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Docket Nos. 50-282
and 50-306

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Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 67 and 61 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 in partial response to your application dated June 24, 1983.

The amendments change the exposure dependence function Bu(z) to 1.0 for all values of peak pellet exposure from 0 to 55 GWD/MTU. These amendments complete all items that you requested by letter dated June 24, 1983.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register Notice.

Sincerely,

Original signed by

Dominic C. DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 67 to DPR-42
2. Amendment No. 61 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated June 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 67 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 28, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated June 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 61 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 28, 1983

# ATTACHMENT TO LICENSE AMENDMENTS

## AMENDMENT NOS. 67 AND 61 TO FACILITY OPERATING LICENSE

## NOS. DPR-42 AND DPR-60

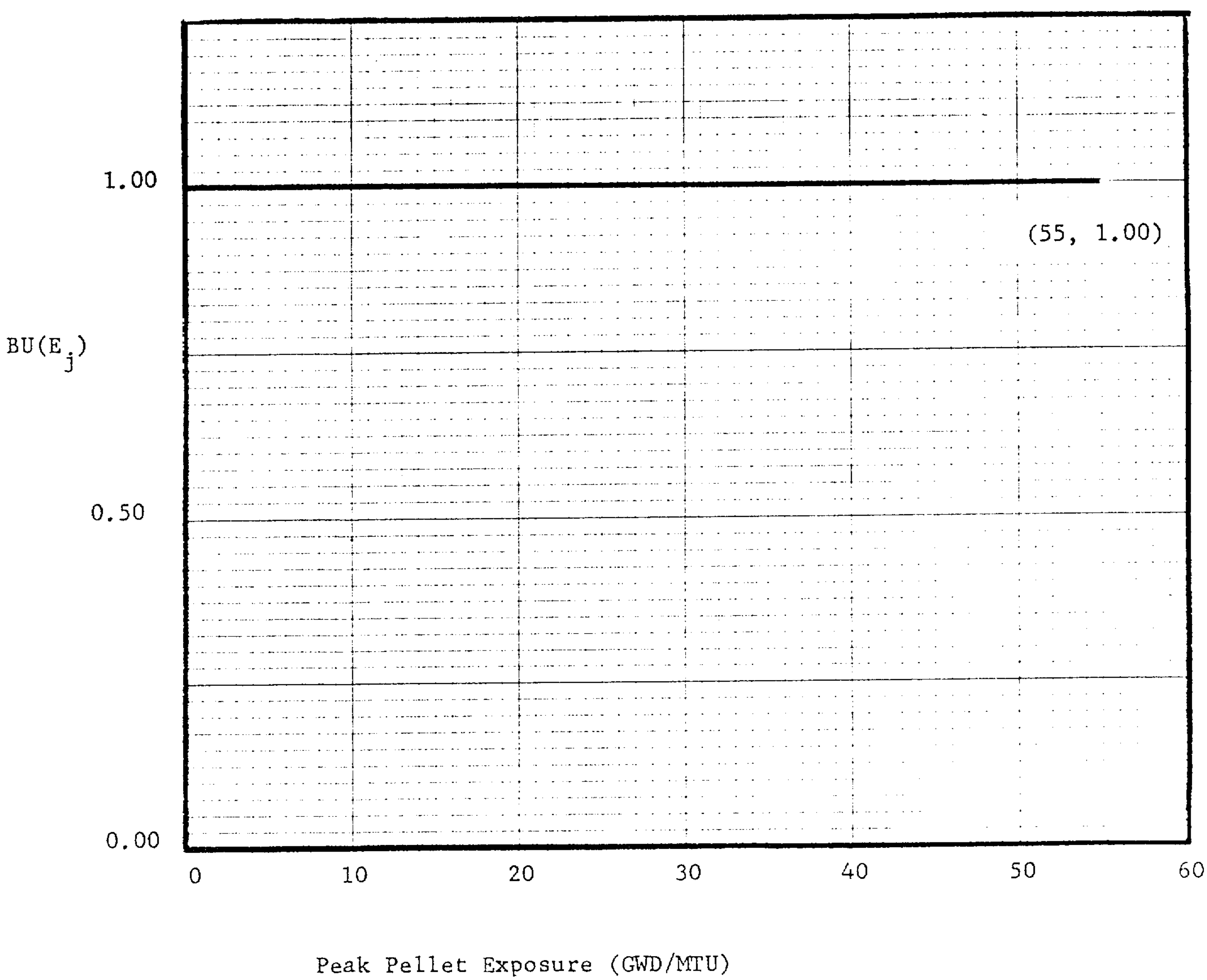
## DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of changes.

| Remove | Insert |
|---|---|
| Figure TS.3.10-7 | Figure TS.3.10-7 |

Normalized Exposure Dependent

Function $BU(E_j)$ for Exxon Nuclear Company Fuel




Prairie Island Unit 1 - Amendment No. ~~29~~, ~~44~~, ~~56~~, ~~58~~, 67
Prairie Island Unit 2 - Amendment No. ~~23~~, ~~38~~, ~~50~~, ~~52~~, 61

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NOS. 67 AND 61

## TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

## NORTHERN STATES POWER COMPANY

## PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

## DOCKET NOS. 50-282 AND 50-306

### Introduction

By letter dated June 24, 1983 Northern States Power Company (NSP) requested changes to the Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant Units Nos. 1 and 2. The proposed TS changes involve the local power hot channel factor $F_Q^N$ limit in the TS 3.10-1, 3.10-2, 3.10-9 and 3.10-11 where (1) a numerical value in the $F_Q^N$ limit expression will be changed from 2.21 to 2.32; and (2) the normalized exposure dependence function Bu(z) curve will be changed to 1.0 for all values of peak pellet exposure from 0 to 55 GWD/MTU in Figure TS.3.10-7. The Commission, by letter dated October 3, 1983 issued Amendment Nos. 66 and 60 related to the local power hot channel factor $F_Q^N$ limit and this evaluation addresses the normalized exposure dependence function Bu(z) curve appearing in the Figure TS.3.10-7.

### Staff Evaluation

In order to support the proposed TS change, the licensee submitted, by letter dated June 24, 1983, a report titled "Prairie Island Unit 1 and 2 Limiting Break LOCA-ECCS Analysis Using EXEM/PWR" (XN-XF-83-38) prepared by Exxon Nuclear Company describing the analysis and results for the Prairie Island Unit Nos. 1 and 2 for a postulated large break LOCA. The analysis was made with the following conditions:

(1) The double-ended cold leg guillotine break with a discharge coefficient of 0.4: The scenario has been identified in the previous analyses as the most limiting break.

(2) An entire core with ENC TOPROD fuel: With respect to LOCA, the TOPROD fuel design is more limiting than the ENC XN-1 and XN-2 fuels due to the smaller pin diameter and the increased core flow area which reduces core reflood rates in the LOCA analysis.

(3) 5% of steam generator tubes being uniformly plugged.

(4) Including 2% power uncertainty resulting in the allowable linear heat generation rate of 15.02 kw/ft corresponding to total power peaking factor of 2.32 and nuclear enthalpy rise factor of 1.55 for the entire fuel exposure.

(5) Maximum peak pellet exposure of 55 GWD/MTU.

The reactor coolant system nodalization was modeled in accordance with an approved ENC ECCS modeling described in XN-NF-77-25(A) (Ref. 1) for a 2-loop Westinghouse PWR with dry containment. The LOCA analysis was performed using ENC's EXEM/PWR ECCS evaluation model (Ref. 2). This evaluation model uses the following computer codes:

(1) RELAP-EM (REf. 3) for the system blowdown and hot channel blowdown calculations;

(2) CONTEMPT-LT/22 as modified in CSB G-1 (Ref. 4) for the containment back pressure calculation;

(3) REFLEX (Ref. 5) for system reflood calculation;

(4) RODEX2 (Ref. 6) for initial fuel rod stored energy, fission gas release and internal gas inventory calculations; and

(5) TOODEE2 (Ref. 7) for the calculation of final fuel rod heatup.

The RELAP-EM, CONTEMPT-LT/22, REFLEX and TOODEE2 codes have previously been approved by the NRC. The RODEX2 code has recently been reviewed by the staff and has been found acceptable for use in the LOCA initial stored energy and rod pressure calculations (formal SER for RODEX2 is being prepared). In addition, an approved cladding swelling and rupture model described in XN-NF-82-07, Revision 1 (Ref. 8) was used in the calculation of the cladding rupture, strain and flow blockage in the ENC's EXEM/PWR ECCS evaluation model and, therefore, this part of the analysis is acceptable. The changes incorporated in the ENC's EXEM/PWR ECCS evaluation model and the LOCA input from the revised RODEX2 eliminates the need for a burnup dependent penalty imposed by Figure TS.3.10-7. Therefore, the proposed change to Figure TS.3.10-7 that normalizes the exposure dependence function Bu(z) curve to 1.0 for all values of peak pellet exposure from 0 to 55 GWD/MTU is acceptable. The overall EXEM/PWR ECCS evaluation is still under review by the staff. However, the review has progressed to the point to conclude that the evaluation model is acceptable for dealing with the exposure dependence function Bu(z).

The LOCA analysis was performed with two analyses: from the beginning of life to 15 GWD/MTU and from 15 GWD/MTU to 55 GWD/MTU. The most limiting fuel conditions in the respective exposure ranges were used in the analysis. The combination of the highest stored energy, rod pressure and decay power was used to bound the LOCA-ECCS analysis over the exposure ranges. The results of these analysis have shown the maximum peak cladding temperatures to be 2091°F and 2142°F, respectively, for the fuel exposure ranges prior to 15 GWD/MTU and up to an exposure of 55 GWD/MTU. The local metal-water reactions are 4.68% and 5.6% for the two fuel exposure ranges, and total core metal-water reaction is less than 1%. These peak clad temperatures (PCTs) and metal-water reactions are within the limit imposed in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors". Therefore the proposed change of the exposure dependence function Bu(z) is acceptable as this function relates to the LOCA analysis.

The effect of increased fuel exposure to 55 GWD/MTU on the off-site doses has been addressed in our letter dated September 27, 1983 to the licensee. By letter dated August 16, 1983 the licensee transmitted a report assessing the potential radiological consequences for high exposure fuel (XN-NF-719)(P) August 1983) prepared by Exxon Nuclear Company (the fuel vendor). The report justifies extended fuel exposures for the core average (EOC) to 30 GWD/MTU, for the maximum fuel assembly (EOL) to 49.5 GWD/MTU and the peak pellet to 55 GWD/MTU related to exposure dependence function Bu(z). Our evaluation transmitted by letter dated September 27, 1983 concludes that the extended burnup described in the fuel vendor's document (XN-NF-719)(P) will not result in higher doses from those previously analyzed for postulated accidents nor will doses exceed the dose guidelines of 10 CFR 100.11. On this basis, the staff concludes that the extended burnup (i.e., average EOC exposure, 30 GWD/MTU; maximum fuel assembly EOL exposure, 49.5 GWD/MTU; and peak pellet exposure, 55 GWD/MTU) at a core thermal power of 1683 MWt is acceptable. Therefore, the proposed change of the exposure dependence function Bu(z) is acceptable as this function relates to off-site doses from potential radiological consequences for high exposure fuel (55 GWD/MTU).

## Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

## Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 28, 1983

Principal Contributors:
D. C. DiIanni
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Y. Hsii
M. Dunenfeld
S. L. Wu

References

1. XN-NF-77-25(A), "ENC ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment Using the ENC WREM-II ECCS Model - Large Break Example Problem", September 1978.

2. XN-NF-82-20, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates", Revision 1, August 1982; Supplement 1, March 1982; Supplement 2, March 1982.

3. Letter From T. A. Ippolito (NRC) to W. S. Nechodam (ENC), "SER for ENC RELAP4-EM Update", March 1979.

4. NRC Branch Technical Position CSP G-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation".

5. Memorandum from R. L. Tedesco to B. K. Grimes, "Review of Exxon Nuclear Company Topical Report (ENC WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA), TAC-10691", March 23, 1979.

6. XN-NF-81-58(P), Revision 2, "Fuel Rod Thermal-Mechanical Response Evaluation Model", February 1983.

7. NUREG-75/057, "ROODEE2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program", May 1975.

8. XN-NF-82-07(P), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model", August 1982.