NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401-1927
Telephone (612) 330-5500

June 15, 1994

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

10 CFR part 50
Section 50.46

Prairie Island Nuclear Generating Plant
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Annual Report of Corrections to ECCS Evaluation Models

Attached is the annual report of corrections to Westinghouse Emergency Core Cooling System (ECCS) Evaluation Models for the calendar year of 1993. This report is being submitted in accordance with the provisions of 10 CFR 50, Section 50.46.

The applicable corrections noted in Attachments 1 and 2 have been applied to our current ECCS analyses of record, and all analyses were found to be in compliance with the applicable acceptance criteria (Attachment 3).

Please contact Mel Opstad (612-295-1653) if you require further information related to this submittal.

Melford T. Opstad

for Roger O Anderson
Director
Licensing and Management Issues

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
J Silberg

Attachments

Reference:

Westinghouse ESBU Nuclear Safety Advisory Letter NSAL-94-004R, dated February 8, 1994, that was transmitted via a February 8, 1994 Westinghouse letter from J Gasperini to R L Lindsey (NSP)

A001
1/1

ATTACHMENT 1

ECCS EVALUATION MODEL CHANGES AND ERRORS

Vessel and Steam Generator Calculation Errors in LUCIFER

ISHII Drift Flux Error

NOTRUMP Point Kinetics Error

NOTRUMP Drift Flux Flow Regime Map Errors

Core Node Initialization Error

NOTRUMP Heat Link Pointer Error

Fuel Rod Model Errors in SBLOCA

Charging/Safety Injection System Issues

Double-Disk Gate Valve Pressure Equalization

Large Break LOCA Fuel Rod Model Errors

High Temperature Fuel Rod Burst Model

Hot Assembly Average Rod Burst Effects

Revised Burst Strain Limit Model

VESSEL AND STEAM GENERATOR CALCULATION ERRORS IN LUCIFER

Background

The LUCIFER code is used to generate the component databases, from raw input data, to be used in the small and large break LOCA analyses. Errors were found in the VESCAL subroutine of the LUCIFER code. These errors were in the geometric and mass calculations of the vessel and steam generator portions of the needed data. All LOCA analyses using the LUCIFER code outputs are affected by these error corrections. The errors were corrected in a manner to maintain the consistency of the LUCIFER code.

The errors were determined to be a Non-Discretionary Change as described in Section 4.1.2 of WCAP-13451 and were corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 SBLOCA Evaluation Model
1981 ECCS Evaluation Model
1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Estimated Effect

Representative plant calculations indicate a net PCT effect of -16°F for small break LOCA and a -6°F for large break LOCA.

ISHII DRIFT FLUX ERROR

Background

An error was discovered both in WCAP-10079-P-A and the relevant coding in NOTRUMP SUBROUTINE ISHIIA which led to an incorrect calculation of the drift flux in NOTRUMP when a laminar film annular flow was predicted. The affected equation in WCAP-10079-P-A is Equation G-74 wherein a factor of 'g', the gravitational constant, was inadvertently omitted from both the documentation and the equivalent coding. The correction of this error returned NOTRUMP to consistency with the ultimate reference for the affected correlation.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant analyses were used to estimate a generic PCT effect of 0°F.

NOTRUMP POINT KINETICS ERROR

Background

An error was discovered in the coding used in the NOTRUMP User External SUBROUTINE VOLHEAT. The coding did not correctly perform the calculation described by Equation 3-12-28 of WCAP-10054-P-A. This calculation is only used during the time when the Point Kinetics option is used to determine the core power before reactor trip. Therefore, any analysis which used the more conservative assumption of constant core power until reactor trip time is not affected by this error. The correction of this error returned NOTRUMP to consistency with WCAP-10054-P-A.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant analyses were used to estimate a generic PCT effect of 0°F.

NOTRUMP DRIFT FLUX FLOW REGIME MAP ERRORS

Background

Errors were discovered in both WCAP-10079-P-A and related coding in NOTRUMP SUBROUTINE DFCORRS where the improved TRAC-P1 vertical flow regime map is evaluated. In Evaluation Model applications, this model is only used during counter-current flow conditions in vertical flow links. The affected equation in WCAP-10079-P-A is Equation G-65 which previously allowed for unbounded values of the parameter C_∞ contrary to the intent of the original source of this equation. This allowed a discontinuity to exist in the flow regime map under some circumstances. This was corrected by placing an upper limit of 1.3926 on the parameter C_∞ as reasoned from the discussion in the original source. As stated, this correction returned NOTRUMP to consistency with the original source for the affected equation.

Further investigation of the DFCORRS uncovered an additional closely related logic error which led to discontinuities under certain other circumstances. This error was also corrected and returned the coding to consistency with WCAP-10079-P-A.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant calculations indicated PCT effects ranging from -13°F to -55°F. For the purposes of tracking PCT, an estimated effect of -13°F will be assigned to this change.

CORE NODE INITIALIZATION ERROR

Background

An error was discovered in how the properties of CORE NODE components were initialized for non-existent regions in the adjoining FLUID NODE. In particularly this led to artificially high core temperatures during the timestep when the core mixture level crossed a node boundary, conservatively causing slightly more core mixture level depression than appropriate during this timestep. Correction of this error allows for a smoother mixture level uncovery transient during node crossings.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

The nature of this error led to an estimated generic PCT effect of 0°F.

NOTRUMP HEAT LINK POINTER ERROR

Background

An error was discovered in how NOTRUMP initialized certain HEAT LINK pointer variables at the start of a calculation. Correction of this error returned NOTRUMP to consistency with the original intent of this section of coding.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant analyses were used to estimate a generic PCT effect of 0°F.

FUEL ROD MODEL ERRORS IN SBLOCA

Background

A number of minor programming errors were corrected in the fuel rod heat up code used in SBLOCA analyses. These corrections were related to:

1. Individual rod plenum temperatures
2. Individual rod stack lengths
3. Clad thinning logic
4. Pellet/clad contact logic
5. Corrected gamma redistribution
6. Including ZrO_2 thickness at t=0 initialization
7. Numerics and convergence criteria of initialization.

These changes were determined to be Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451 and were implemented in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model

Estimated Effect

The cumulative effect of the error corrections and convergence criteria change was found to be less than approximately $\pm 4°F$. This change is therefore judged to have a negligible effect on PCT and on a generic basis the estimated effect will be reported as 0°F.

CHARGING/SAFETY INJECTION SYSTEM ISSUES

Background

Westinghouse has recently completed its evaluation of a potential safety issue regarding four specific issues related to the design and use of the miniflow line for the charging/safety injection (CHG/SI) pumps. Two of these issues involved SBLOCA PCT penalties for certain plants. One issue involves the operation of the centrifugal charging pump (CCP) miniflow line during accident conditions. A CCP runout condition may occur if the CCP injection lines were balanced with the CCP miniflow path closed and credit was taken for operator action to isolate the miniflow line during the accident. Also, the existence of this condition may impact the ECCS flows assumed in plant specific Small Break LOCA analyses. The other issue involves miniflow orifices that are used for the CHG/SI pumps. Westinghouse has supplied two different orifice types: 60 or 70 gpm orifice at a differential head of 6000 feet. Additional confirmation testing indicates that the orifice plates will allow a higher than design flow rate through the orifice at the design differential head. As a result, a discrepancy may exist between the installed miniflow line capacity and the ECCS analysis assumptions. The discrepancy would occur if the ECCS analysis assumed that the miniflow line resistance was based on the orifice allowing design flow at the design head as opposed to the higher as tested flow and head. Consequently, the miniflow path may permit more flow than previously determined which may reduce SI flow during injection.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model

Estimated Effect

The PCT effect on the Small Break LOCA Evaluation Model for this issue varied depending on the affected plant ECCS configuration and capability. The specific PCT penalty, for affected plants, is shown on the attached PCT Summary Sheet. If Westinghouse is in possession of sufficient plant configuration information to determine that the penalty is certainly applicable, this effect is included in Section D of the summary Sheet. Otherwise, it is included in Section E pending confirmation of the assumed plant configuration.

DOUBLE-DISK GATE VALVE PRESSURE EQUALIZATION

Background

Westinghouse completed the evaluation of a potential issue concerning use of double-disk gate valves in the emergency core cooling system (ECCS) as hot leg isolation valves. Use of these double-disk gate valves may involve an inner disc pressure equalization line that could set up a leak path into the hot leg during cold leg injection following a loss of coolant accident (LOCA). This condition could lead to inadequate cold leg injection resulting in an increase in PCT.

The design characteristic of a double-disk gate valve provides isolation by the downstream disk sealing against the valve seat. The mechanical seating force and the hydraulic force from the upstream pressure (SI pump) act to provide force to the valve seal surfaces. The double-disk gate valve design results in a volume of fluid which is enclosed between the discs when the valve is closed. As the fluid volume heats up, pressure greater than system pressure may develop and may cause the disks to bind against the seats to the extent that the valves can not be opened. To avoid this, many double-disk gate valves have been modified to include a pressure equalization line or a small hole in one of the disks to relieve the pressure between the disks. Based on generic leakage calculations it was determined that the double-disk gate valves modified to eliminate concerns for thermal binding could leak as much as 30 gpm per valve. This leakage into the RCS hot legs will increase steam binding during reflood and result in an increase in the calculated peak cladding temperature.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model
1978 ECCS Evaluation Model
1981 ECCS Evaluation Model
1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Estimated Effect

The PCT effect on the Large Break LOCA Evaluation Model for this issue varied depending on the affected plant ECCS configuration and capability. The specific PCT penalty, for affected plants, is shown on the attached PCT Summary Sheet. An assessment of this issue on Small Break LOCA Evaluation Model PCT results showed a nominal benefit which is being reported generically as a 0°F impact.

LARGE BREAK LOCA FUEL ROD MODEL ERRORS

Background

Minor errors in the rod heat up code used in Large Break LOCA analyses were corrected. These errors concerned conditions which exist during periods of pellet/clad contact and the internal book-keeping logic associated with clad thinning.

These changes were determined to be Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451 and were implemented in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1981 ECCS Evaluation Model with BASH

Estimated Effect

Representative plant calculations have shown that these corrections have a negligible effect on PCT for near Beginning-of-Life (BOL) fuel rod conditions (i.e. < 2000 MWD/MTU). These effects become prevalent as burnup increases, but are not expected to be of any significance until pellet/clad contact is predicted for steady-state operating conditions (typically > 8000 MWD/MTU). These corrections therefore result in a negligible PCT impact for Large Break LOCA licensing basis PCT's which are calculated with near BOL conditions. This impact is being reported generically as 0°F.

HIGH TEMPERATURE FUEL ROD BURST MODEL

Background

A model for calculating the prediction of zircaloy cladding burst behavior above the previous limit of 1742°F was implemented. This model was described to the NRC in:

> Letter ET-NRC-92-3746, N. J. Liparulo (W) to R. C. Jones (NRC), "Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperatures", September 16, 1992.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1981 ECCS Evaluation Model with BASH

Estimated Effect

The effect of the extended burst model has been directly incorporated in the Analysis of Record for those plants who are affected.

HOT ASSEMBLY AVERAGE ROD BURST EFFECTS

Background

The rod heat up code used in Small Break LOCA calculations contains a model to calculate the amount of clad strain that accompanies rod burst. However, the methodology which has historically been used is to not apply this burst strain model to the hot assembly average rod. This was done so as to minimize the rod gap and therefore maximize the heat transferred to the fluid channel, which in turn would maximize the hot rod temperature. However, due to mechanisms governing the zirc-water temperature excursion (which is the subject of the SBLOCA Limiting Time-in-Life penalty for the hot rod), modeling of clad burst strain for the hot assembly average rod can result in a penalty for the hot rod by increasing the channel enthalpy at the time of PCT. Therefore, the methodology has been revised such that burst strain will also be modeled on the hot assembly average rod.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model

Estimated Effect

Representative plant calculations have shown that this change introduces an approximately 10% increase in the SBLOCA Limiting Time-in-Life penalty on the hot rod. However, this penalty is being offset in affected plants PCT Summary Sheets by the Revised Burst Strain Limit Model described on the following page. These models will be implemented concurrently in the Small Break Evaluation Model rod heat-up code in 1994.

REVISED BURST STRAIN LIMIT MODEL

Background

A revised burst strain limit model which limits strains is being implemented into the rod heat up codes used in both Large Break and Small Break LOCA. This model, which is identical to that previously approved for use for Appendix K analyses of Upper Plenum Injection plants with WCOBRA/TRAC, as described in WCAP-10924-P-A, Rev. 1, Vol. 1, Add. 4, "Westinghouse Large Break LOCA Best Estimate Methodology: Volume 1: Model Description and Validation, Addendum 4: Model Revisions," 1991.

This has been determined to be a Non-Discretionary Change as discussed in Section 4.1.2 of WCAP-13451 and is being implemented in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1975 SBLOCA Evaluation Model
1985 SBLOCA Evaluation Model
1978 ECCS Evaluation Model
1981 ECCS Evaluation Model
1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Estimated Effect

The estimated effect on Large Break LOCA PCT's ranges from negligible to a moderate, unquantified benefit which will be inherent in calculations once this model is implemented. In Small Break LOCA, representative plant calculations indicate that the magnitude of the benefit is conservatively estimated to be exactly offsetting to the penalty introduced by the Hot Assembly Average Rod Burst issue documented on the previous page. This model will be implemented in both Large Break and Small Break Evaluation Models during 1994.

ATTACHMENT 2

RECENTLY COMPLETED POTENTIAL ISSUE EVALUATIONS

Large Break LOCA Rod Internal Pressure Issues

Small Break LOCA Limiting Time in Life

LARGE BREAK LOCA ROD INTERNAL PRESSURE ISSUES

Issue Description

Westinghouse recently completed an evaluation of a potential issue concerning the impact of increased beginning of life rod internal pressure (RIP) uncertainties on LOCA analyses. Historically, beginning of life fuel pressure and temperature uncertainties, were based upon end of life considerations. These RIP uncertainties were found to be potentially nonconservative. During the evaluation of this issue, a second issue related to the applicability of generic IFBA fuel analyses to updated LOCA Evaluation Models was also identified and combined with this issue since the underlying mechanisms were the same.

Technical Evaluation

The technical evaluation of this issue concluded that both the RIP uncertainty and the current IFBA designs with 200 psig initial fill pressure fuel typically will result in a maximum ±15°F PCT variation. Consequently, RIP manufacturing uncertainties and 200 psig initial fill pressure IFBA fuel do not have significant effects on the large break LOCA analyses. Also, based on these results, it was concluded that only nominal RIP (with an upper bound bias) should be used in the LOCA analyses for fuel designs with an initial cold fill pressure ≥ 200 psig. This is consistent with past LOCA analysis.

Specific analyses were performed for all plants with initial fill pressure < 200 psig. It was demonstrated that the acceptance criteria of 10 CFR 50.46 continued to be met for each of these plants.

Assessment of Safety Significance

A 10 CFR, Part 21 evaluation concluded that the effects of low initial fill pressure and increased RIP uncertainty will not represent a defect creating a substantial safety hazard and, more likely than not, will not result in a failure to comply with any applicable regulation relating to a substantial safety hazard. This conclusion was based upon the implementation of an extended burst and blockage correlation for burst temperatures above 1742°F and a more realistic minimum burnup assumption at hot full power conditions. In addition, any new reloads which would utilize low (<200 psig) initial fill pressure fuel would be specifically analyzed.

Recommended Actions

Resolution of this issue may have resulted in a plant specific PCT change and would be shown on the attached annual 50.46 report PCT Margin Utilization Sheets. Aside from determining reporting requirements relative to 10 CFR 50.46, no other utility action is necessary.

Plant Applicability List

See attached table.

SMALL BREAK LOCA LIMITING TIME IN LIFE - ZIRC/WATER OXIDATION TEMPERATURE EXCURSION

Issue Description

Westinghouse recently completed an evaluation of a potential issue with regard to burst/blockage modeling in the Westinghouse small break LOCA evaluation model. This potential issue involved a number of synergistic effects, all related to the manner in which the small break model accounts for the swelling and burst of fuel rods, modeling of the rod burst strain, and resulting effects on clad temperature and oxidation from the metal/water reaction models and channel blockage.

Technical Evaluation

Fuel rod burst during the course of a small break LOCA analysis was found to potentially result in a significant temperature excursion above the clad temperature transient for a non-burst case. Since the methodology for SBLOCA analyses had been to perform the analyses at a near beginning of life (BOL) condition, where rod internal pressures are relatively low, most analyses did not result in the occurrence of rod burst, and therefore may not have reflected the most limiting time in life PCT. In order to evaluate the effects of this phenomenon, Westinghouse has developed an analytical model which allows the prediction of rod burst PCT effects based upon the existing analysis of record.

Assessment of Safety Significance

A 10 CFR Part 21 evaluation concluded that the effects of the burst/blockage modeling in the Westinghouse small break LOCA evaluation model will not represent a defect creating a substantial safety hazard and, more likely than not, will not result in a failure to comply with any applicable regulation relating to a substantial safety hazard.

Recommended Actions

Resolution of this issue may have resulted in a plant specific PCT change. Since evaluation of the issue was in progress in 1992, some PCT effect may have previously been reported as a temporary impact. The evaluation for this issue has been finalized, and remaining PCT effects are now considered a permanent change with respect to evaluating 1993 reporting requirements. Aside from determining reporting requirements relative to 10 CFR 50.46, no other utility action is necessary.

Plant Applicability List

See attached table.

ATTACHMENT 3

Prairie Island 1 & 2
LOCA Peak Clad Temperature (PCT)
Margin Utilization Sheets

Revision Date: 1/25/94

Plant Name: Prairie Island Units 1 and 2
Utility Name: Northern States Power

Eval. Model: NOTRUMP
Fuel: 14x14 OFA ZIRLO(TM)
FQ=2.80 FΔH=2.00 SGTP=25%

	Reference*	Clad Temperature		Notes
A. ANALYSIS OF RECORD (7/93)	1	PCT=	1195 °F	1
B. PRIOR PERMANENT ECCS MODEL ASSESSMENTS		ΔPCT=	0 °F	
C. 10 CFR 50.59 SAFETY EVALUATIONS	Table A	ΔPCT=	0 °F	
D. 1993 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)				
1. Effect of SI in Broken Loop	2	ΔPCT=	150 °F	
2. Effect of Improved Condensation Model	2	ΔPCT=	−150 °F	
3. Drift Flux Flow Regime Errors	3	ΔPCT=	−13 °F	
4. LUCIFER Error Corrections		ΔPCT=	−16 °F	
E. TEMPORARY ECCS MODEL ISSUES**				
1. Effect of Leaking Double Disk Gate Valves	4	ΔPCT=	0 °F	2
F. OTHER MARGIN ALLOCATIONS				
1. None		ΔPCT=	0 °F	
LICENSING BASIS PCT + MARGIN ALLOCATIONS		**PCT=**	**1166 °F**	

* References for the Peak Clad Temperature Margin Utilization summary can be found in Table B.

** 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.

Notes:

1. Includes annular pellet evaluation.
2. A temporary penalty of 0 °F will be assessed for this issue until the exact configuration of the Prairie Island Units 1 and 2 double disk gate valves have been verified.

Revision Date: 1/25/94

Plant Name: Prairie Island Units 1 and 2
Utility Name: Northern States Power

Eval. Model: WCOBRA/TRAC, Addendum 4
Fuel: 14x14 OFA ZIRLO(TM)
FQ=2.40 FΔH=1.75 SGTP=15%

	Reference*	Clad Temperature		Notes
A. ANALYSIS OF RECORD (11/93)	5	PCT=	2089 °F	
B. PRIOR PERMANENT ECCS MODEL ASSESSMENTS		ΔPCT=	0 °F	
C. 10 CFR 50.59 SAFETY EVALUATIONS	Table A	ΔPCT=	0 °F	
D. 1993 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)				
1. None		ΔPCT=	0 °F	
E. TEMPORARY ECCS MODEL ISSUES**				
1. Effect of Leaking Double Disk Gate Valves	4	ΔPCT=	0 °F	1
F. OTHER MARGIN ALLOCATIONS				
1. None		ΔPCT=	0 °F	
LICENSING BASIS PCT + MARGIN ALLOCATIONS		PCT=	2089 °F	

* References for the Peak Clad Temperature Margin Utilization summaries can be found in Table B.
** 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.

Notes:

1. A temporary penalty of 0 °F will be assessed for this issue until the exact configuration of the Prairie Island Units 1 and 2 double disk gate valves have been verified.

Table A – 10 CFR 50.59 Safety Evaluations

Revision Date: 1/25/94

Plant Name: Prairie Island Units 1 and 2
Utility Name: Northern States Power

	Reference	Clad Temperature		Notes
I. SMALL BREAK ECCS SAFETY EVALUATIONS		ΔPCT=	0 °F	
A. None				
TOTAL 10 CFR 50.59 SMALL BREAK ASSESSMENTS		**PCT=**	**0 °F**	
II. LARGE BREAK ECCS SAFETY EVALUATIONS		ΔPCT=	0 °F	
A. None				
TOTAL 10 CFR 50.59 LARGE BREAK ASSESSMENTS		**PCT=**	**0 °F**	

Notes:
None

Table B – References

1. NSP-93-521, "Northern Power Company Prairie Island Units 1 and 2 Small Break Loss-of-Coolant Accident Final Engineering Report for the ZIRLO(TM) Fuel Upgrade," July 30, 1993.

2. NSP-93-222, "Northern Power Company Prairie Island Units 1 and 2 Safety Injection in the Broken Loop," September 22, 1993.

3. NSP-93-224, "Northern Power Company Prairie Island Units 1 and 2 10 CFR 50.46 Notification and Reporting Information," September 27, 1993.

4. NSP-93-216, "Northern States Power Company Prairie Island Units 1 and 2 Double Disk (DD) Gate Valve Pressure Equalization," June 23, 1993.

5. NSP-93-529, "Northern Power Company Prairie Island Units 1 and 2 Large Break Loss-of-Coolant Accident Final Engineering Report for the ZIRLO(TM) Fuel Upgrade," November 3, 1993.