 1997 to generate 30 sets of CR atmospheric dispersion factors (χ/Q values). The licensee selected the six limiting cases for input into the LOCA and MSLB dose assessments performed in support of the current LAR. The 1993 through 1997 data were provided for NRC staff review in the form of hourly meteorological data files in the ARCON96 atmospheric dispersion computer code input format (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). All releases were considered to be ground level and, as such, were modeled using wind data measured at 10 meters and 60 meters above ground level. The atmospheric stability categorization was also based on temperature difference measurements between these two levels. The resulting CR atmospheric dispersion factors represent a change from the values in the current PINGP USAR analysis. The 1993 through 1997 meteorological data are discussed further in the SE associated with PINGP, Unit 1 and Unit 2, Amendment Nos. 166 and 156, respectively, dated September 10, 2004 (ADAMS Accession No. ML042430504). The licensee used χ/Q values derived from the PINGP USAR for postulated releases to the EAB and the LPZ. These χ/Q values are discussed further in Section 3.1.3 below.

3.1.2 CR Atmospheric Dispersion Factors

The licensee calculated CR air intake and inleakage χ/Q values using 1993 through 1997 onsite meteorological data and the ARCON96 atmospheric dispersion computer code. RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual sitings, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for PINGP. The NRC staff notes the following:

- The licensee generated CR χ/Q values for five sets of release locations from both Unit 1 and Unit 2, namely: 1) the shield building wall, 2) shield building stack, 3) auxiliary building make-up (MU) air intake, 4) main steam safety valves (MSSVs) and steam generator power-operated relief valves (SG PORVS), and 5) SG PORVs and main steam line break-point. The shield building stack and SG PORVs and main steam line break-point releases were assumed to be point sources. Releases from the shield building wall, MU air intake, and MSSVs and SG PORVS were modeled as diffuse sources. The licensee modeled three intake locations, the Unit 1 and Unit 2 CR air intakes and the center of the CR, which was assumed to represent the average location of inleakage to the CR.

- The licensee provided the assumptions, inputs, and resultant χ/Q values for a full 30-day time period and identified the limiting six sets of χ/Q values for NRC staff review. These χ/Q values appear in Table 3.1 below. Control room air intake χ/Q values are limited to the 0-2 hour time period because of CR isolation. Similarly, releases from the shield building stack and MU air intakes were assumed to enter the CR only through the unfiltered inleakage pathway because releases from these sources were assumed to occur after CR isolation. Resultant χ/Q values associated with postulated releases from the SG PORVs and MSL break-point group of χ/Q bound the χ/Q values associated with the MSSVs and SG group.

- The MU air intake louvers on the sides of the auxiliary buildings face away from the CR and were modeled as two-dimensional vertical diffuse sources. The NRC staff has determined that this is acceptable in the current LAR since the source is applied only to the inleakage part of the dose assessment. However, if effluent from the MU air intake louvers is modeled to other receptors in future dose assessments, assumptions regarding the specific case should be evaluated.

- The licensee noted that the puff release of secondary system TS activity from the faulted steam generator following a MSLB assumed no credit for atmospheric dispersion.

In summary, the NRC staff has qualitatively reviewed a sample of the inputs used by the licensee and their assessment of CR post-accident dispersion conditions generated from the PINGP onsite meteorological data and the licensee's atmospheric dispersion modeling. The NRC staff has found them generally consistent with site configuration drawings and staff practice. On the basis of this review, the NRC staff has concluded that the limiting CR χ/Q values that the licensee used in its dose assessment, as listed in Table 3.1, are acceptable for use in the dose assessment for the current LAR. The NRC staff has not determined the acceptability of the non-limiting χ/Q values for possible use in future dose assessments.

3.1.3 EAB/LPZ Atmospheric Dispersion Factors

The licensee stated that the dose assessment for this LAR used guidance in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," but did not state that the current licensing basis EAB and LPZ χ/Q values were based upon RG 1.4. The NRC staff noted several differences in assumptions between the inputs used in the χ/Q calculations that appear in Appendix H of the PINGP USAR and the RG 1.4 default guidance. In Enclosure 1 of its May 1, 2009 letter response (ADAMS Accession No. ML091210703) to an NRC request for additional information (RAI), the licensee provided further discussion and a reference to Table XIV of Appendix H. This table was submitted to the Atomic Energy Commission (now NRC) in the early 1970's. Specific areas of note are as follows:

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," states that the maximum EAB total effective dose equivalent for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the applicable dose criteria. Therefore, the EAB χ/Q value should be representative of the limiting two hour time period. The χ/Q value listed in Table XIV at a distance of 715 meters, the PINGP EAB distance, used by the licensee in the dose assessment for the current LAR, is for a 0-8 hour time period. This is also the EAB χ/Q value used in the dose analysis supporting Amendments 166 and 156. Since the standard Gaussian short-term centerline atmospheric dispersion equation and inputs and assumptions discussed in Appendix H are also suitable for a 0-2 hour time period, the resultant χ/Q value could be applied to both a 0-2 hour and a 0-8 hour time period. Therefore, in this case, it is acceptable to use the value calculated for the 0-8 hour time period in the dose estimate for a 0-2 hour time period. However, the NRC staff notes that if a different methodology or different assumptions were used, it may not be appropriate to apply χ/Q values calculated for a 0-8 hour time period to a 0-2 hour dose assessment.

- The current LAR also utilizes the 0-8 hour LPZ χ/Q value used in the dose assessment that supports Amendments 166 and 156. However, the Amendment 166 and 156 dose assessment was limited to the 0-8 hour period and did not use 8-24 hour, 1-4 day or 4-30 day χ/Q values as required for the current LAR. Therefore, for the current LAR, the licensee derived LPZ χ/Q values at a distance of 2,414 meters for time periods greater than 8 hours by interpolation from χ/Q values in Table XIV, which lists distances ranging from 400 to 100,000 meters. While the NRC staff notes the following regarding the resultant LPZ χ/Q estimates, the interpolation itself appears reasonable.

The NRC staff made a scoping comparison calculation using RG 1.145, the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radiological Materials from Nuclear Power Stations") and current staff practice. The resultant 0-2 hour EAB and 0-8 hour and 8-24 hour LPZ χ/Q values calculated by the NRC staff were lower than the licensing basis χ/Q values used by the licensee in its dose assessment, while the 1-4 and 4-30 day LPZ χ/Q values were higher than the licensing basis χ/Q values used by the licensee. If other inputs are unchanged, a higher χ/Q value results in a higher dose estimate. The NRC staff also qualitatively estimated the effect of the differences in χ/Q values on the resultant dose estimates and has concluded that the licensee's use of the PINGP licensing basis χ/Q values is acceptable for this specific LAR. However, the NRC staff notes that this may not be the case for other dose applications. The licensee's EAB and LPZ χ/Q values are presented in Table 3.2.

3.2 Technical Evaluation of Radiological Consequences of Design-Basis Accidents

The NRC staff reviewed the impact of the proposed changes on DBA radiological consequence analyses, as documented in Chapter 14 of the PINGP USAR. The specific DBA analyses that were reviewed were as follows:

- Loss of Coolant Accident, and
- Main Steam Line Break Accident

The licensee determined, and the NRC staff agrees, that the LOCA and MSLB accident are the only DBA analyses that are affected by the proposed licensing and design basis changes. Information regarding the effect of the licensee's proposed changes on the design-basis LOCA and MSLB accident dose consequence analyses was provided by the licensee in Enclosure 2 of the application (ADAMS Accession No. ML081790439). The findings of this SE are based upon the descriptions and results of the licensee's assessment, additional information provided in response to the NRC staff RAIs, and other supporting information docketed by the licensee.

The following sections detail the safety evaluation performed by the NRC staff to address the impact of the licensee's proposed changes on the dose consequence of the design-basis LOCA and MSLB accident.

3.2.1 Loss of Coolant Accident (LOCA)

The current PINGP design-basis LOCA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The PINGP licensing basis is presented in USAR Chapter 14.9, "Environmental Consequences of Loss-Of-Coolant Accident."

For the LOCA analysis, the licensee assumed that the core isotopic inventory, which is available for release into the containment atmosphere, is based on maximum full power operation of the core at the current licensed thermal power of 1,650 MWth. The current licensed thermal power is then assumed to be increased by a factor of 1.02, to a value of 1,683 MWth, in order to account for power uncertainty. Additionally, current licensed values for fuel enrichment and burnup are assumed when determining the core isotopic inventory. Consistent with the current PINGP licensing basis, as based on TID-14844 methodology, the licensee assumed an instantaneous release of this activity, apportioned as 100-percent of the core noble gases, 50 percent of the core halogens (iodine), and 1 percent of the core remainder, into the containment following the postulated LOCA.

For the LOCA analysis, the licensee left the current licensing basis accident dose consequence analysis methodology unchanged. However, taking corrective actions, the licensee did change various analysis inputs, parameters, and assumptions as the result of its design basis verification effort.

3.2.1.1 Changes to Current Licensing and Design Basis

As a result of its corrective action efforts, the licensee made specific licensing and design basis changes to their PINGP LOCA analysis, as presented in the subject LAR. The following are the issues of radiological dose consequence concern that were identified by the licensee. The changes the licensee has proposed to address these issues, as well as the evaluation of these changes, as provided by the NRC staff, is also shown below.

Licensing Basis Issues

1. The current PINGP TS allow for the in-service purge system to operate during power operations, thus leading to an additional design-basis LOCA activity release pathway.

    a. Proposed Change - Remove the containment in-service purge capability from the PINGP TS. The proposed TS change modifies TS 3.6.3, "Containment Isolation Valves," to require verification that the in-service purge valves are blind-flanged in operating modes 1 through 4. Also, the proposed modification to TS 3.3.5, "Containment Ventilation Isolation Instrumentation," would remove maintenance requirements regarding instrumentation associated with the in-service purge function in operating modes;

    b. Evaluation - This TS change will not allow the in-service purge system to be operated during MODES 1, 2, 3, and 4, and would thereby eliminate one post-accident activity release path. This change reduces accident dose consequences, and is acceptable to the NRC staff.

2. The current design-basis LOCA analysis uses thyroid dose conversion factors (DCFs) from TID-14844 to estimate the thyroid doses at the EAB, LPZ and CR. These DCFs are not state-of-the-art, and more limiting than other DCF references that are found acceptable to the NRC staff.

    a. Proposed Change - Use inhalation dose conversion factors based on Table 2.1 of Federal Guidance Report (FGR) 11 to estimate the thyroid doses at the EAB, LPZ, and CR.
    b. Evaluation - The dose conversion factors contained in FGR 11 and 12 are generically acceptable to the NRC staff for use in calculating post-accident doses in DBA analyses.

3. The current design-basis LOCA analysis dismisses the dose contribution of engineered safety features (ESF) leakage based upon NUREG-0800, SRP 15.6.5.

    a. Proposed Change - Calculate the contribution from ESF leakage in the LOCA analysis.

    b. Evaluation - The licensee assumed that 50 percent of the core iodine and 1 percent of the core remainder is homogeneously mixed into the post-LOCA sump water volume of 250,874 gallons. Then, an ESF leakage rate of 1,106 cm^3 per hour into the Auxiliary Building Special Ventilation Zone is assumed. As documented in the PINGP USAR Table 6.7.2, this value represents one-half of the maximum (design) leakage possible from one operating system. ESF leakage is postulated to start at initiation of the recirculation mode which, at PINGP is conservatively assumed to be at 16.25 minutes. The NRC staff finds this to be a reasonable assumption. The licensee noted, and the NRC staff agrees that, due to the long term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses. The licensee states that the peak sump water temperature occurs at 6600 seconds and is 252.9 °F. The fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of the leakage that flashes to vapor. This assumption is conservative and acceptable to the NRC staff. The licensee calculated the flashing fraction at the assumed temperature to be less than 10 percent, and consequently, 10 percent of the halogens associated with this leakage is assumed to become airborne and is exhausted, without mixing or without holdup, to the environment via the Shield Building Vent Stack, after being processed by the Auxiliary Building Special Ventilation System (ABSVS) filters. The NRC staff accepts an assumption of 10 percent flashing, as it is commensurate with iodine flashing fractions calculated using conservative methodologies. Also, the assumed release pathway for this leakage was conservatively postulated by the licensee, and therefore, acceptable as well. The licensee assumes that the 1 percent of remaining core particulate stays in solution and is not released to the environment. This assumption is acceptable and also commensurate with applicable regulatory guidance.

4. The current design-basis LOCA analysis does not account for Refueling Water Storage Tank (RWST) back-leakage as a post-accident activity release path.

    a. Proposed Change - Evaluate RWST as a post-LOCA activity release path.

    b. Evaluation - The licensee postulated that sump water back-leakage into the RWST, which is located in the Auxiliary Building, occurs at a rate of 5 gallons per hour. Further, this release is assumed to be directly to the environment via the Auxiliary Building MU Air Intake louvers, which are the closest opening in the Auxiliary Building, located near the RWST. The licensee notes, and the NRC staff agrees, that a significant portion of the iodine associated with the RWST back-leakage is retained within the tank due to the equilibrium iodine distribution balance between the RWST gas and liquid fluid phases, which actually will result in a time-dependent iodine partition coefficient. Although they calculated the assumed RWST time-dependent venting, the licensee used a simplified conservative model similar to that which they used to simulate ESF leakage. The licensee assumed RWST back-leakage starts at initiation of the recirculation mode, at 16.25 minutes. The licensee further conservatively assumed that 0.1 percent of the iodine in the leaked fluid is airborne, is released out of the RWST via the vent, and is dispersed to the environment. The licensee assumed that all RWST iodine back-leakage is of the elemental chemical form. The NRC staff finds that the licensee's RWST back-leakage model is conservative and acceptable, as it does not take full credit for lower back-leakage rates that would be calculated when accounting for temperature and relative humidity (RH) variations of the fluid in the RWST and changing water pH.

Design Basis Issues

1. Current design-basis LOCA analysis does not account for uncertainty in core power when assuming core activity and inventory.

    a. Proposed Change - Assume 102 percent of core power (1,683 MWth) in LOCA analysis to account for power uncertainty.

    b. Evaluation - Increasing the licensed core power by 2 percent for analysis of the design-basis LOCA is commensurate with 10 CFR Part 50, Appendix K requirements. Therefore, this change is acceptable to the NRC staff.

2. The particulate removal rate (lambda) credited to sprays is incorrectly calculated.

    a. Proposed Change - Change the particulate removal rate (lambda) credited to sprays from 0.45 hr^{-1} to 5 hr^{-1}, by recalculating using the NUREG-0800, SRP 6.5.2.

    b. Evaluation - The use of NUREG-0800, SRP 6.5.2 guidance is acceptable to the NRC staff.

3. An erred delay is input to the current design-basis LOCA analysis at PINGP.

    a. Proposed Change - Change spray initiation time from 1 minute to 72 seconds, after accident initiation.

    b. Evaluation - When sprays are automatically initiated within 72 seconds, following accident initiation, the calculated dose consequences have been confirmed to be within applicable acceptance criteria. Therefore, this change is acceptable to the NRC staff.

4. An erred spray cut-off time is input to the current design-basis LOCA analysis at PINGP.

    a. Proposed Change - Change the spray cut-off time from 14.816 minutes to 14.516 minutes, as calculated to limit the elemental iodine removal to a dose factor (DF) of 100.

    b. Limiting the elemental iodine DF to 100 is commensurate with regulatory guidance, and the resulting calculated dose consequences have been confirmed to be within applicable acceptance criteria. Therefore, this change is acceptable to the NRC staff.

5. The current PINGP TS surveillance allows for crediting the ABSVS and Shield Building Ventilation System (SBVS) filter efficiency with 70 percent removal of elemental and organic forms of iodine.

a. Proposed Change - Change ABSVS and SBVS filter efficiency credited in the design basis analysis from 90 percent to 70 percent, for elemental and organic forms of iodine, to be consistent with current TSs.

b. Evaluation - Confirmatory analysis confirms that 70 percent filtration is sufficient to adequately mitigate activity release, and the calculated dose consequences have been confirmed to be within applicable acceptance criteria. Therefore, this change is acceptable to the NRC staff.

6. ABSVS credit does not account for Auxiliary Building drawdown.

   a. Proposed Change - Change Auxiliary Building drawdown time prior to crediting ABSVS from 0 seconds to 6 minutes.

   b. Evaluation - This change correctly considers ventilation system limitations by accounting for the time required to achieve adequate negative pressurization before crediting filtration features. The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

7. The CR volume assumed in the current DBA analysis is incorrect.

   a. Proposed Change - Change assumed CR volume from 44,200 ft^3 to 61,315 ft^3.

   b. Evaluation - The design parameter change is consistent with the PINGP design basis, and is therefore acceptable to the NRC staff.

8. Normal operation ventilation air intake prior to CR isolation was not accounted for.

   a. Proposed Change - Maximum normal operation ventilation air intake prior to CR isolation at 2 minutes was changed from 0 cubic feet per minute (cfm) to 2000 fm.

   b. Evaluation - The change is consistent with the PINGP design basis. The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

9. The CR unfiltered inleakage assumption of the DBA analysis does not account for the actual findings of tracer gas tests performed at PINGP.

   a. Proposed Change - Change the assumed unfiltered inleakage from 44 cfm to 165 cfm, per results of tracer gas test.

   b. Evaluation - The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

10. The current DBA accident analysis does not account for unfiltered inleakage into the CR due to ingress/egress.

    a. Proposed Change - Assume an additional unfiltered inleakage of 10 cfm, as recommended by NUREG-0800, SRP Section 6.4.

    b. Evaluation - The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

11. Incorrect CR Emergency Ventilation recirculation flow rate was assumed in the DBA analysis.

    a. Proposed Change - Change the CR Emergency Ventilation recirculation flow rate from 3000 cfm to 3600 cfm.

    b. Evaluation - The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

12. The filtration assumed for the CR recirculation filters is inconsistent with the PINGP TS Surveillance Requirements.

    a. Proposed Change - Change the filter efficiencies credited in the design basis analysis from 95 percent for all iodine forms, to 95 percent for elemental and organic iodine and 99 percent for particulate iodine, to be consistent with the current TS.

    b. Evaluation - This analytical parameter change is consistent with the current design and licensing basis of PINGP; therefore, this change is acceptable to the NRC staff.

3.2.2 Main Steam Line Break Accident

The current PINGP design basis MSLB analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The PINGP licensing basis is presented in USAR Chapter 14.5.5.6, "Dose Analyses for MSLB Outside of Containment." As part of the corrective actions taken to address the non-conformance issues that were discovered, the licensee re-analyzed the MSLB accident.

Because for PINGP, the licensee postulated no fuel damage associated with this accident, the main radiation source is the activity in the primary and secondary coolant system. For the primary coolant, the licensee addressed two spiking cases, a pre-incident iodine spike and a coincident iodine spike, commensurate with the applicable regulatory guidance.

- Pre-incident spike - the initial primary coolant iodine activity is assumed to be 60 times the proposed TS limit of 0.5 μCi/gm DE I-131, which is the proposed PINGP TS limit for full power operation. The licensee assumed the initial primary coolant noble gas activity to be at PINGP TS levels.

- Coincident spike - In accordance with the current PINGP licensing basis, immediately following the accident, the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the TS coolant concentrations. The duration of the postulated spike is assumed to be 8 hours. The licensee assumed the initial primary coolant noble gas activity to be at PINGP TS levels. The secondary coolant iodine activity, just prior to the accident is assumed to be at the PINGP TS limit of 0.1 μCi/gm DE I-131.

For the MSLB accident analysis, the licensee left the current licensing basis accident dose consequence analysis methodology unchanged. However, taking corrective actions, the licensee did change various analysis inputs, parameters, and assumptions as the result of its design basis verification effort.

3.2.2.1 Changes to Current Licensing and Design Basis

As a result of its corrective action efforts, the licensee made specific licensing and design basis changes to their PINGP MSLB accident analysis, as presented in the subject LAR. The following are the issues of radiological dose consequence concern that were identified by the licensee. The changes the licensee has proposed to address these issues, as well as the evaluation of these changes, as provided by the NRC staff, is also shown below.

<u>Licensing Basis Issues</u>

1. Reduction in TS-allowed primary coolant activity desired by the licensee.

    a. Proposed Change - Change the PINGP TS 3.4.17, "RCS Specific Activity," to reduce the limit on the primary coolant iodine concentration from 1.0 μCi/gm dose equivalent (DE) I-131 to 0.5 μCi/gm DE I-131.

    b. Evaluation - This revision will implement a limit that is more conservative than the existing requirement. The NRC staff finds this change acceptable, as it has been properly modeled and accounted for in the licensee's revised design basis analysis, and the dose consequences remain within applicable acceptance criteria.

<u>Design Basis Issues</u>

1. The current steam generator (SG) liquid mass used in the DBA analysis is incorrect.

    a. Proposed Change - Change the SG liquid mass from 109,155 lbs per SG to 107,100 lbs per SG.

    b. Evaluation - The change is consistent with the PINGP design basis, and represents a conservative increase in the activity concentration in the coolant available for release. The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

2. The current primary coolant volume used in the DBA analysis is incorrect.

   a. Proposed Change - Change the primary coolant volume from 5227 ft^3 to 5290 ft^3.

   b. Evaluation - The change is consistent with the PINGP design basis. The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

3. The CR volume assumed in the current DBA analysis is incorrect.

   a. Proposed Change - Change assumed CR volume from 165,000 ft^3 to 61,315 ft^3.

   b. Evaluation - The design parameter change is consistent with the PINGP design basis, and is therefore acceptable to the NRC staff.

4. The current DBA analysis does not account for unfiltered inleakage into the CR due to ingress/egress.

   a. Proposed Change - Assume an additional unfiltered inleakage of 10 cfm.

   b. Evaluation - The calculated dose consequences accounting for this change have been confirmed to be within applicable acceptance criteria; therefore, this change is acceptable to the NRC staff.

5. The filtration assumed for the CR recirculation filters is inconsistent with the PINGP TS Surveillance Requirements.

   a. Proposed Change - Change the filter efficiencies credited in the design basis analysis from 95 percent for all iodine forms, to 95 percent for elemental and organic iodine and 99 percent for particulate iodine, to be consistent with the current TS.

   b. Evaluation - This analytical parameter change is consistent with the current design and licensing basis of PINGP; therefore, this change is acceptable to the NRC staff.

3.3 Technical Evaluation of Containment and Ventilation Systems Changes

The licensee's effort to validate the current design and licensing basis of the PINGP radiological accident analyses revealed three significant issues. They are: 1) use of a non-conservative modeling factor to address activity buildup rate in the CR prior to reaching equilibrium concentrations; 2) the ventilation systems' elemental iodine filter efficiency assumed in the analysis did not incorporate the safety factor of 2 discussed in RG 1.52; and 3) the analysis was not based on the limiting unfiltered in-leakage into the CR envelope determined from tracer gas testing.

In addition to the modeling aspects, there are other technical deficiencies such as not addressing RWST back-leakage in the LOCA analysis, and allowable primary coolant leakage in MSLB accident analysis. Some of the plant-specific design parameters used in the current analyses were also found to be incorrect. This section of the SE addresses the issues related to filter efficiencies, unfiltered in-leakage into the CR, the proposed changes to TS 3.3.5 and TS 3.6.3, and other issues related to containment and ventilation systems.

3.3.1 PINGP Containment and Ventilation Systems Background

PINGP is a two unit plant, with each unit employing a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The containment consists of a primary containment system and a secondary containment system. The primary containment, also known as the reactor containment vessel, consists of a low leakage steel shell, including all its penetrations, designed to confine the radioactive materials that could be released by a loss of integrity of the reactor coolant pressure boundary. The primary containment also houses the ESF systems, including the containment isolation system. The principal function of the containment isolation system is to confine the fission products within the primary containment system boundary during post-accident conditions.

The secondary containment consists of the shield building, its associated ESF systems, and a special ventilation zone in the auxiliary building. The shield building is a reinforced concrete structure surrounding the reactor containment vessel. The shield building is designed to provide biological shielding for DBA conditions. In addition, it provides a means for collecting and filtration of fission-product leakage from the reactor containment vessel following a DBA. The SBVS is the ESF utilized for this function. The ABSVS is designed to collect any potential containment system leakage that might bypass the shield building annulus and pass it through charcoal filters before exhausting to the environment.

3.3.2 SBVS and ABSVS

The SBVS consists of two full-capacity redundant fan and filter trains. Each train consists of a heater, a pre-filter, moisture separators, an absolute filter, charcoal filter, an activated charcoal adsorber section, a recirculation fan and an exhaust fan. The absolute filter and the charcoal adsorber section are credited in the accident dose consequence analyses. The system takes suction from the shield building annulus (SBA) and is designed with the capability to either exhaust or recirculate the air, or perform a combination of both. The discharge for each fan splits into two flow paths, the exhaust and recirculation paths. The exhaust flow is directed to the exhaust vent located in the SBA and the recirculation flow is returned to the annulus. The system is started automatically on receipt of a safety injection (SI) signal.

In the re-evaluation of dose consequences, the SBVS charcoal filter efficiency for elemental and organic iodines was changed from 90 percent to 70 percent. NRC Generic Letter (GL) 99-02 stated that (1) the laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the radiological consequence analysis is still valid at the end of the operating cycle, (2) because ASTM D3803-1989 "Standard Test Method for Nuclear-

Grade Activated Carbon" is a more accurate and demanding test than older tests, licensees that upgrade their TS to this new protocol will be able to use a safety factor as low as 2 for determining the acceptance criteria for charcoal filter efficiency, and (3) this safety factor can be used for systems with or without humidity control because the lack of humidity control is already accounted for in the test conditions (systems without humidity control can test at 95 percent RH and systems with humidity control can test at 70 percent RH). PINGP TS section 5.5.9 "Ventilation Filter Testing Program (VFTP)", subsection c, requires a methyl iodine penetration of less than 15 percent for the SBVS charcoal adsorber, when tested in accordance with ASTM D3803-89 at conditions of 30° C and 95 percent RH. By applying a safety factor of 2, the licensee has reduced the filter efficiency to 70 percent as permitted.

The ABSVS is a standby ventilation system, common to the two PINGP units. The system serves the areas within the auxiliary building where there is potential for any inleakage from the shield building annulus. Such areas in the auxiliary building are designated as Zone SV. The ABSV system also consists of two redundant trains. Each train consists of a heater, pre-filter, absolute filter, a charcoal adsorber section, and an exhaust fan. The ABSV system is started automatically on a safety injection signal, a high radiation signal in the auxiliary building exhaust stack, or by manual initiation. Activation of the ABSV system automatically isolates the normal supply and exhaust ducts and the associated fans to the auxiliary building. In the re-evaluation of dose consequences, the ABSVS charcoal filter efficiency for elemental and organic iodines was changed from 90 percent to 70 percent, and the drawdown time prior to crediting the ABSVS was changed from 0 seconds to 6 minutes. As discussed above and supported by the PINGP TS section 5.5.9 "Ventilation Filter Testing Program (VFTP)", subsection c, the licensee has reduced the filter efficiency from 90 percent to 70 percent as permitted. The drawdown time of 6 minutes is acceptable based on the surveillance requirement (SR) 3.7.12.3 in PINGP TS 3.7.12, "Auxiliary Building Special Ventilation System (ABSVS)".

3.3.3 Control Room Special Ventilation System (CRSVS)

PINGP Unit 1 and Unit 2 are served by a single CR. The CRSVS consists of two independent, redundant trains that circulate and filter the CR air. The normal operating portion of the system is serviced by two ventilation intakes. The emergency operating portion of the system consists of a filter train and a cleanup fan. During normal operation, total outside air supplied to the CR is 2000 cfm. The outside air intake(s) close and the emergency filter train system starts in a filtered recirculation mode upon receipt of an SI signal or high radiation in the normal intake air duct. The dose consequence reanalyses assumed a delay of approximately 2 minutes in crediting the CR isolation and filtered recirculation. The delay accounts for the possibility of a Loss of Offsite Power and the time required for the emergency diesel generator to become fully operational (including sequencing delays), damper closure times and the time it takes for the emergency recirculation filter fans to come up to speed. The dose analysis revised the quantity of unfiltered ventilation air intake from 0 cfm to 2000 cfm in the first two minutes. This change is conservative, and therefore, is acceptable to the NRC staff. Additional changes to the current design basis input incorporated into the revised dose consequence analysis, and the NRC staff's review of the changes are provided below:

- The filter efficiencies of CRSVS were changed from 95 percent for all iodine species to 99 percent for particulate iodine, and 95 percent for elemental and organic iodine. The revised efficiencies are supported by the acceptance criteria for HEPA and charcoal adsorber testing in TS section 5.5.9, “Ventilation Filter Testing Program (VFTP),” and therefore, are acceptable to the NRC staff.

- The emergency filtered recirculation flow rate was changed from 3000 cfm to 3600 cfm. PINGP TS SR 3.7.10.4 requires periodic verification that CRSVS flow rate through the filters in recirculation mode is between 3600 cfm and 4400 cfm. As stated above, the acceptance criteria for testing the filter efficiency at this flow rate is in TS section 5.5.9, “Ventilation Filter Testing Program (VFTP).” Therefore, this change is acceptable to the NRC staff.

- Unfiltered in-leakage into the CR was revised from 44 cfm to 165 cfm based on the results of tracer gas tests. In addition, unfiltered in-leakage into the CR due to ingress/egress was changed from 0 cfm to 10 cfm. In response to the NRC staff's RAI, the licensee stated in a letter dated March 16, 2009, that the 165 cfm was based on tracer gas testing performed in July of 1998, and it includes an uncertainty of ± 5 cfm. The licensee further stated that tracer gas testing was again performed at PINGP during November and December of 2004, the results of which were bounded by the July 1998 testing. The NRC staff, therefore, finds the unfiltered air quantity used in the dose consequence reanalysis acceptable.

3.3.4 Containment Inservice Purge System (CIPS)

The CIPS is used to reduce the concentration of noble gases within containment prior to and during personnel access and to provide low volume normal purge and ventilation. The 18-inch purge supply and exhaust lines are provided with two containment automatic isolation valves. Containment ventilation instrumentation (CVI) close the containment isolation valves in the CIPS. CVI initiates on an SI signal, by manual actuation of containment isolation, or by manual actuation of containment spray (CS) system. Currently, blind flanges on these lines provide containment isolation function during Modes 1, 2, 3, and 4. The system can be used for ventilation of containment in Modes 1, 2, 3, and 4, provided the valves are leak tested, and the blind flanges removed and replaced by a spool piece. The TS requirements for the CIPS containment isolation valves and CVI instrumentation are contained in TS 3.6.3, “Containment Isolation Valves,” and TS 3.3.5, “Containment Isolation Instrumentation.” The licensee stated that the LOCA dose consequence reanalysis assumes that the CIPS is isolated. To assure that the release assumption of the analysis is met, the licensee is proposing to isolate the CIPS during modes 1, 2, 3, and 4, and eliminate the provisions in TS 3.6.3, which allow CIPS operations in these modes. The licensee is also proposing to eliminate TS 3.3.5, “Containment Ventilation Instrumentation,” in its entirety.

In response to the NRC staff's RAI, the licensee stated that no physical modifications to the CIPS valves or instrumentation are needed to maintain the plant in a safe configuration. Once the blind flanges are installed, the status of the valves and instrumentation is not addressed by the TS. The licensee further stated that, if the amendment request is approved, the plant may continue to maintain the availability of the CIPS isolation valves and instrumentation during Modes 5 and 6, including manual initiating functions, to provide defense-in-depth to minimize releases from a refueling outage event, even though there are no TS requirements to have CIPS actuation and isolation available during these modes. The only TS requirements that apply are during the movement of recently irradiated fuel assemblies in containment, as required by TS 3.9.4, "Containment Penetrations." TS 3.9.4.c describes acceptable means of isolation as, "... closed by a manual or automatic isolation valve, blind flange, or equivalent." As further explained in the TS 3.9.4 Bases Section for isolation, in MODE 6, the potential for containment pressurization as a result of an accident is low. Therefore, the requirements to isolate the containment from the outside atmosphere can be less stringent and the requirements are referred to as "containment closure" rather than "containment OPERABILITY." The Bases further state that, since there is no potential for containment pressurization, the Appendix J leakage criteria and test are not required.

The licensee is proposing to require the CIPS penetrations to be blind flanged at all times during plant operation in Modes 1, 2, 3, and 4, with the penetrations tested to meet containment leakage limits. The requirements provide the lowest functional capability of equipment required for safe operation of the facility and meet Criterion 3 of 10 CFR 50.36 as equipment that is part of the primary success path and which functions or actuates to mitigate a DBA. The requirements of 10 CFR 50.36, therefore, are satisfied. With the blind flanges installed and tested to meet containment leakage limits, the functional capability to isolate CIPS isolation valves during plant operating modes 1, 2, 3, and 4, provided by TS 3.3.5, "Containment Vent Isolation Instrumentation," is neither required nor does it meet any of the 10 CFR 50.36 criteria, and therefore, can be removed from PINGP TS. The requirements of GDC 56 are satisfied because the containment isolation function is now provided by the blind flanges. The CIPS isolation valves with resilient seals will not be relied upon to provide containment integrity, and therefore, the requirements of GDC 53 will continue to be met. In a letter dated March 16, 2009, the licensee confirmed that the LOCA and MSLB accident dose reanalysis will have no impact on post-accident containment pressure and temperature. Therefore, the requirements of GDC 50 are satisfied.

3.3.5 Containment Spray (CS) Initiation

In its June 26, 2008, application, the licensee stated that the CS initiation time was updated from 1 minute to 42 seconds in the LOCA dose consequence reanalysis. However, in its response to the NRC staff's RAI, the licensee determined that this change is non-conservative. In its letter dated March 16, 2009, the licensee stated that the time was selected from Table 14.6-2 of PINGP USAR, Rev. 29, and that, while it is conservative with respect to peak cladding temperature, core melting, and hydrogen generation, it is not conservative for dose analysis. The non-conservative design input was subsequently entered in the NSPM Corrective Action Process. The licensee stated that the containment pressure analysis in PINGP USAR Appendix K uses a maximum CS initiation delay time of 72 seconds. The licensee had re-run the LOCA dose analysis with the 72 second spray initiation time. The analysis resulted in LOCA doses slightly higher than those reported in the licensee's submittal dated June 26, 2008. However, the doses remained below the acceptance values for the CR, EAB, and LPZ.

3.3.6 Containment and Ventilation Systems Technical Specification Changes

3.3.6.1 Proposed Change to TS 3.6.3, "Containment Isolation Valves"

The licensee proposed the following revisions to TS 3.6.3:

- Note 1 under ACTIONS identifies exceptions to penetration flow path(s) that may be unisolated intermittently under administrative controls. The licensee proposed to add the 18-inch CIPS isolation valves to the exceptions.

- CONDITION D applies to leakage not within limits. The licensee proposed to delete the inservice purge valves from the applicability of CONDITION D.

- The licensee proposed to revise SR 3.6.3.2 from "Verify each 18-inch containment inservice purge penetration is blind flanged and meets SR 3.6.1.1," to "Verify each 18-inch containment inservice purge penetration blind flange is installed." Additionally, the FREQUENCY in SR 3.6.3.2 is proposed to be changed from "After each use of the 18-inch containment inservice purge system" to "Prior to entering MODE 4 from MODE 5."

  The change to delete the reference to SR 3.6.1.1 is acceptable to the NRC staff because the requirements for leak testing of the blind flanges are contained in TS section 5.5.14, "Containment Leakage Rate Testing Program." The change in FREQUENCY is acceptable, because the only time the blind flanges may be removed under the revised TS is in MODE 5 or MODE 6.

- The current SR 3.6.3.6 applies to leakage rate testing of the 18-inch CIPS valves. The licensee proposed to remove SR 3.6.3.6 and denote it as "Not Used."

The licensee is proposing to require the CIPS penetrations to be blind flanged at all times during plant operation in Modes 1, 2, 3, and 4, with the penetrations tested to meet containment leakage limits. The requirements provide the lowest functional capability of equipment required for safe operation of the facility and meet Criterion 3 of 10 CFR 50.36, as equipment that is part of the primary success path and which functions or actuates to mitigate a DBA. The requirements of 10 CFR 50.36 are therefore satisfied.

In addition, the requirements of GDC 56 are satisfied because the containment isolation function is now provided by the blind flanges. The CIPS isolation valves with resilient seals will not be relied upon to provide containment integrity, and therefore, the requirements of GDC 53 will continue to be met.

Therefore, the NRC staff finds that the proposed changes to TS 3.6.3 are acceptable.

3.3.6.2 Proposed Change to TS 3.3.5, "Containment Ventilation Isolation Instrumentation."

- The licensee proposed to delete TS 3.3.5 in its entirety and to denote it in the TSs as "Not Used."

With the blind flanges installed and tested to meet containment leakage limits, the functional capability to isolate CIPS isolation valves during plant operating Modes 1, 2, 3, and 4, provided by TS 3.3.5, "Containment Vent Isolation Instrumentation," is neither required nor does it meet any of the 10 CFR 50.36 criteria, and therefore, can be removed from PINGP TS.

Therefore, the NRC staff finds this change acceptable.

3.4 Summary of Regulatory and Technical Evaluation

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to reassess the radiological consequences of the postulated DBA with the proposed changes to the PINGP licensing and design basis. The NRC staff finds that the licensee will continue to meet the applicable dose acceptance criteria, as identified in Section 2.0 of this evaluation, following implementation of the proposed changes. The NRC staff further finds reasonable assurance that PINGP, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff finds the proposed license amendment and TS changes to be acceptable with respect to the radiological dose consequences of DBAs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (73 FR 52420). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: A. Boatright, NRR
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Date: June 19, 2009

Table 3.1

Prairie Island CR Atmospheric Dispersion Factors (χ/Q Values, sec/m^3)

Time Interval	0-2 hours	2-8 hours	4-hours	1-4 days	4-30 days
Shield Bldg Wall/CR Air Intake	2.36×10^{-3}	----	----	----	----
Shield Bldg Wall/CR Inleakage	1.45×10^{-3}	1.13×10^{-3}	4.94×10^{-4}	3.78×10^{-4}	3.05×10^{-4}
Shield Bldg Stack/CR Inleakage	2.39×10^{-3}	1.98×10^{-3}	8.74×10^{-4}	6.12×10^{-4}	4.71×10^{-4}
Aux. Bldg. MU Air Intake/CR Inleakage	2.19×10^{-3}	1.89×10^{-3}	8.49×10^{-4}	5.95×10^{-4}	5.28×10^{-4}
SG PORVs & MSL Break Point/CR Air Intake	3.07×10^{-2}	----	----	----	----
SG PORVs & MSL Break-Point/CR Inleakage	5.01×10^{-3}	4.09×10^{-3}	----	----	----

Note: Puff release of secondary system TS activity from the faulted steam generator following a MSLB assumed no credit for atmospheric dispersion.

Table 3.2

Prairie Island EAB and LPZ Atmospheric Dispersion Factors (X/Q Values, sec/m^3)

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

June 19, 2009

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO LOSS-OF-COOLANT (LOCA) ACCIDENT AND MAIN STEAM LINE BREAK (MSLB) ACCIDENT RADIOLOGICAL DOSE CONSEQUENCES ANALYSES AND AFFECTED TECHNICAL SPECIFICATIONS (TAC NOS. MD9140 AND MD9141)

Dear Mr. Wadley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. DPR-42 and Amendment No. 180 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 26, 2008, as supplemented by letters dated March 16 and May 1, 2009.

The amendments revise the Facility Operating Licenses by revising the licensing basis LOCA and MSLB accident radiological dose consequences as currently described in the PINGP Updated Safety Analysis Report Sections 14.5 and 14.9. The amendments also revise PINGP TSs 3.3.5, "Containment Ventilation Isolation Instrumentation", 3.4.17, "RCS Specific Activity", and 3.6.3, "Containment Isolation Valves", which are necessary to implement the revised analyses.

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

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

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 191 to DPR-42
2. Amendment No. 180 to DPR-60
3. Safety Evaluation

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