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

JUN 26 2008

L-PI-08-057
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

Enclosed please find Attachment 1, "Westinghouse LOCA [loss of coolant accident] Evaluation Model Changes," which is the 2007 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 Attachment 2. The limiting LOCA analysis for PINGP 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.

Summary of Commitments

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

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Attachments (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

8 Pages follow

PBOT AND PMID SAMPLING
(Discretionary Change)

Background
The portrayal of the PBOT and PMID sampling in the ASTRUM topical (Reference 1) as uniform was slightly misleading in relationship to the actual sampling algorithm used for analyses, which is the same as approved for the CQD (Reference 2). As such, the sampling algorithm for ASTRUM was modified to coincide explicitly with the depiction in the topical. This change represents a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
This change is considered a forward-fit improvement, and as such has no impact on existing analyses.

Reference(s)
1. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

2. WCAP-12945-P- A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," 1998.

HOTSPOT FUEL RELOCATION
(Non-Discretionary Change)

Background

In the axial node where burst is predicted to occur, a fuel relocation model in HOTSPOT is used to account for the likelihood that additional fuel pellet fragments above that elevation may settle into the burst region. It was discovered that the effect of fuel relocation on local linear heat rate was being calculated, but then cancelled out later in the coding. This change represents a Non-Discretionary Change 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

1996 and 1999 BELOCA EMs analyses were assessed on a plant-specific basis, via the HOTSPOT reanalysis of a representative WCOBRA/TRAC case using the corrected code version at the burst elevation/burst model enabled sub-case. The HOTSPOT 95% probability PCT results were used to establish the plant-specific PCT penalty.

2004 ASTRUM EM analyses were assessed on a plant-specific basis, via the reanalysis of all of the burst cases from the original HOTSPOT calculations using the corrected HOTSPOT code version.

Plant-Specific Text

Based on the plant-specific calculations, the estimated effect of this error correction is 0°F.

STEAM GENERATOR NOZZLE VOLUME ACCOUNTING ERROR
(Non-Discretionary Change)

Background
It was discovered that many plant-specific WCOBRA/TRAC calculations shared a common error of double accounting of the volume of one or both SG Plenum Nozzles. The extent of over-accounting is plant-specific but would be in the vicinity of 7-9 ft^3 per nozzle. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
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
RCS Loop inventory does not significantly contribute to core cooling during blowdown since most of the fluid in both the intact and broken RCS loops will exit the break without entering the core, making RCS Loop volume a tertiary player in system behavior. A small volume error of this nature is anticipated to be negligible throughout the transient, such that an estimated effect of 0°F is assigned for 10 CFR 50.46 reporting purposes.

ERRORS IN REACTOR VESSEL NOZZLE DATA COLLECTIONS
(Non-Discretionary Change)

Background
Some minor errors were discovered in the reactor vessel nozzle data collections that potentially affect the vessel inlet and outlet nozzle fluid volume, metal mass, and surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated into the plant-specific input databases 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)
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
These errors were evaluated to have a negligible impact on the Large Break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

NOTRUMP-EM REFINED BREAK SPECTRUM
(Non-Discretionary Change)

Background
During the course of reviewing several extended power uprate and replacement steam generator Small Break LOCA (SBLOCA) analyses, the Nuclear Regulatory Commission (NRC) questioned the break spectrum analyzed in the NOTRUMP evaluation model (EM). The NRC was concerned that the resolution of the break spectrum used in the NOTRUMP EM (1.5, 2, 3, 4, and 6 inch cases) may not be fine enough to capture the worst break with regard to limiting peak clad temperature as per 10 CFR 50.46. That is, the plant could be SBLOCA limited with regard to overall LOCA results.

In response to this, Westinghouse performed some preliminary work indicating that in some cases more limiting results could be obtained from non-integer break sizes; however, the magnitude of the impact was far less than that shown in preliminary work performed by the NRC. Based on this, Westinghouse performed evaluations to determine if all currently operating plants would maintain compliance with the 10 CFR 50.46 acceptance criteria when considering a refined SBLOCA break spectrum. It should be noted that use of a refined break spectrum is not an error, but a change, since evaluating only integer break sizes has been the standard practice since the initial licensing of NOTRUMP.

This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

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

Estimated Effect
Consistent with the method described in Reference 1, for plants with low SBLOCA peak cladding temperatures (PCTs) (i.e., less than 1700°F) and overall SBLOCA results that are significantly non-limiting when compared with large break LOCA (LBLOCA) results, no explicit refined break spectrum calculations were performed, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes. For plants with high SBLOCA PCTs (i.e., equal to or greater than 1700°F), explicit refined break spectrum calculations were performed, and PCT penalties were assessed, if necessary.

Reference(s)
1. LTR-NRC-06-44, "Transmittal of LTR-NRC-06-44 NP-Attachment, 'Response to NRC Request for Additional Information on the Analyzed Break Spectrum for the Small Break Loss of Coolant Accident (SBLOCA) NOTRUMP Evaluation Model (NOTRUMP EM), Revision 1,' (Non-Proprietary)," July 14, 2006.

ERRORS IN REACTOR VESSEL NOZZLE DATA COLLECTIONS
(Non-Discretionary Change)

Background
Some minor errors were discovered in the reactor vessel nozzle data collections that potentially affect the vessel inlet and outlet nozzle fluid volume, metal mass and surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated into the plant-specific input databases 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 vessel inlet and outlet nozzle fluid volume, metal mass and surface area are relatively minor and would be expected to produce a negligible effect on large break and small break LOCA analysis results, leading to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

PUMP WEIR RESISTANCE MODELING
(Non-Discretionary Change)

Background
Review of the reactor coolant pump data collections identified instances of either including a weir resistance for a design without a weir or double-counting the weir resistance for a design with a weir. The corrected resistances have been evaluated for impact on existing analysis results 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 Model(s)
1981 Westinghouse Large Break LOCA Evaluation Model with BASH
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
Resolving the identified discrepancies has been evaluated as having a negligible effect on existing 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.

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

6 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: 1 /15/08

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 e); 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	11,12	(d)
9 . LHSI Error Evaluation (Plant Specific, 2002 Report)	30	13,14	(g)
10 . Sensitivity Study for FQ=2.5, LHSI Correction, etc. (as listed in note (f)) (Plant Specific, 2003 Report)	-47	16,18,19	(f,h)
11 . Broken Loop Nozzle Loss Coefficient (Plant Specific)	-19	18,19,21, 25	(h)
12 . SECY Cold Leg Nozzle Expansion	13	25	
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	11	
3 . Accumulator Pressure Extended to +/- 55 psi Range	21	11	
4 . 2 Reconstituted Rods Evaluation	1	9	
5 . SATP Core Average Burnup	17	20,22	
6 . Sensitivity Study for Framatome Replacement Steam Generators	32	23	
7 . HAUP LOCA Evaluation	3	24	
C. 2007 ECCS MODEL ASSESSMENTS			

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: 1 /15/08

1 . None	0		

D. OTHER*

1 . Removal of Reference 14 LHSI Error Evaluation	-30	16	(g)

LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =** 2043

* 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 . LTR-LIS-06-277, "Reconstitution Evaluation, 10 CFR 50.46 Reporting Plant Specific Text, and Updated Rackup Sheets for Prairie Island Unit 1, Cycle 24," 5/2006.

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 . NSP-02-9, "Nuclear Management Company Prairie Island Units 1 and 2 LBLOCA Accumulator Pressure and Volume Ranges Evaluation," February 15, 2002.

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

13 . NSP-02-59/LTR-ESI-02-194, "Final Evaluation of Large Break LOCA Error," December 2002.

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

15 . 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.

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

17 . 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.

18 . NSP-04-38, "Nuclear Management Company Prairie Island Units 1 and 2 10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.

19 . WCAP-16206-P, "SATP Engineering Report for Prairie Island," February 2004.

20 . 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: 1 /15/08

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

22 . NF-NMC-04-129, "Nuclear Management Company Prairie Island Unit 1, Cycle 23 Final RSE," August 2004.

23 . 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

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

25 . 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 15). 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 17 values, SG/Loop ΔP Re-tuning, Core Power Restoration.

(g) The note (f) sensitivity study allows for the removal of the Reference 13 engineering assessment.

(h) Items A.10 and A.11 presented as aggregate -66 °F entry prior to Reference 21 decomposition.

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 /15/08

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 . 2 Reconstituted Rods Evaluation	1	4	
C. 2007 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS	**PCT =** 1410		

* 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.

4 . LTR-LIS-06-277, "Reconstitution Evaluation, 10 CFR 50.46 Reporting Plant Specific Text, and Updated Rackup Sheets for Prairie Island Unit 1, Cycle 24," 5/2006.

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 /15/08

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. 2007 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 /15/08

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. 2007 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.