NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

March 30, 1979

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42
50-306 DPR-60

Additional Information Related to
Prairie Island License Amendment dated December 29, 1978

On December 29, 1978, Northern States Power Company requested a License Amendment covering use of a new fuel type for Units 1 and 2. In discussions with the NRC staff, additional information in regard to the subject License Amendment was requested in the following areas:

(a) Fuel
(b) Nuclear Design
(c) Accident and Transient Analysis
(d) ECCS Analysis
(e) Startup Physics Tests
(f) Power Distribution Control

Attachment A and B contain the additional information pertaining to the aforementioned areas. Attachment C contains revised technical specification pages related to power distribution control. Attachment D contains the affidavit of Mr R Nilson of Exxon Nuclear Company submitted in accordance with 10CFR 2.790(b)(1)(ii). This affidavit describes the basis for the request for exemption from public disclosure of the document XN-NF-79-6[P] included as Attachment E.

Forty (40) copies of XN-NF-79-6[P] are being transmitted at this time. Nonproprietary copies of this document will be submitted in the near future.

Very truly yours,

L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/JAG/ak

Attachments

7904030302

cc: J G Keppler
G Charnoff

ATTACHMENT E CONTAINS 10CFR 2.790 PROPRIETARY INFORMATION

DPR-42
DPR-60

ATTACHMENT A

to

Letter dated March 30, 1979

Attachment A contains additional information related to the December 29, 1978 Prairie Island License Amendment Request

A. Fuel

1. Rod Bow

The Prairie Island Unit 1 Cycle 5 reload does not contain a σ95/95 factor to account for region to region variation in rod bowing for the following reasons:

Analysis of the ENC rod bow data including batch-to-batch variation shows that ENC's current practice is more than adequate to envelope the data within the required 95/95 statistical statement. Therefore, no reduction in operating margin greater than that already accounted for needs to be applied.

The current practice is to use the rod bow data base applicable to the spacer span exhibiting the maximum 95/95 fractional closure. The present data base includes three regions of fuel in two different reactors but of fuel of identical design. The observed region (batch) to region 95/95 variation is enveloped by a factor of 1.21 up to a batch burnup of 13,100 MWD/MTU; the data trend line running essentially parallel to, but 21 percent higher than as shown in Figure 5.5 of XN-NF-78-34.

Based on approximately 19,000 rod spacing measurements for ENC fuel, the 95/95 closure (and maximum observed closure) nowhere approaches those reported in the public domain. This is illustrated in Figure 5.3, 5.4 and 5.5 of XN-NF-78-34. The staff SER does not explicity require use of only worst span data to determine the 95/95 closure. The added conservatism might be appropriate if the observed maximum closure were near the top of the fuel where both LOCA and DNB margins are less.

Comparing the present practice against the maximum observed region 95/95 closure results in the maximum region 95/95 closure being enveloped with 10 percent margin, including the 1.2 cold-to-hot multiplier. Magnitude of closure is less for ENC fuel, the region-to-region variation will be less. It is concluded that the present practice provides 10 percent margin against the worst observed batch and that no further conservatism need be applied.

The following is a description of the ENC procedure used to calculate δ_B nuclear augmentation due to rod bow included in the calculation of F_Q^N:

A three-dimensional Monte Carlo code (KENO IV) was used in a two-dimensional mode to calculate the relative changes in the fuel rod powers due to a bowed fuel rod in a 7 x 7 array of discrete fuel rods. The standard KENO cross section set, sixteen group Hansen Roach, was used in the calculation. In the 7 x 7 array input into KENO IV, a bowed rod was simulated by a displacement of the fuel rod at the plane analyzed. This technique shows the effect of an increase and decrease in local neutron moderation for fuel rods adjacent to the bowed rod. Several of the methods employed to model the array of fuel rods in the code were:

a) discrete cylindrical UO_2 fuel regions consisting of unexposed 2.95 w/o enriched U-235;

b) a discrete cylindrical clad region surrounding each fuel region;

c) and a non-borated moderator.

All cases were run with spectrally reflecting boundaries at the edges of the 7 x 7 array.

Two rod bow cases were analyzed to determine the change in local fuel rod powers due to a bowed or displaced fuel rod. In the first case a single fuel rod was displaced along the diagonal of the fuel rod cell. The gap between this displaced fuel rod and the adjacent neighboring fuel enveloped the extent of closure anticipated for ENC reload fuel. In the second case, the four fuel rods adjacent to the central rod in a 7 x 7 array were displaced along the x and y axes to an extent so as to envelope anticipated closure for ENC reload fuel. This second case is considered to be one of the most limiting in the rod bow analysis due to the magnitude of the increase in the neutron moderation around the center fuel rod.

The resulting change in relative rod power as determined above, was correlated as a function of the change in local neutron moderator area for the central fuel rod in the 7 x 7 array as decribed above. Since the change in local neutron moderator area is proportional to rod bow (gap closure), the nuclear augmentation due to rod bow was determined.

It should be noted that the increase in nuclear augmentation factor due to rod bow with burnup is offset by the actual F_Q^N decrease with burnup. Representative values for the two Prairie Island units are shown below:

Burnup (MwD/MTU)	F_Q^N[1] Unit 1, Cycle 4	F_Q^N[1] Unit 2, Cycle 3
0	1.69	1.685
9500	1.51	--
9700	--	1.50

[1] Does not include engineering (1.03) and measurement and calculational (1.05) uncertainties.

This 11% decrease compensates for the 2% increase in nuclear augmentation factor due to rod bow calculated by the staff.

2. Gadolinia Characteristics

The gadolinia demonstration program in Prairie Island will enhance the data base obtained from the Palisades program and BWR programs described in XN-NF-78-47. To provide the basis for these programs, Exxon Nuclear has investigated the consequences of spiking urania fuel with gadolinia (Gd_2O_3). This investigation has included evaluation of thermal conductivity, fuel densification, and fission gas release. The first two areas have been investigated for gadolinia concentrations as high as 5%, whereas the fission gas release work has been based on gadolinia concentrations of 1%. No change in fuel performance was noted for densification nor fission gas release when the fuel was spiked with gadolinia. A slight decrease in thermal conductivity has been identified for gadolinia bearing fuel and this effect increases with increasing gadolinia content.

The methods and models used for predicting gadolinia depletion are described in Appendix A to XN-NF-78-47. As described, the gadolinia bearing fuel cell is depleted and the cross sections calculated with a multi-group transport theory code. Prior to incorporation into the diffusion theory core model, the cross sections are adjusted to preserve the reaction rates calculated with the transport theory code.

The impact on the core power distribution of off-nominal extremes of poison worth were investigated and also reported in Appendix A of XN-NF-78-47. These analyses confirm that even if the actual gadolinium poison worth varies from the predicted worth, the safe full power operation of the plant will not be compromised.

The safe operation of the plant with respect to power peaking is confirmed through periodic measurements of power distribution as described in Sections B and F. The core will be regularly monitored to insure conformance with Technical Specification, Section 3.10.

Accuracy of predictions for the Palisades plant having 32 gadolinia pins is described in Section A2 of Appendix A to XN-NF-78-47. Measured power distributions show power differences of 1% to 2% variance from ENC prediction in the gadolinia bearing assemblies. The periodic boron sampling and power distribution measurements required by the Prairie Island technical specifications provide a comparison of the predicted and actual poison and fuel depletion rates.

3. Fuel Surveillance

To our knowledge, examination by ENC of similar PWR fuel assemblies and BWR fuel assemblies at other facilities has not revealed any abnormalities. Thus there is no reason to expect any abnormal mechanical behavior in the Prairie Island fuel. However, to confirm these expectations, ENC does intend to conduct visual examination and rod bow checks of several fuel assemblies after irradiation.

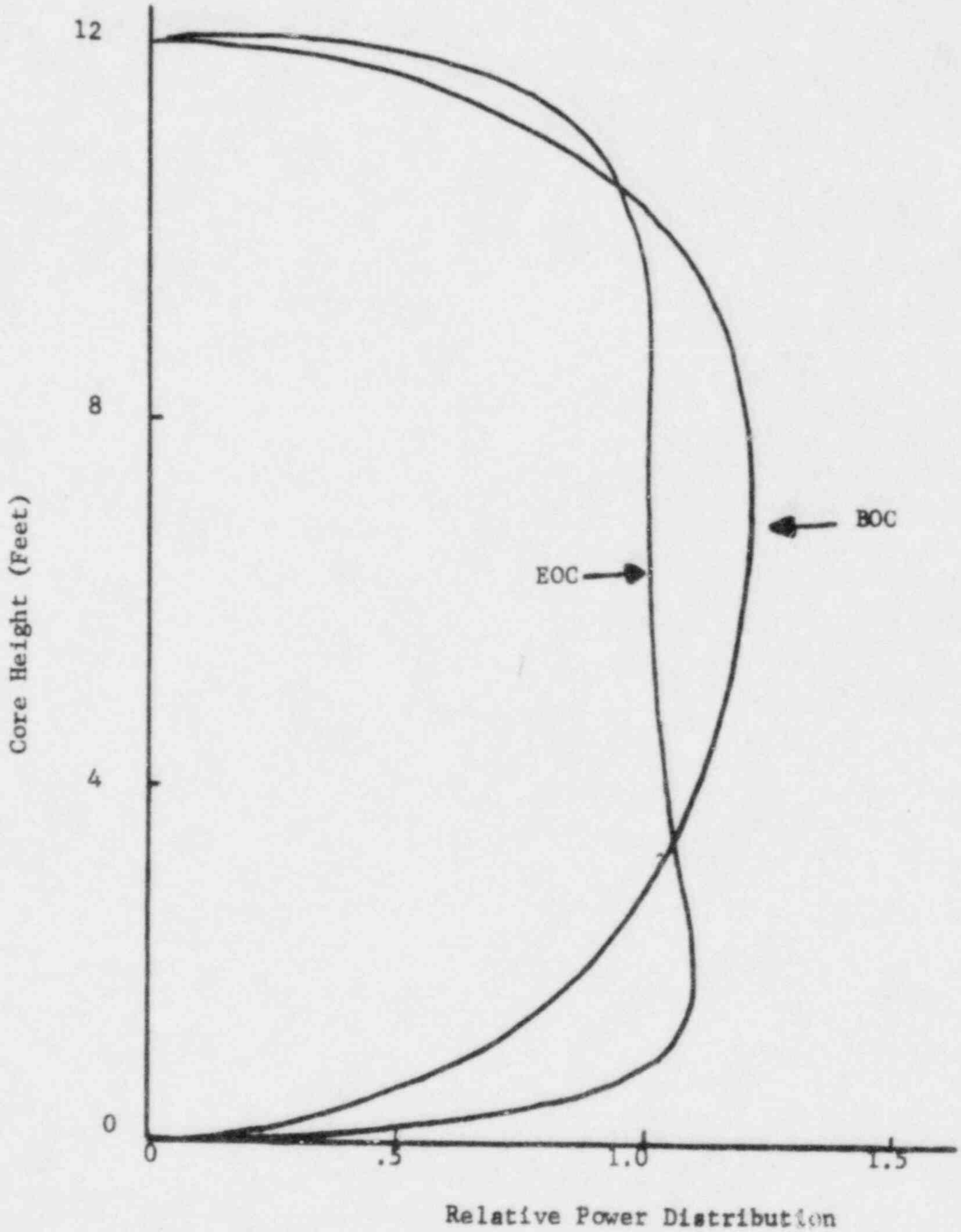
B. Fuel Design

The Prairie Island reload core will contain 64 fuel pins, loaded with 1 w/o Gd_2O_3 in a uranium dioxide matrix. To ensure that power distribution in the core stays close the predicted values, the plant nuclear engineering staff will conduct monthly flux maps in accordance with Specification 3.10.B.1.b. If the measured $F_Q^N(Z)$ is less than, but within 2% of the 3.10.B.1.b F_Q^N limit, the plant staff will take weekly flux maps until such time as the difference is more than 2%. This program will continue until such time as the NRC has completed review of XN-NF-77-57 and supporting data.

The moderator temperature coefficient has been predicted to become negative at >70% power. A test will be conducted at a power greater than 70% to confirm the design prediction. This test will be conducted at a stable xenon condition.

Figure B1 illustrates the predicted axial power distribution for the hot full power BOC (0 MWD/MTU) and EOC (11 300 MWD/MTU).

The measurement uncertainty for the peak linear heat rate of 5% had been previously substantiated by Westinghouse and was incorporated into the plant Technical Specifications. This uncertainty is the sum of the uncertainty associated with hardware, software, and data input to the core monitoring system. The basis for this uncertainty has been substantiated by ENC in report XN-NF-79-6[P]. Measurements and measurements uncertainty are addressed in the report along with the uncertainty in pre-operational data input to the monitoring system.



PRAIRIE ISLAND UNIT 1, CYCLE 5 CORE AVERAGE
AXIAL POWER DISTRIBUTION HOT FULL POWER

FIGURE B1

C. Accident and Transient Analyses

1. DNBR

The DNBR values listed in the plant transient analysis (PTS) XN-NF-78-35, differ from those reported in the Final Safety Analysis Report, Section 14 for several reasons. The plant transient analysis assumed lower values for F_Q, F_H^N and T_{inlet} compared to the Cycle 1 analysis. These differences are summarized in Table C1.

Table C1

Comparison of Variables Related to DNBR

Parameter	FSAR	PTS
F_Q	2.79	2.32
F_H^N	1.58	1.55
T_{inlet}	539.5F	534.5F

DNBR is reduced by higher power, higher temperature, lower flow, or lower pressure conditions. The reduction in the F_Q, $F^N_{\Delta H}$ and T_{inlet} would cause the DNBR values to be higher than for Cycle 1. The methods used by Westinghouse for calculating DNBR for Cycle 1 are described in the FSAR Section 3.2.2. The Exxon Nuclear Company methods for calculating DNBR are documented in References C1 and C2.

The values for plant parameters (e.g., F_Q, etc.,) in the PTS document for Cycle 5 are more representative of current actual plant parameters though the assumed values are still quite conservative. As an example, the plant Technical Specifications restrict the maximum HFP F_Q to $\leq$ 2.21 and RCS flow to $\geq$ 190,800 gpm. Other factors, e.g., actual T_{inlet} ($\sim$530.5F), Temperature in the over temperature ΔT ($\sim$560.5F) reactor protection setpoint, assure that the actual plant operation is conservative with respect to the plant transient analysis assumptions. These aforementioned current specifications and operating practices assure that the actual DNBR margin would be significantly greater than that reported in the PTS document.

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2. Rod Ejection

In the rod ejection accident analysis, reported in XN-NF-78-47, the maximum expected RCS pressure during the transient was not included.

Based on a conservative assumption for the reactivity worth of the ejected rod, a Prairie Island specific plant transient simulation indicates that a volumetric expansion of $\Delta V = 112\ ft^3$ results for the primary coolant during the first 3 sec of the transient. With the pressurizer steam volume initially at 400 ft^3 (full power operation) and no credit taken for pressurizer spray and power operated relief valves, a pressurization calculation indicates that about 22 percent of the original steam mass may be lost through the pressurizer safety valves. The maximum pressure during this event is calculated to be below 2700 psia, which is less than the ASME Section III Service Level C limit as specified in the acceptance criteria of Section 15.4.8 of NUREG-75/087.

XN-NF-78-47, Section 7, described the results of the Rod Ejection Accident analysis. An EOC delayed neutron fraction of .0053 was used in calculating the pellet energy deposition from the ejected rod accident at EOC. This value is consistent with the cycle 5 EOC neutronics characteristics reported in Table 5.1 of the same document.

The generic rod ejection analysis as reported in XN-NF-78-44 was performed adiabatically; consequently, the moderator temperature coefficient was set to equal zero. The method is conservative with respect to peak deposited enthalpy, when compared to an analysis with heat conduction and a positive moderator temperature coefficient.

This conclusion is based upon review of the opposing effects of heat conduction and moderator feedback on the deposited enthalpy. As heat is absorbed by the moderator, a positive moderator temperature coefficient, as assumed at BOC, for Cycle 5, will increase the core reactivity and hence increase the power level versus time above that calculated adiabatically. As the power is diminished due to the scramming rods entering the core, heat conduction out of the fuel will decrease the deposited enthalpy. Thus, the peak deposited enthalpy occurs at approximately 1.5 seconds into the transient. For the adiabatic calculation, since no heat transfer is allowed, the peak deposited enthalpy occurs at the end of the transient (5.0 seconds).

By comparison, the effect of heat conduction lowers the peak deposited enthalpy enough to balance the effects of increased power due to positive moderator feedback. Thus, the adiabatic treatment predicts a higher deposited enthalpy than does the case where heat transfer and moderator feedback are allowed. For BOC, HZP transients the adiabatic method calculated as much as 14 cal/gm higher deposited enthalpy than an identical transient assuming heat transfer and a positive moderator temperature coefficient of +4.48 pcm/$^{\circ}$F. Hence, the generic rod ejection analysis provides a conservative evaluation of the rod ejection accident for the Prairie Island Unit 1 Cycle 5 reload where a positive moderator temperature coefficient is assumed at BOC.

Section 6.2 of XN-NF-78-44 describes the basic assumptions in the pressurization model. Some of these basic assumptions have been revised by ENC. Attachment B is included to show in detail the calculational steps for the updated example problem presented in Section 6.2. The values for the specific volume after compression and the final pressure shown in Attachment B differ from the XN-NF-78-44 values because some of the pressurizer water had originally been postulated to participate in the isentropic compression process resulting in a less conservative estimate of the overpressurization. The results presented in Attachment B present a more conservative estimate of overpressurization. ENC plans in the near future to issue a revision to XN-NF-78-44 to cover this change.

3. Small Steam Line Break

In the small steam line break analysis, the following initial conditions were used:

Reactor power:	Hot zero power
Pressurizer pressure:	2,280 psia
Pressurizer level:	20-percent
Reactor coolant flow:	Full nominal flow
Reactor inlet temperature:	547.3°F
Steam flow to turbine:	Zero
Steam generator level:	12.0 ft
Steam dome pressure:	1,102 psia
Initial steam flow through small break:	251 lb/sec
Moderator temperature reactivity feedback:	Figure 3.32 in XN-NF-78-35

The purpose of the small steam line break analysis is to demonstrate sufficient shutdown margin under the most unfavorable operating conditions for the case of the largest conceivable cooldown short of a steam line rupture. As the initial conditions show, the primary coolant inventory is minimized (lower pressurizer level), whereas the secondary coolant inventory is maximized, in order to maximize the cooldown for the reactor. These conditions are then combined with the most negative moderator temperature reactivity feedback (see steep negative slope in Figure 3.32 in XN-NF-78-35). The initial steamflow through the break is chosen to envelope the discharge flow from the largest single steam valve which is postulated to have failed full open. As shown in Figure 3.44 in XN-NF-78-35, a sufficient shutdown margin is maintained, even under these most conservative assumptions, until boric acid from the HPSIS (high pressure safety injection system) reaches the reactor core.

4. Large Steam Line Break

In the large steam line break analysis, the coolant temperature range experienced during the steamline break transient is much larger than during the full-power transients. The moderator and Doppler feedback effects are input as tabular values of reactivity versus temperature. The reactivity functions used are shown in Figures 3.31 and 3.32 in XN-NF-78-35. For the moderator reactivity, a factor of 1.2 has been applied to the original bounding nominal values. For the Doppler reactivity, values have been used which bound the expected reactivity feedback.

5. Locked Rotor

In XN-NF-78-35, it was stated that a statistical analysis was performed to determine that less than one percent of the fuel rods would experience DNB during the postulated locked pump rotor accident.

The following is a description of the numerical approach used:

1. The thermal margin (MDNBR) range from 1.5 down to 1.09 has been divided into segments, assuming a constant thermal margin within each segment for numerical integration purposes.

2. Using steady state thermal-hydraulics analyses and a core power map, the fuel rods were assigned to power groups corresponding to those thermal margin segments.

3. For the various rod power groups, the probability of experiencing DNB depending on the calculated W-3 MDNBR value was determined, using a probability plot which is based on the data on which the W-3 correlation is based.

4. In each of the defined thermal margin segments, the number of rods operating in that segment at the time of the lowest MDNBR value was multiplied by the corresponding probability of reaching DNB.

5. Summing up all those products performed under Paragraph 4 then results in the total number of rods expected to reach DNB.

References

C1 XN-74-5 "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS PWR), J D Kahn
C2 XN-75-48 "Definition and Justification of Exxon Nuclear Company DNB Correlation for Pressurized Water Reactors, K P Galbraith *et al*.

D. ECCS Analysis

Reference D1 described several errors in the TOODEE2 computer code. Reference D2 described the modifications required to correct these problems. ENC has considered the effects of the indicated TOODEE2 changes in conjunction with a Prairie Island exposure sensitivity study. These code changes were found to have approximately a $1^{\circ}F$ change in peak clad temperature.

The interim UPI model was described in XN-NF-78-46. In the interim UPI model obtained from the NRC, the steam source from upper plenum injection results from addition of three energy release rate terms which represent -

(1) the decay heat behind the top quench and cooldown between the quench fronts

(2) the energy release rates associated with rod quenching

(3) a negative heat release due to sensible heat required to raise the subcooled LPSI flow to saturation.

The quench energy release rate is approximated by an average expression which becomes:

$$\text{Average quench energy release rate} = \frac{\text{Constant}}{\text{QTIME}}$$

where QTIME is the time to quench the rods to the midplane as a function of UPI flow (given by the staff model). The constant in the above expression must then be:

$$\text{Constant} = \int_{\text{top of core}}^{\text{core midplane}} MC_p \; (T_{quench} - T_{sat})$$

or the energy released in quenching the core to the midplane. The term representing $(T_{quench} - T_{sat})$ from the NRC model was retained. The mass times heat capacity product was revised to represent the mass times heat capacity of the ENC fuel in the Prairie Island reactor from the top of the core to the midplane.

References

D1 N Lauben (NRC) to Z R Rosztoczy (NRC), dated December 15, 1978.
D2 Peter Bergquist *et al*, FV-78-0010/01, November 15, 1978.

E. Startup Physics Tests

Startup physics testing has been conducted for the Unit 1 and 2 initial startups and subsequent reactor startups after refueling to confirm conformance of the core parameters to design values or technical specification limits. Summary procedure descriptions have been provided in References E1, E2, and E3. The step-by-step operational procedures have been reviewed and startup testing observed by Region III I and E personnel in connection with various plant reloads (References E4). The procedures employed for reload startup testing have not substantially changed since the original procedures were developed for the first reloads. Any changes to these procedures are reviewed by the plant Operations Committee, a group of senior plant management, operations, and engineering personnel.

Acceptance criteria are established for each startup test for one of two reasons -

(1) To ensure that the core parameters are within an acceptable tolerance of design values, or

(2) To ensure that technical specification limits are met.

The acceptance criteria are similar to those specified for the original plant startups and are consistent with those accepted by the nuclear industry and the Regulatory Staff (Reference E5). The current acceptance criteria for Prairie Island startup physics testing are shown in Table E-1.

Table E-1
Startup Physics Test Acceptance Criteria

Test	Acceptance Criteria (Note 1)	Comments
Isothermal Temperature Coefficient	<0	-
Moderator Temperature Coefficient	<0	Note 2
Boron Worth	±10%	-
Differential & Integral Rod Worth		
Individual control bank	±15%	-
All control banks in	±10%	Note 3
Flux maps		
Relative Assembly Power	±10% ($\rho i \geq 0.9$)	-
	±15% ($\rho i < 0.9$)	-
Quadrant Tilt	1.02	-
Critical Boron Concentration	±.75%	-

Notes

1. Values stated as ± are relative to the cycle design value. Other values are absolute.
2. Measured at high power (>70%) at a stable xenon condition.
3. If the measured worth is more than 10% less than design, an N-1 measurement is performed.

The results of the zero power physics testing are reviewed by the Prairie Island Operations Committee prior to exceeding 5% power. This practice has been followed since the initial plant startup in 1973. Deviations from acceptance criteria are evaluated by the Operations Committee and appropriate action taken.

References

E1 Prairie Island Final Safety Analysis Report, Section 13.
E2 Unit 1 Startup Report for the Initial Plant Startup, L O Mayer (NSP) to E Case (NRC), October 31, 1974.
E3 Unit 2 Startup Report for the Initial Plant Startup, L O Mayer (NSP) to A Giambusso (NRC) May 15, 1975.
E4 NRC Inspection Reports 50-282/76-10, 50-282/76-22 (50-306/76-20), 50-282/76-19 (50-306/76-15), 50-282/77-08 (50-306/77-05), 50-282/77-17 (50-306/77-13), 50-306/78-02, 50-282/78-09, 50-282/78-23 (50-306/78-23).
E5 Paul Check (NRC) to Reactor Safety Branch (NRC), November 28, 1978.

F. Power Distribution Control

Attachment C contains revised pages TS -iv, 3.10-1, -1A, -2, and Figures TS 3.10-5 and -8.

These changes are proposed based on generic ENC-NRC discussions held on March 1, 1979 in regard to the PDC2 method. These changes include -

(1) Monthly confirmation of $F_Q^N(Z)$
(2) Definition of equilibrium condition and conditions for which monthly confirmation is to be conducted
(3) Setting - 3% as the target Δ I at EOC
(4) Revision of V(Z) curve based on analytical data
(5) Confirmation of $F_Q^N(Z)$ at the highest appropriate power
(&) Revision of K(Z) curve specifically for F_Q of 2.21

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ATTACHMENT B

to

Letter dated March 30, 1979

This attachment contains the step-by-step description of the method used to calculate the pressure rise in the RCS on the rod ejection accident calculation as described in XN-NF-78-44.

Rod Ejection Example Problem

Step 1 - Determine total energy released during rod ejection event. From XTRAN digital simulation the value stated in Table 6.1, Ref. 1, was obtained:

$$E = 8.342 \times 10^3 \text{ MW-sec} = 7.907 \text{ MBTu}.$$

Step 2 - Calculate the internal energy of the primary coolant before and after the ejection event. Hot standby conditions:

$$P = 2280 \text{ psia}, \ T = 547 \text{ F}.$$

From Ref. 2, Page 183, obtain enthalpy and specific volume:

$$h = 543.421 \text{ Btu/lb}$$

$$v_1 = 0.02124 \text{ ft}^3\text{/lb}.$$

With these values, the internal energy $u = h - Pv$ before the event is

$$u_1 = 534.451 \text{ Btu/lb}.$$

For a typical 4-loop plant, the mass of the primary coolant is

$$M_1 = 465.1 \times 10^3 \text{ lb}.$$

The internal energy of the primary coolant after the event is then

$$u_1^* = u_1 + E/M_1$$
$$u_1^* = 534.451 + 17.001 = 551.452 \text{ Btu/lb}.$$

Step 3 Calculate the liquid volume increase caused by the internal energy increase. An upper bound on the volume increase is obtained by ignoring the pressure effect on the specific volume. The following values are calculated from data on page 183, Reference 2:

T	h	v	u
560	559.650	0.021636	550.514
570	572.470	0.021958	563.198

These values are for P = 2280 psia. Interpolating these data for v at u_1^* = 551.452 Btu/lb results in

$$v_1^* = 0.021660 \text{ ft}^3/\text{lb}.$$

v_1^* = is the specific volume of the coolant after the rod ejection event. The volume increase of the primary coolant is then

$$\Delta V = (v_1^* - v_1)\, M_1$$

$$\Delta V = (0.021660 - 0.021240) \times 465.1 \times 10^3$$

$$\Delta V = 195.342 \text{ ft}^3.$$

Step 4 - Apply an isentropic compression by an amount of ΔV to the steam in the pressurizer.

A typical 4-loop plant has a pressurizer volume of 1800 ft^3. At hot standby conditions 20 percent of this is filled with water. The steam volume is then

$$V_g = 0.8 \times 1800 = 1440 \text{ ft}^3$$

From Page 90 of Reference 2 the values for specific volume and entropy for saturated steam at 2280 psia are obtained:

$$V_g = 0.15354 \text{ ft}^3/\text{lb}$$

$$s_g = 1.2591 \text{ Btu/(lb} \times \text{F)}$$

The mass of the steam is

$$M_g = 1440/0.15354 = 9378.7 \text{ lb}$$

The volume of the steam after compression is

$$V_1^* = 1244.658/9378.7 = 0.132711 \text{ ft}^3/\text{lb}.$$

The specific entropy after compression is the same as the one before compression:

$$s_1^* = s_1 = 1.2591 \text{ Btu/(lb} \times \text{F)}.$$

Step 5 - Bracket the pressure corresponding to s_1^* and v_1^* by two estimated values and interpolate linearly. For an estimated pressure and s_1^* the corresponding specific volume is calculated (from Reference 2). This specific volume will differ from v_1^* because the estimated pressure differs from the final pressure. For another estimated pressure and s_1^*, again the specific volume is calculated.

Interpolation for P at v = v_1^* results in the pressure solution.

Estimate P = 2700 psia

from Reference 2, page ;85:

for $s_1^* = 1.2591$ Btu/(lb x F), v = .134556 ft^3/lb

Estimate P = 2800 psia

from Reference 2, page 187:

for $s_1^* = 1.2591$ Btu/(lb x F), v = 0.130800 ft^3/lb

from

P	V
2700	0.134556
2800	0.130800

interpolation for P at $v = v_1^* = 0.132711$ ft^3/lb results in

P = 2749 psia.

This value is conservative because:

1. The liquid expansion of the primary system is overestimated.
2. No credit is taken for pressurizer relief valves, safety valves and spray.
3. All energy is instantly deposited in the primary liquid with no heat transfer to reactor structure, steam generator tubes, steam generator secondary side.

The value is below the ASME Section III Service Level C Limit.

References:

1) XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," January, 1979.

2) ASME Steam Tables, Third Edition, 1967.

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ATTACHMENT C

to

Letter dated March 30, 1979

Attachment C contains the following revised Technical Specification pages:

TS iv
TS 3.10-1
TS 3.10-1A
TS 3.10-2

Figure TS 3.10-5
Figure TS 3.10-8

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Reactivity

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

B. Power Distribution Limits

1. a. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits

$$F_Q^N \leq (2.145/P) \times K(Z) \quad \text{for } P > 0.5$$

$$F_Q^N \leq (4.29/P) \times K(Z) \quad \text{for } P \leq 0.5$$

$$F_{\Delta H}^N \leq 1.55\ (1 + 0.2(1-P))(1-RBP(BU))^1$$

b. $F_Q^N(Z)$ shall be measured at equilibrium conditions according to the following schedule:

(1) At the time of target flux difference determination, or

(2) At least once per 31 effective full-power days, or

(3) Upon reaching equilibrium conditions after exceeding by 10% or more of rated thermal power, the thermal power at which target flux difference was last determined, whichever occurs first

and must meet the following limit:

$$F_Q^N(Z) \leq (2.145/P^1) \times (K(Z)/V(Z)) \text{ for } P^1 \geq 0.50$$

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1. The (1-RBP(BU)) multiplier is only applicable for Westinghouse Fuel.

1. c. In Specification 3.10.B.1, the following definitions apply:

   (1) P is the fraction of full power at which the core is operating
   (2) K(Z) is the function given in Figure TS.3.10-5
   (3) Z is the core height location of F_Q^N
   (4) RBP(BU) is the Rod Bow Penalty as a function of region average burnup as shown in Figure TS.3.10-7
   (5) Region is defined as those assemblies with the same loading date
   (6) V(Z) is the function given in Figure TS.3.10-8
   (7) P_1 is the largest fraction of full power at which the plant will operate prior to the next target flux measurement.
   (8) The F_Q^N of b, above, is not applicable in the following core regions as measured in core height from the bottom of the fuel; the lower region from 0 to 10% inclusive, and the upper region from 90 to 100% inclusive.
   (9) Equilibrium conditions are defined as -
      (a) The delta flux difference shall be constant within $\pm$ 1% ΔI over the previous 24 hour period.
      (b) The power level shall be constant within $\pm$ 2% over the previous 24 hour period.

2. a. Following initial loading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

      1. The measured peaking factor, F_Q^N, shall be increased by five percent to account for measurement error.

      2. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

   b. If either measured hot channel factor exceeds its limit specified under 3.10.B.1.a, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q^N or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized up to 50% power for the purpose of physics testing. Identify and correct the cause of the out of limit condition prior to increasing thermal power above 50% power, thermal power may then be increased provided $F_Q(Z)$ is demonstrated through in-core mapping to be within its limits.

c. If the measured hot channel factor F_Q^N exceeds its limit as specified under 3.10.B.1.b, then one of the following actions shall be taken.

1. Within 48 hours, place the reactor in a configuration for which Specification 3.10.B.1.b is satisfied; or

2. Reduce thermal power by 1% for each percent that the measured F_Q^N exceeds the limit specified in 3.10.B.1.b. Thermal power may be increased to a power such that the associated F_Q^N would comply with 3.10.B.1.b.

3. The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target differences must be updated monthly. This may be done either by using the measured value for that month or by linear interpolation using the most recent measured value and a value of -3 percent at the end of the cycle life.

4. Except during physics tests, and except as provided by Item 5 through 8 below, the indicated axial flux difference for at least the number of operable excore channels required by TS.3.5 shall be maintained within a $\pm 5\%$ band about their target flux differences (defines the target band on axial flux difference).

5. At a power level greater than 90 percent of rated power, if the indicated axial flux difference of two operable excore channels deviates from its target band, either such deviation shall be eliminated, or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

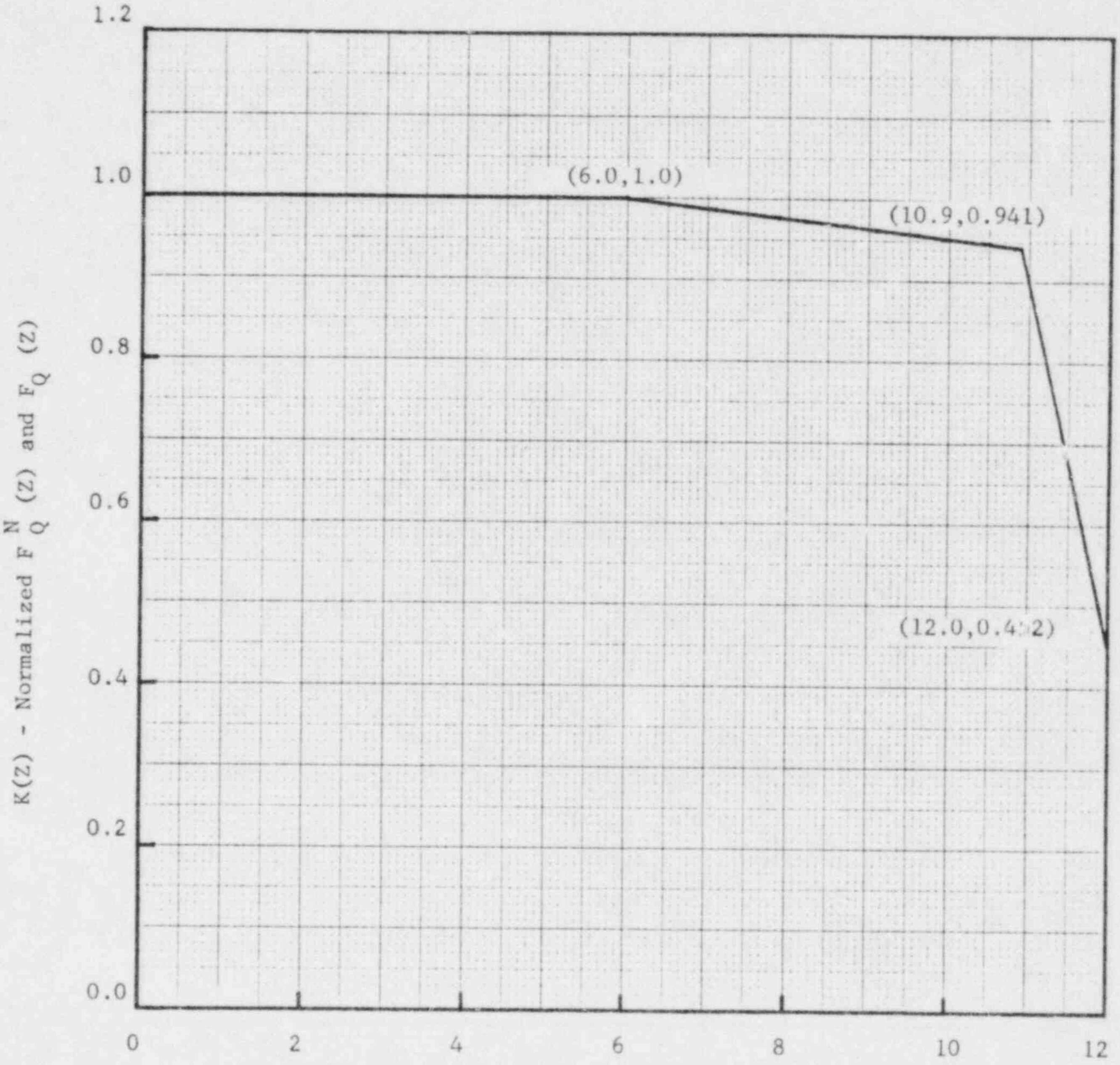
6. At a power level no greater than 90 percent of rated power,

   a. The indicated axial flux difference may deviate from its $\pm$ 5% target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference and the target flux difference does not exceed an envelope bounded by -10 percent and +10 percent at 90% power and increasing linearly to -25 percent and +25 percent at 50 percent power as shown in Figure TS.3.10-6.

   b. If 6.a is violated for two operable excore channels then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

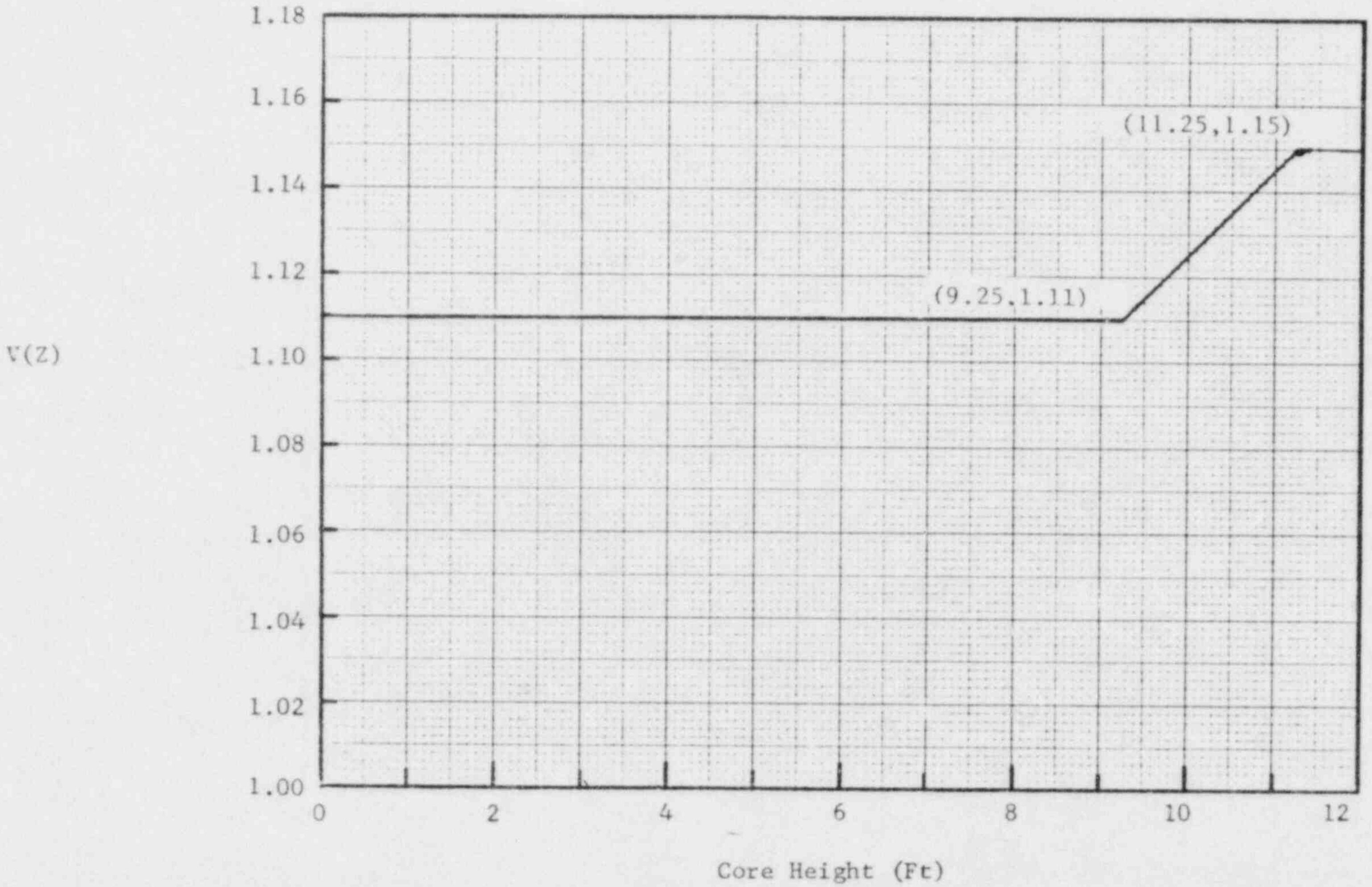
*May be extended to 16 hours during incore/excore calibration.

FIGURE TS.3.10-5
REV



HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE FOR F_Q = 2.21



V(Z) as a Function of Core Height

FIGURE TS.3.10-8
REV

DPR-42
DPR-60

ATTACHMENT D

to

Letter dated March 30, 1979

Attachment D contains affidavit of R. Nilson of Exxon Nuclear Company related to exemption from public disclosure of ENC documents XN-NF-79-6[P].

A F F I D A V I T

STATE OF Washington)
) ss.
COUNTY OF Benton)

I, Roy Nilson, being duly sworn, hereby say and depose:

1. I am Manager, Licensing, for Exxon Nuclear Company, Inc., ("ENC") and as such I am authorized to execute this Affidavit.

2. I am familiar with ENC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the document XN-NF-79-6(P), entitled "Exxon Nuclear Analysis of Power Distribution Measurement Uncertainty for Westinghouse PWRs," referred to as "Document", which is being submitted by Northern States Power Company in support of its Cycle 5 fuel reload application for the Prairie Island Nuclear Plant - Unit 1. Information contained in this Document has been classified by ENC as proprietary in accordance with the control system and policies established by ENC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by ENC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as being proprietary and confidential.

5. The Document has been made available to the United States Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of ENC and would be helpful to competitors of ENC when competing with ENC.

7. The information contained in the Document is considered to be proprietary by ENC because it reveals certain distinguishing aspects of reactor core modeling and statistical techniques which secure competitive economic advantage to ENC for fuel management and safety analysis optimization and improved marketability, and includes information utilized by ENC in its business which affords ENC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into ENC's reactor core modeling, statistical techniques and fuel management procedures and would result in substantial harm to the competitive position of ENC.

9. The Document contains proprietary information which is held in confidence by ENC and is not available in public sources.

10. In accordance with ENC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside ENC only as required and under suitable agreement providing for non-disclosure and limited use of the information.

11. ENC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis. Checks are made routinely to assure the policy procedures are being met.

12. This Document provides information which reveals reactor core modeling and statistical methods developed by ENC over the past several years. ENC has invested several hundred thousand dollars and many man-years of effort in the related core modeling and statistical techniques. Assuming a competitor had available the same background data and incentives as ENC, the competitor might, at a minimum cost, develop the information for the same expenditure of manpower and money as ENC.

13. Based on my experience in the industry, I do not believe that the background data and incentives of ENC's competitors are sufficiently similar to the corresponding background data and incentives of ENC that it is reasonable to expect such competitors would be in a position to duplicate ENC's proprietary information contained in the documents.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

SWORN TO AND SUBSCRIBED
before me this 9th day of
March, 1979.

Delores V. Lund
NOTARY PUBLIC

DPR-42
DPR-60

CONTAINS 10CFR 2.790 INFORMATION
PROPRIETARY

ATTACHMENT E

to

Letter dated March 30, 1979

Attachment E is the proprietary Exxon Nuclear Company report:

XN-NF-79-6 "Exxon Nuclear Analysis of Power Distribution Measurement Uncertainty for Westinghouse PWR's" February 1979