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Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota

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U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Corrections to Emergency Core Cooling System (ECCS) Evaluation Models

Enclosed please find enclosure 1, "Westinghouse loss of coolant accident (LOCA) Evaluation Model Changes," which is the 2008 annual report of corrections to the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 ECCS evaluation models. This report is submitted in accordance with the provisions of 10 CFR 50.46 and summarizes changes made to both the large break LOCA (LBLOCA) and small break LOCA (SBLOCA) analyses.

The SBLOCA and LBLOCA peak clad temperature (PCT) assessment sheets for Unit 1 and Unit 2 are enclosed as enclosure 2. The limiting LOCA analysis PCT for PINGP Unit 1 and Unit 2, with consideration of all 10 CFR 50.46 assessments, remains the LBLOCA analysis as summarized in enclosure 2.

Neither enclosure 1 nor enclosure 2 need be withheld from public disclosure.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE 1

Westinghouse LOCA Evaluation Model Changes

6 Pages follow

ERRORS IN REACTOR VESSEL LOWER PLENUM SURFACE AREA CALCULATIONS (Non-Discretionary Change)

Background
Two errors were discovered in the calculations of reactor vessel lower plenum surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The differences in vessel lower plenum surface area are relatively minor and would be expected to produce a negligible effect on large and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

DISCREPANCY IN METAL MASSES USED FROM DRAWINGS
(Non-Discretionary Change)

Background
Discrepancies were discovered in the use of metal masses from drawings. The updated reactor vessel metal masses and fluid volumes have been evaluated for impact on current licensing-basis analysis results and will be incorporated on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The differences in the reactor vessel metal mass and fluid volume are relatively minor and would be expected to produce a negligible effect on large and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

GENERAL CODE MAINTENANCE
(Discretionary Change)

Background
Various changes have been made to enhance the usability of the codes and to help preclude errors in analyses. This includes items such as modifying input variable definitions, units, and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and, eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The nature of these changes leads to an estimated PCT impact of 0°F.

GENERAL CODE MAINTENANCE
(Discretionary Change)

Background
A number of coding changes were made as part of normal code maintenance. Examples include additional information in code outputs, improved automation and diagnostics in the codes, increased code dimensions, and general code cleanup. All of these changes are considered to be Discretionary changes in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The nature of these changes leads to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

HOTSPOT BURST TEMPERATURE LOGIC ERRORS
(Non-Discretionary Change)

Background
The HOTSPOT code has been updated to incorporate the following corrections to the burst temperature logic: (1) change the rod internal pressure used to calculate the cladding engineering hoop stress from the value in the previous time step to the value in the current time step; (2) revise the average cladding heat-up rate calculation to reset selected variables to zero at the beginning of each trial and use the instantaneous heat-up rate when fewer than five values are available; and, (3) reflect the assumed saturation of ramp rate effects above 28°C/s for Zircaloy-4 cladding from Equation 7-66 of Reference 1. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
Sample calculations for each change showed no effect on peak cladding temperature, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes.

Reference(s)
1. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," S. M. Bajorek et al., March 1998.

MODELING ERRORS IN ASTRUM ANALYSES
(Non-Discretionary Change)

Background

Several small modeling errors in the Prairie Island plant models were identified during the course of a recent Best Estimate Large Break LOCA analysis. These errors include a small difference in modeled accumulator line losses, the splitting of core average assembly rods into average core channels, and a discrepancy in the volume modeled in the CCFL inner global channel. The impact of each of these modeling errors was assessed individually on a plant-specific basis. These changes represent Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451, and are only applicable to the 2006 OFA ASTRUM analyses.

Affected Evaluation Model(s)

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Each of these issues was individually evaluated to have a negligible impact on the PCT. Therefore, a 0°F peak cladding temperature (PCT) impact is assigned to each issue for 10 CFR 50.46 reporting purposes.

ENCLOSURE 2

LBLOCA and SBLOCA Peak Clad Temperature Assessment Sheets

4 pages follow

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Plant Name: Prairie Island Unit 1
Utility Name: Nuclear Management Company, LLC
Revision Date: 1 /27/09

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	7/1/06	**Limiting Break Size:**	Split
FQ:	2.5	**FdH:**	1.77		
Fuel:	OFA	**SGTP (%):**	10		
Notes:					

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1594	1	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . 2 Reconstituted Rods Evaluation	0		
C. 2008 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS	PCT = 1594		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . WCAP-16584-P, "Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for the Prairie Island Nuclear Plant Unit 1 Using ASTRUM Methodology," 7/2006.

Notes:

(a) None

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Small Break

Plant Name: Prairie Island Unit 1
Utility Name: Nuclear Management Company, LLC
Revision Date: 1 /27/09

Analysis Information

EM:	NOTRUMP	**Analysis Date:**	11/21/03	**Limiting Break Size:**	6 inch
FQ:	2.8	**FdH:**	2		
Fuel:	OFA	**SGTP (%):**	10		
Notes:	Zirlo™ (14X14), Framatome RSG				

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1409	1,2,3	(a)
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2008 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS PCT =	1409		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . NSP-04-10 "Safety Analysis Transition Program Transmittal of Engineering Report," February 20, 2004.

2 . WCAP-16206-P, "Safety Analysis Transition Program Engineering Report for the Prairie Island Nuclear Power Plant, Volume 1 Engineering Analyses," February 2004.

3 . OC-PX-2004.009, "SBLOCA Analysis Loop Seal Restriction Option," Mercier to Brown, March 5, 2004.

Notes:

(a) The 6-inch break is limiting when the loop seal restriction is applied to all break sizes.

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Plant Name: Prairie Island Unit 2
Utility Name: Nuclear Management Company, LLC
Revision Date: 1 /27/09

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	6/1/06	**Limiting Break Size:**	Split
FQ:	2.5	**FdH:**	1.77		
Fuel:	OFA	**SGTP (%):**	25		
Notes:					

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1546	1	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2008 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1546		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . WCAP-16508-P, "Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for the Prairie Island Nuclear Plant Unit 2 Using ASTRUM Methodology," 6/2006

Notes:

(a) None

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Small Break

Plant Name: Prairie Island Unit 2
Utility Name: Nuclear Management Company, LLC
Revision Date: 1 /27/09

Analysis Information

EM:	NOTRUMP	**Analysis Date:**	9/1/00	**Limiting Break Size:**	3 inch
FQ:	2.8	**FdH:**	2		
Fuel:	OFA	**SGTP (%):**	25		
Notes:	Zirlo™ (14X14)				

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1142**	**1**	**(a)**
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . No Items for 2000, 2001 & 2002 Reports	0	2,4,5	
2 . NOTRUMP Bubble Rise / Drift Flux Model Inconsistency Corrections	35	6,7	
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2008 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . Evaluation for Reduced Auxilary Feedwater Flow Rate	0	3	
LICENSING BASIS PCT + PCT ASSESSMENTS PCT =	1177		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . NSP-00-045, "SBLOCA Re-analysis with Revised NOTRUMP Code," October 2, 2000.

2 . NSP-01-006, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2000," March 6, 2001.

3 . NSP-02-36, "SBLOCA Limited FSAR Update and Evaluation for Revised Auxiliary Feedwater Flow Rate," October 2002.

4 . NSP-02-5, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2001," March 2002.

5 . NSP-03-19, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2002," March 2003.

6 . NSP-03-68, "10 CFR 50.46 Mid-Year Notification and Reporting for 2003," November 2003.

7 . NSP-03-38, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.

Notes:

(a) Accumulator water volume sensitivity of +/- 30 cubic feet included.