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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

January 8, 1980

Director of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Fuel Rod Strain and Flow Blockage Models

On November 9 and 27, 1979, Mr D G Eisenhut requested that all operating light water reactor licensees review vendor and NRC information on fuel cladding strain and fuel assembly flow blockage models and confirm that the present models are conservative with respect to the NRC Staff's model. If regions exist where either the clad strain or assembly flow blockage models are less conservative than the Staff's model, additional calculations should be performed to demonstrate compliance with 10CFR50.46.

We have contracted with Exxon Nuclear Company and Westinghouse, fuel suppliers for the Prairie Island Nuclear Generating Plant, to perform additional analyses in our behalf.

The NRC staff met with Westinghouse during November and December, 1979 in support of all Westinghouse units. References 1 through 4 contain Westinghouse's response to NRC fuel rod model concerns.

The current ECCS analysis for Westinghouse fuel uses the February, 1978 Evaluation model. The assumed F_Q in the analyses submitted on February 21, 1979 was 2.28. Recent recalculations by Westinghouse utilizing increased flow blockage, in accordance with NRC staff concerns, resulted in a reduction of F_Q to 2.24 (see Attachment 1). These additional Westinghouse calculations will not affect the Technical Specifications or impose additional limitations on the Prairie Island plant operations since the existing Technical Specifications limit F_Q to 2.21 globally (i.e. $F_Q^N \leq 2.145$ at HFP) based on the Exxon Nuclear Company fuel.

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For the ENC fuel, Exxon Nuclear Company previously conducted analyses using the ENC-WREM-IIA PWR-evaluation model. Descriptions of pertinent fuel rod model information was presented to the NRC in References 6 through 8. Followup plant specific calculations indicate that the 10CFR 50.46 criteria are satisfied with the existing F_Q limit of 2.21 (see Attachment 2). Thus, no changes to the existing Technical Specifications on F_Q limit appear to be required based on these recent analyses. Additional F_Q exposure dependence sensitivity studies are being conducted for higher exposures of the ENC fuel. We intend to submit Reference 5 for NRC review when we submit a Technical Specification change planned for spring 1980.

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/JAG/jh

cc: J G Keppler
G Charnoff

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References

1. Letter NS-TMA-2147, T M Anderson (W) to D G Eisenhut (NRC), dated November 2, 1979.

2. Letter NS-TMA-2158, T M Anderson (W) to D G Eisenhut (NRC), dated November 16, 1979.

3. Letter NS-TMA-2163, T M Anderson (W) to D G Eisenhut (NRC), dated November 16, 1979.

4. Letter NS-TMA-2174, T M Anderson (W) to D G Eisenhut (NRC), dated December 7, 1979.

5. Exxon Nuclear Company Reports, "Exposure Sensitivity Study for ENC-XN-1 Reload Fuel at Prairie Island Unit 1 using the ENC-WREM-IIA PWR Evaluation Model", XN-NF-79-18 [P], March 1979 (proprietary); XN-NF-79-18 [NP] May 1979 (non-proprietary).

6. Letter G E Owsley (ENC) to D G Eisenhut (NRC), dated November 2, 1979.

7. Letter G F Owsley (ENC) to D G Eisenhut (NRC), dated November 4, 1979.

8. Letter G F Owsley (ENC) to D G Eisenhut (NRC), dated November 16, 1979.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Docket No. 50-282
50-306

LETTER DATED JANUARY 8, 1980
RESPONDING TO NRC REQUEST
FOR REVIEW OF FUEL ROD STRAIN AND FUEL ASSEMBLY FLOW BLOCKAGE MODELS

Northern States Power Company, a Minnesota corporation, by this letter dated January 8, 1980 hereby submits information in response to NRC request for information concerning fuel rod strain and fuel assembly flow blockage models.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L. O. Mayer
L O Mayer
Manager of Nuclear Support Services

On this 8th day of January, 1980, before me a notary public in and for said County, personally appeared L O Mayer, Manager of Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Jeanne M Hacker

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JEANNE M. HACKER
NOTARY PUBLIC · MINNESOTA
HENNEPIN COUNTY
My Commission Expires May 6, 1986

Attachment 1
January 8, 1980

WESTINGHOUSE FUEL

1.0 Introduction

This attachment summarizes information on the large break LOCA analyses specific to the Prairie Island Westinghouse fuel. The data contained herein is based on additional calculations performed by Westinghouse subsequent to the November 1, 1979 meeting.

The current ECCS analysis (Reference 1) for the Westinghouse fuel utilizes the February 1978 model. The limiting break is the 0.4 DECLG. Data pertinent to the analysis is as follows:

Core Peaking	2.28
Clad Burst Region	
Hot Rod Maximum Temperature Calculated	1840F
Elevation	5.75ft
Clad Non-ruptured Region	
Hot Rod Maximum Temperature Calculated	2179F
Elevation	7.5 ft
Clad strain during blowdown at this elevation	4.2 %
Maximum clad strain at this elevation	7.2 %

Maximum temperture for this node occurs when the core reflood rate is less than 1.0 inch per second and reflood heat transfer is based on the steam cooling calculation

Average Hot Assembly Rod Burst Elevation	N/A ft
Hot Assembly Blockage Calculated	0 %

2.0 Burst Node Calculations

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in forms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges.

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From NS-TMA-2174:

For the Burst node of the clad:

- $0.01\ \Delta FQ \leftrightarrow \sim 150^{o}F$ Burst Node ΔPCT

- Use of the NRC burst model could require an FQ reduction of 0.015

- The minimum estimated impact of using the NRC strain model is a required FQ reduction of 0.03

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (.015 + .03)\ (150^{o}F/.01) = 675^{0}F$$

Margin to the $2200^{o}F$ limit is:

$$\Delta PCT_2 = 2200.^{o}F - PCT_B = 360^{o}F$$

The FQ reduction required to maintain the $2200^{o}F$ clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\Delta FQ}{150^{o}F}\right)$$

$$= (\underline{675} - \underline{360})\ \left(\frac{.01}{150}\right)$$

$$= \underline{\quad 0.021 \quad}$$

3.0 Non-Burst Node Calculations

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generally in this calculation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is $+20^{o}F$ per percent decrease in strain at the maximum clad temperature locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain

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that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

ΔPCT_3 = (20°F/.01 strain) (MAX Strain - Blowdown Strain)

$= \left(\frac{20}{.01}\right) (.072 - .042)$

$= \underline{\qquad 60 \qquad}$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

ΔPCT_4 = 1.25°F (50 - Percent Current Blockage) + 2.36°F (75-50)

$= 1.25 (50 - \underline{\ 0\ }) + 2.36 (75-50)$

$= 121°F$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 181.°F$

Margin to the 2200°F limit is

$\Delta PCT_5 = 2200°F - PCT_N = 21\ °F$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6) \left(\frac{.01 \Delta FQ}{10°F\ PCT}\right)$

$\Delta FQ_N = \underline{\qquad 0.16 \qquad}$

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N

or; $\Delta FQ_{Penalty} = \underline{\qquad 0.16 \qquad}$

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4.0 Improved Analytical/Modeling Effects

The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

5.0 Net Effect on F_Q

Considering the appropriate F_Q credit identified in section 4.0 and the $\Delta FQ_{Penalty}$ calculated in section 3.0, the net peaking factor limit adjustment required for plant operation during this interim period is calculated as follows:

$$\begin{aligned} F_Q \text{ Adjustment} &= F_Q \text{ Credit} - F_Q \text{ Penalty} \\ &= .12 - .16 \\ &= -.04 \end{aligned}$$

Thus the appropriate F_Q for the Westinghouse fuel used at Prairie Island is equal to 2.24 (2.28 - 0.04 = 2.24)

6.0 References

(1) Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC), dated February 21, 1979.

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Attachment 2
January 8, 1980

Exxon Nuclear Company Fuel

1.0 Introduction

This attachment summarizes information on the large break LOCA analyses specific to the Exxon Nuclear Company (ENC) fuel used in Prairie Island, Units 1 and 2. The data contained herein is based on additional information supplied by Exxon Nuclear Company prior to and subsequent to the November 1, 1979 meeting.

The current ECCS analyses for the Prairie Island Units 1 and 2 ENC fuel have been previously reported (References 1, 2). Additional F_Q exposure sensitivity studies have been conducted by ENC and reported to NSP (Reference 3). This report will be provided to the NRC in support of a license amendment request after additional calculations have been completed for higher peak pellet burnup cases.

2.0 Model Review

ENC has reported that the change in calculated peak clad temperature (PCT) for ENC fuel at Prairie Island when the NRC model for clad swelling and rupture (Reference 4) is used in place of the ENC model is a decrease of less than 2°F (Table 1).

TABLE 1

Effect of NRC Rupture and Flow Blockage
Model on Prairie Island ENC ECCS Analyses

Peak Pellet Exposure ($^{GWD}/_{MTM}$)	0	27
FQ	2.21	2.21
Heatup Rate at Rupture, °c/sec	7.0	2.0
PCT Impact of NRC Model vs ENC Model, °F	-1.4	- 16

The calculation is for the 0.4DECLG limiting break for the Prairie Island Units (Reference 5). The present calculations were made for ENC Reload XN-1 fuel at Prairie Island Unit 1 (30 mil clad). The results are enveloping of ENC Reload XN-1 fuel for Prairie Island Unit 2 (Reference 6). The present sensitivity calculations were made in accordance with ENC's approved WREM-IIA PWR ECCS Evaluation Model (Reference 7, 8, 9, 10). The fuel rod internal pressure corresponds to the ENC model (Reference 11) for nominal conditions.

3.0 Conclusions

In view of the reduced PCT with the NRC clad swelling and rupture model, the current total peaking limit of 2.21 for ENC fuel at Prairie Island insures conformance to 10CFR50.46.

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4.0 References

1. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC), dated December 29, 1978; Exhibit C.

2. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC), dated October 30, 1979; Attachment

3. "Exposure Sensitivity Study for ENC XN-1 Reload Fuel at Prairie Island Unit 1 using the ENC-WREM-IIA PWR Evaluation Model; XN-NF-79-18 [P], March 1979 (proprietary); XN-NF-79-18 [NP], May 1979 (non proprietary).

4. D A Powers and L O Mayer, "Cladding Swelling and Rupture Models for LOCA Analysis," Draft NUREG-0630, November 8, 1979.

5. "ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 using ENC WREM-IIA PWR Evaluation Model," XN-NF-78-46, November 1978.

6. "Prairie Island Unit 2 Nuclear Plant Cycle 5 Safety Analysis Report," XN-NF-79-67(NP), August 1979.

7. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA," XN-NF-78-30, August 1978.

8. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," XN-75-41:

    a. Volume I; July 1975
    b. Volume II, August 1975
    c. Volume III, Revision 2, August 1975
    d. Supplement 1, August 1975
    e. Supplement 2, August 1975
    f. Supplement 3, August 1975
    g. Supplement 4, August 1975
    h. Supplement 5, Revision 5, October 1975
    i. Supplement 6, October 1975
    j. Supplement 7, November 1975

9. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II." XN-76-27, July 1976; Supplement 1, September 1976; Supplement 2, November 1976.

10. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA: Responses to NRC Request for Additional Information," XN-NF-78-30(A) & XN-NF-78-30, Amendment 1(A), May 1979.

11. "Flow Blockage and Exposure Sensitivity Study for ENC D C Cook Unit 1 Reload Fuel Using ENC WREM-II Model," XN-76-51; Supplement 1, January 1977; Supplement 2, February 1978; Supplement 3, April 1978.

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