NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

DEC 1 4 2004

L-PI-04-131
10 CFR 50.46

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Prairie Island Nuclear Generating Plant Unit 1
Docket 50-282
License No. DPR-42

Corrections to Emergency Core Cooling System (ECCS) Evaluation Models

Reference: Letter L-PI-04-111, dated September 24, 2004, "Clarification of Actions for Corrections to Emergency Core Cooling System (ECCS) Evaluation Models," from Nuclear Management Company, LLC (NMC), to the Nuclear Regulatory Commission

Attached is a report of changes to the Prairie Island Nuclear Generating Plant (PINGP) Emergency Core Cooling System (ECCS) Evaluation Models. This report is being submitted in accordance with the provisions of 10 CFR 50, Section 50.46, as a 30-day report.

The report includes Large Break Loss of Coolant Accident (LBLOCA) and Small Break Loss of Coolant Accident (SBLOCA) changes reported by Westinghouse that are applicable from the beginning of Fuel Cycle 23.

The LBLOCA Peak Clad Temperature (PCT) changes are due to a sensitivity study performed for new steam generators (installed during the refueling outage that ended November 23, 2004). The change since the Last Acceptable Model submitted to the NRC on August 5, 2004 is +32 °F. The PCT (2026 °F, see Attachment 1) for the LBLOCA analysis continues to remain below the 10 CFR 50.46 PCT acceptance criterion. However, the accumulated absolute value of the PCT changes and errors since the original 1995 baseline Analysis of Record is 782 °F. NMC has committed by letter (see Reference) to provide a new LBLOCA analysis for PINGP by March 31, 2006.

The SBLOCA PCT changes are due to a reanalysis to address the installation of the new steam generators. The change since the Last Acceptable Model submitted to the NRC on August 5, 2004 is +232 °F. The PCT (1409 °F, see Attachment 2) for the SBLOCA analysis continues to remain below the 10 CFR 50.46 PCT acceptance criterion. The accumulated absolute value of the PCT changes and errors since the

A001

original baseline Analysis of Record is 267 °F. Since the current analysis was a full scope reanalysis, no schedule for reanalysis is necessary.

The summary sheets attached to this letter need not be withheld from public disclosure.

Summary of Commitments

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

Please contact Jack Leveille (651-388-1121, Ext 4142) if you have any questions related to this letter.

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector – Prairie Island Nuclear Generating Plant

Enclosure

ENCLOSURE

LOCA Peak Clad Temperature Summary
Prairie Island Nuclear Generating Plant

(includes plant specific changes and non-zero non-plant specific changes)

4 Pages Follow

(Attachment 1 - Prairie Island Unit 1 LBLOCA – 3 pages)
(Attachment 2 - Prairie Island Unit 1 SBLOCA – 1 page)

ATTACHMENT 1
Page 1 of 3

Westinghouse LOCA Peak Clad Temperature Summary for SECY UPI Large Break

Plant Name: Prairie Island Unit 1
Utility Name: Nuclear Management Company, LLC
Revision Date: 8 /10/04

Analysis Information

EM:	SECY UPI WC/T	**Analysis Date:**	3/1/95	**Limiting Break Size:**	Cd = 0.4
FQ:	2.4	**FdH:**	1.77		
Fuel:	OFA	**SGTP (%):**	15		

Notes: Zirlo™, OSG SGTP Evaluated up to 24.64% (see also Note f); Fq increased to 2.5 (Item A.10); RSG Study at 10% SGTP.

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	2180	1,2	(a)
MARGIN ALLOCATIONS (Delta PCT)			
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS			
1. Fixed Heat Transfer Node Assignment Error/Accumulator Water Injection Error (1995 Report)	-175	3	
2. 1-D Transition Boiling Heat Transfer Error (1997 Report)	59	5	
3. Vessel Channel DX Error (1997 Report)	-14	5	
4. Input Consistency (1997 Report)	-66	5	
5. No Items for 1996, & 1998 Reports	0	4,6	
6. Accumulator Line/Pressurizer Surge Line Data / Plant Specific Accumulator Level & Line Volume / Plant Specific Restart Error: Reanalysis (1999 Report)	113	7	(b)
7. Modeling Updates and Unheated Conductor Input Corrections (Plant Specific, 2000 Report)	-147	8,10	(c)
8. Accumulator Pressure +/- 30 psi Range (Plant Specific, 2001 Report)	8	12, 13	(d)
9. LHSI Error Evaluation (Plant Specific, 2002 Report)	30	14, 15	(h)
10. Sensitivity Study for FQ=2.5, LHSI Correction, etc. (as listed in note (g)) (Plant Specific, 2003 Report)	-47	17,19,20	(g,i)
11. Broken Loop Nozzle Correction (Plant Specific) (2003 Report)	-19	20,22	(i)
B. PLANNED PLANT CHANGE EVALUATIONS			
1. Sensitivity Study for Steam Generator Tube Plugging Increase to 25%	52	8	
2. Accumulator Water Volume +/- 25 ft3 Range	12	12	
3. Accumulator Pressure Extended to +/- 55 psi Range	21	12	
4. 5 Reconstituted Rods Evaluation	0	9,11	(e)
5. SATP Core Average Burnup	17	21,23	
6. Sensitivity Study for Framatome RSG	32	24	
C. 2004 PERMANENT ECCS MODEL ASSESSMENTS			
1. None	0		
D. TEMPORARY ECCS MODEL ISSUES*			
1. None	0		

ATTACHMENT 1
Page 2 of 3.

Westinghouse LOCA Peak Clad Temperature Summary for SECY UPI Large Break

Plant Name:	Prairie Island Unit 1
Utility Name:	Nuclear Management Company, LLC
Revision Date:	8/10/04

E. OTHER			
1. Removal of Reference 14 LHSI Error Evaluation	-30	17	(h)

LICENSING BASIS PCT + MARGIN ALLOCATIONS	**PCT =**	2026

- It is recommended that these temporary PCT allocations which address current LOCA model issues not be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1. 95NS-G-0021, "Updated UPI LBLOCA," March 24, 1995.
2. WCAP-13919, Addendum 1, "Prairie Island Units 1 and 2 WCOBRA/TRAC Best Estimate UPI Large Break LOCA Analysis Engineering Report Addendum 1: Updated Results," December 1996.
3. NSP-96-202, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting," February 20, 1996.
4. NSP-97-201, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting," April 17, 1997.
5. NSP-98-012, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 1997," February 27, 1998.
6. NSP-99-010, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 1998," April 29, 1999.
7. NSP-00-005, "Northern States Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 1999," February 2000.
8. NSP-00-057, "Northern States Power Company Prairie Island Units 1 and 2 LOCA Evaluation of 25% SGTP with Other Modeling Updates," December 11, 2000.
9. 00NS-G-0076/CAB-00-390, "Prairie Island Unit 1 Cycle 21 LOCA Reload Confirmation and FCEP Checklist," December 15, 2000.
10. 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.
11. Rothrock (NMC) to Swigat (W), "Prairie Island Unit 1 LOCA PCT," May 30, 2001.
12. NSP-02-9, "Nuclear Management Company Prairie Island Units 1 and 2 LBLOCA Accumulator Pressure and Volume Ranges Evaluation," February 15, 2002.
13. NSP-02-5, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2001," March 2002.
14. NSP-02-59/LTR-ESI-02-194, "Final Evaluation of Large Break LOCA Error," December 2002.
15. NSP-03-19, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2002," March 2003.
16. MP92-TAH-0394 / ET-NSL-OPL-I-92-518, "NSPC Prairie Island Units 1 and 2, SG Tube Flow Area Reduction under LOCA / SSE - Final Report", October 21, 1992.
17. NSP-04-10 "Safety Analysis Transition Program Transmittal of Engineering Report," February 20, 2004.
18. NSP-93-513, Rev 1/ET-NSL-OPL-I-93-313, Rev. 1, Letter from T. A. Hawley (W) to K. E. Higar (NSP), "Final Transmittal of Assumptions to be used for the Large and Small Break LOCA Analyses, Rev. 1", July 7, 1993. Confirmed by : Letter from K. E. Higar (NSP) to Mr. T. Hawley (W), "Acceptance of NSP-93-513, Rev. 1", July 30, 1993.
19. NSP-04-38, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.
20. WCAP-16206-P, "SATP Engineering Report for Prairie Island," February 2004.
21. NF-NMC-04-49, "Nuclear Management Company Prairie Island Unit 1 Cycle 22 Final RSE," April 2004.

Westinghouse LOCA Peak Clad Temperature Summary for SECY UPI Large Break

Plant Name: Prairie Island Unit 1

Utility Name: Nuclear Management Company, LLC

Revision Date: 8 /10/04

22 . NSP-04-65, "Nuclear Management Company Prairie Island Units 1 & 2 Safety Analysis Transition Program Repsonse to 10 CFR 50.46 Inquiry," April 21, 2004.

23 . NF-NMC-04-xxx, Unit 1, Cycle 23 RSE, Aug. 2004.

24 . NSP-04-114, "Nuclear Management Company Prairie Island Units 1 & 2, Safety Analysis Transition Program, Transmittal of LBLOCA Replacement Steam Generator (RSG) Engineering Report Addendum," (WCAP-16206-P-Addendum 1), June 2004.

Notes:

(a) P-bar-HA increased from 1.57 to 1.59

(b) Reanalysis for all listed issues

(c) Reanalysis for both issues

(d) Related JCO in existence (NSP-01-030). NMC cognizant of uncertainty application and PCT sheet categorization.

(e) Reconstitution for Cycle 21 recanted per Reference 11.

(f) It is assumed that NMC is applying the 0.36% SGTP allowance factor branch of the SG LOCA / SSE issue (Reference 16). Thus the 25% SGTP Study (Item B.1) supports a net SGTP limit of 24.64%.

(g) Sensitivity Study for: FQ=2.50, PAD 4.0 Implementation, Restoration of LHSI to Reference 18 values, SG/Loop ΔP Re-tuning, Core Power Restoration.

(h) The note (g) sensitivity study allows for the removal of the Reference 14 engineering assessment.

(i) Items A.10 and A.11 presented as aggregate -66 °F entry prior to Reference 22 decomposition.

Westinghouse LOCA Peak Clad Temperature Summary for Small Break

Plant Name: Prairie Island Unit 1

Future-B

Utility Name: Nuclear Management Company, LLC
Revision Date: 3/3/04

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	(a)
MARGIN ALLOCATIONS (Delta PCT)			
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT CHANGE EVALUATIONS			
1 . None	0		
C. 2003 PERMANENT ECCS MODEL ASSESSMENTS			
1 . None	0		
D. TEMPORARY ECCS MODEL ISSUES*			
1 . None	0		
E. OTHER			
1 . None	0		
LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 1409		

* It is recommended that these temporary PCT allocations which address current LOCA model issues not 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.

Notes:

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