NMC
Committed to Nuclear Excellence

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

JUN 09 2006

L-PI-06-052
10 CFR 50.46

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

References: (1) Letter L-PI-05-079, dated August 30, 2005, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Corrections to Emergency Core Cooling System (ECCS) Evaluation Models," from NMC to NRC.

Enclosed please find Attachment 1, "Westinghouse LOCA (loss of coolant accident) Evaluation Model Changes," which is the 2005 annual report of corrections to the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 ECCS Evaluation Models. A number of large break LOCA (LBLOCA) analyses assessments for 2005 were previously reported in Reference 1. This report is submitted in accordance with the provisions of 10 CFR 50, Section 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 Attachment 2. The limiting LOCA analysis for Prairie Island Unit 1 and Unit 2, with consideration of all 10 CFR 50.46 assessments, remains the LBLOCA analysis, as summarized in Attachment 2.

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

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Summary of Commitments

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

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (2)

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

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET NOS 50-282 AND 50-306

Westinghouse LOCA Evaluation Model Changes

6 Pages follow

PRESSURIZER FLUID VOLUMES
(Non-Discretionary Change)

Background
The Westinghouse Systems and Equipment Engineering group has recommended that the previously transmitted pressurizer fluid volumes be replaced with nominal cold values. This change resolves a discrepancy in the prior calculations while providing a close approximation of the actual as-built values. The revised values have been evaluated for impact on current licensing basis analyses and will be incorporated into the plant-specific input databases on a forward-fit basis. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model
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 differences between the previously transmitted and revised volumes are very small and would be expected to produce a negligible effect on large and small break LOCA analysis results, leading to an estimated zero degree PCT impact.

CONTAINMENT RELATIVE HUMIDITY ASSUMPTION
(Non-Discretionary Change)

Background
Large Break LOCA analyses have historically used maximum initial relative humidity to specify the initial containment air and steam partial pressures. This assumption is conservative for a given total initial containment pressure, but is non-conservative for a given initial containment air partial pressure. The historical assumption has been revised accordingly. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model
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
SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

Estimated Effect
An evaluation for the plants within Westinghouse Pittsburgh large break LOCA analysis cognizance concluded that no PCT assessments are required, leading to an estimated PCT effect of 0°F.

PRESSURIZER FLUID VOLUMES
(Non-Discretionary Change)

Background
The Westinghouse Systems and Equipment Engineering group has recommended that the previously-transmitted pressurizer fluid volumes be replaced with nominal cold values. This change resolves a discrepancy in the prior calculations while providing a close approximation of the actual as-built values. The revised values have been evaluated for impact on current licensing-basis analyses and will be incorporated into the plant-specific input databases on a forward-fit basis. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The differences between the previously-transmitted and revised volumes are very small 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.

LOWER GUIDE TUBE ASSEMBLY WEIGHT
(Non-Discretionary Change)

Background
An error was discovered in the lower guide tube assembly weight for three units that resulted in a small over-estimation of the upper plenum metal mass. The corrected values have been evaluated for impact on current licensing-basis analyses and will be incorporated into the plant-specific input databases on a forward-fit basis. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The differences in upper plenum metal mass are very small 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 NOTRUMP RWST DRAINDOWN CALCULATION
(Non-Discretionary Change)

Background
For small break LOCA calculations where the break size is greater than the safety injection (SI) line diameter, and where the SI line is connected directly to the reactor coolant system (RCS), it is assumed that the broken loop safety injection flows do not inject to the RCS, but rather spill to containment. Typically, this is modeled in NOTRUMP-EM analyses by setting the flows injected to the broken loop equal to zero, which neglects the continued depletion of the refueling water storage tank (RWST) inventory. As a result, the RWST draindown time is incorrectly calculated, potentially resulting in an inaccurate modeling of enthalpy changes and/or SI interruptions that can occur at switchover to sump recirculation. Therefore, the SI spilling flows need to be explicitly modeled in order to correctly calculate the RWST draindown time.

Affected Evaluation Models
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
For Westinghouse plants using the NOTRUMP-EM, the larger small breaks are typically non-limiting and the transients are of short duration. Therefore, correct modeling of the spilling flows in the RWST draindown calculation for these breaks would be expected to produce a negligible effect on SBLOCA results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

GENERAL CODE MAINTENANCE
(Enhancements/Forward-Fit Discretionary Change)

Background
Various changes in code input and output format have been made to enhance usability and help preclude errors in analyses. This includes both input changes (e.g., more relevant input variables defined and more common input values used as defaults) and input diagnostics designed to preclude unreasonable values from being used, as well as various changes to code output which have no effect on calculated results. In addition, various updates were made to eliminate inactive coding, improve active coding, and enhance commenting, both for enhanced usability and to facilitate code debugging when necessary. 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 Models
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.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET NOS 50-282 AND 50-306

LBLOCA and SBLOCA Peak Clad Temperature Assessment Sheets

8 pages follow

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Large Break

Plant Name: Prairie Island Unit 1
Utility Name: Nuclear Management Company, LLC
Revision Date: 2 /23/06

Analysis Information

EM:	SECY UPI	**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)**
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR 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 Loss Coefficient (Plant Specific)	-19	19,20,22, 26	(i)
B. PLANNED PLANT MODIFICATION 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 Replacement Steam Generators	32	24	
7 . HAUP LOCA Evaluation	3	25	(j)
C. 2005 ECCS MODEL ASSESSMENTS			
1 . SECY Cold Leg Nozzle Expansion	13	26	

4/24/06

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Large Break

Plant Name: Prairie Island Unit 1
Utility Name: Nuclear Management Company, LLC
Revision Date: 2 /23/06

D. OTHER*

1 . Removal of Reference 14 LHSI Error Evaluation	-30	17	(h)
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	~~2042~~ 2039	(j)	

jcd 4/24/06

* It is recommended that the licensee determine if these PCT allocations 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 Appendix K Large Break

Plant Name:	Prairie Island Unit 1
Utility Name:	Nuclear Management Company, LLC
Revision Date:	2 /23/06

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-04-129, "Nuclear Management Company Prairie Island Unit 1, Cycle 23 Final RSE," August 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.

25 . NSP-05-155, "Nuclear Management Company, Reactor Vessel Head Replacement Project, Prairie Island Units 1 & 2," May 18, 2005.

26 . NSP-05-191, "Miscellaneous LBLOCA SECY EM Error Notification," August 2005.

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.

(J) The 3°F penalty for the Head Assembly Upgrade Project (HAUP) will not be effective until the new head assembly is installed in the Spring of 2006. Therefore, this 3°F penalty will not be included in this summary of changes and errors for the 2005 annual report.

[signature]
4/24/06

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: 2 /23/06

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. 2005 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 Appendix K Large Break

Plant Name: Prairie Island Unit 2
Utility Name: Nuclear Management Company, LLC
Revision Date: 2 /23/06

Analysis Information

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

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

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	2180	1,2	(a)
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR 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 & 2004 Reports	0	4,6,23	
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	8,9	(c)
8 . Accumulator Pressure +/- 30 psi Range (Plant Specific) (2001 Report)	8	10,11	(d)
9 . LHSI Error Evaluation (Plant Specific) (2002 Report)	30	12,13	(g)
10 . Sensitivity Study for FQ=2.5, LHSI Correction, etc. (as listed in note (f)) (Plant Specific) (2003 Report)	-47	15,17,18	(f,h)
11 . Broken Loop Nozzle Loss Coefficient (Plant Specific)	-19	17,18,20, 24	(h)
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . Sensitivity Study for Steam Generator Tube Plugging Increase to 25%	52	8	
2 . Accumulator Water Volume +/- 25 ft3 Range	12	10	
3 . Accumulator Pressure Extended to +/- 55 psi Range	21	10	
4 . Cycle 22 SATP Core Average Burnup	17	19	
5 . HAUP LOCA Evaluation	3	21	
6 . SATP Core Average Burnup Extension for Cycle 23 Redesign	7	22	
C. 2005 ECCS MODEL ASSESSMENTS			
1 . SECY Cold Leg Nozzle Expansion	13	24	
D. OTHER*			
1 . Removal of Reference 12 LHSI Error Evaluation	-30	15	(g)

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Large Break

Plant Name: Prairie Island Unit 2

Utility Name: Nuclear Management Company, LLC

Revision Date: 2 /23/06

LICENSING BASIS PCT + PCT ASSESSMENTS	**PCT =**	2017

* It is recommended that the licensee determine if these PCT allocations 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. 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.
10. NSP-02-9, "Nuclear Management Company Prairie Island Units 1 and 2 LBLOCA Accumulator Pressure and Volume Ranges Evaluation," February 15, 2002.
11. NSP-02-5, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2001," March 2002.
12. NSP-02-59/LTR-ESI-02-194, "Final Evaluation of Large Break LOCA Error," December 2002.
13. NSP-03-19, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2002," March 2003.
14. 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.
15. NSP-04-10 "Safety Analysis Transition Program Transmittal of Engineering Report," February 20, 2004.
16. 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.
17. NSP-04-38, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.
18. WCAP-16206-P, "SATP Engineering Report for Prairie Island," February 2004.
19. NF-NMC-04-50, "Nuclear Management Company Prairie Island Unit 2 Cycle 22 Final RSE," April 2004.
20. 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.
21. NSP-05-155, "Nuclear Management Company, Reactor Vessel Head Replacement Project, Prairie Island Units 1 & 2," May 18, 2005.
22. NF-NMC-05-38 Rev. 1, "Prairie Island Unit 2 Cycle 23 Final RSE," May 13, 2005
23. NSP-05-65, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2004," April 2005.

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Large Break

Plant Name:	Prairie Island Unit 2
Utility Name:	Nuclear Management Company, LLC
Revision Date:	2 /23/06

24 . NSP-05-191, "Miscellaneous LBLOCA SECY EM Error Notification," August 2005.

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) It is assumed that NMC is applying the 0.36% SGTP allowance factor branch of the SG LOCA / SSE issue (Reference 14). Thus the 25% SGTP Study (Item B.1) supports a net SGTP limit of 24.64%.

(f) Sensitivity Study for: FQ=2.50, PAD 4.0 Implementation, Restoration of LHSI to Reference 16 values, SG/Loop ΔP Re-tuning, Core Power Restoration.

(g) The note (f) sensitivity study allows for the removal of the Reference 12 engineering assessment.

(h) Items A.10 and A.11 presented as aggregate -66 °F entry prior to Reference 20 decomposition.

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: 2 /23/06

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. 2005 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 Auxilary 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.