Docket Nos. 50-282/306

MAR 11 1977

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Northern States Power Company
ATTN: Mr. L. O. Mayer, Manager
Nuclear Support Services
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Gentlemen:

In response to your request for a license amendment dated November 4, 1976, and modified by your letter of January 28, 1977, the Commission has issued the enclosed Amendment Nos. 19 and 13 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments provide for additional DNBR margin to account for the effects of fuel rod bowing. During our review of your proposed request we found that certain changes were necessary for clarification. Your staff has agreed to these changes and they have been incorporated.

In addition certain minor administrative errors in earlier amendments have been corrected. Some of these errors were noted in your January 4, 1977 amendment request. The balance of that request is still under review.

Copies of our related Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 19 and 13 to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice

cc w/enclosures: see next page

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SURNAME	MGrotenhuis:ah	MHFletcher	RMDiggs	RBaer	[illegible]	DLZiemann
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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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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. 19
License No. DPR-42

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated November 4, 1976, as modified by filing dated January 28, 1977, 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 (i) 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 License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 19, 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

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 11, 1977

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. 13
License No. DPR-60

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated November 4, 1976, as modified by filing dated January 28, 1977, 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 (i) 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 License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 13 , 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

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1977

ATTACHMENT TO LICENSE AMENDMENT NOS. 19 AND 13

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated licenses with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
iii	iii
iv	iv
3.6-3A	3.6-3A
3.9-9	3.9-9
3.10-1	3.10-1
3.10-7	3.10-7
	3.10-7A (new page)
3.10-13	3.10-13
	3.10-13A (new page)
4.14-1	4.14-1

LIST OF TABLES

LIST OF TABLES (contd)

LIST OF FIGURES

E. Emergency Air Treatment Systems

1. Except as specified in Specification 3.6.E.3 and 3.6.E.6 below, all trains of the Shield Building Ventilation System, the Auxiliary Building Special Ventilation System, the Spent Fuel Pool Special Ventilation System and the diesel generators required for their operation shall be operable at all times.

2. a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show ≥99% DOP removal for particles having a mean diameter of 0.7 microns and ≥99% halogenated hydrocarbon removal.

   b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

   c. The Spent Fuel Pool Special Ventilation System fans only shall operate within ±10% of 4000 cfm per train.

3. From and after the date that one train of the Shield Building Ventilation System or one train of the Auxiliary Building Special Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days (unless such train is made operable), provided that during such seven days the redundant train is verified to be operable daily.

4. If the conditions for operability of the Shield Building Ventilation System cannot be met, procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not required for the affected unit.

5. If the conditions for operability of the Auxiliary Building Special Ventilation System cannot be met, procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not required in either unit.

Basis

It is expected that the releases of radioactive materials in liquid waste will be kept within the design objective levels and will not exceed on an instantaneous basis the concentration limits specified in 10 CFR Part 20. These levels provide reasonable assurance that the resulting annual exposure to the whole body or any organ of an individual will not exceed 5 millirems per year. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that using this operational flexibility under unusual operating conditions, the licensee shall exert every effort to keep levels of radioactive material in liquid wastes as low as practicable and that annual releases will not exceed a small fraction of the annual average concentration limits specified in 10 CFR Part 20.[1]

Liquid radwaste leaving the plant is mixed with cooling tower blowdown flow (150 cfs) before entering the discharge canal where it is further mixed with water (860 cfs at low river flow) entering the canal from Sturgeon Lake. This total dilution flow of 1010 cfs (452,000 gpm) results in a dilution factor of 2.2×10^{-6} min/gal which applies at the point of discharge to the main flow of the Mississippi River. The volume of liquid discharged, the actual dilution flow, and analysis of the proportional composite sample provide the basis[2] for reporting the quantity and concentration of activity released.

The operating manual will identify all equipment installed in the liquid waste handling and treatment systems and will specify detailed procedures for operating and maintaining this equipment.

It is expected that the releases of radioactive materials in gaseous waste will be kept within the design objective levels and will not exceed 10 millirems per year at the site boundary. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operation conditions which may temporarily

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

$F_Q^N(Z) \leq (2.09/P) \times K(Z)$ for $P > .5$

$F_Q^N(Z) \leq (4.18) \times K(Z)$ for $P \leq .5$

$F_{\Delta H}^N \leq 1.52\ (1 + 0.2\ (1-P))$

where P is the fraction of full power at which the core is operating. K(Z) is the function given in Figure TS.3.10-5 and Z is the core height location of F_Q^N.

2. 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,

J. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of rated power.

K. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq$ 564°F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 190,800 gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

Basis

Design criteria have been chosen for Condition I and II events which are consistent with the fuel integrity analyses of Section 3.2 of the FSAR. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.(2)

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the Final Acceptance Criteria (FAC) limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. F_Q is the product of F_Q^N and F_Q^E.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

F_Q^N is the Nuclear Hot Channel Factor defined as the maximum local neutron flux in the core divided by the average neutron flux in the core.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

The rod position indicator channel is sufficiently accurate to detect a rod ± 7 inches away from its demand position. A misalignment less than 15 inches does not lead to over-limit power peaking factors. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with the safety analysis.

A recent evaluation of DNB test data from experiments of fuel rod bowing in subchannels containing thimble cells has identified that it is appropriate to impose a penalty factor to the accident analyses DNBR results. This evaluation has not been completed, but to assure that this effect is accommodated in a conservative manner, an interim thimble cell rod bow penalty as a function of fuel burnup, is applied to the measured values of the enthalpy rise hot channel factor, $F^N_{\Delta H}$. The rod bow penalty is partly accommodated by excess reactor coolant flow. This flow rate shall be verified by calorimetric flow data and/or by elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred.

References

(1) FSAR Section 3.2.1
(2) FSAR Section 3.2.2
(3) FSAR Appendix F
(4) WCAP 8091
(5) FSAR Section 13A

4.14 CONTROL ROOM AIR TREATMENT SYSTEM TESTS

Applicability

Applies to the periodic testing requirements for the Control Room Special Ventilation System.

Objective

To specify tests for assuring the operability of the Control Room Special Ventilation System.

Specification

A. At least once per operating cycle or once every 18 months, whichever occurs first, the following shall be demonstrated:

1. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate (±10%).

2. Automatic initiation of the Control Room Special Ventilation System shall be demonstrated with a simulated high radiation or Safety Injection signal.

B. 1. The tests of Specification 3.13.B. shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

2. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could effect the HEPA bank bypass leakage.

3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

4. Each circuit shall be operated at least 15 minutes every month.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 19 AND 13 TO FACILITY 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 November 4, 1976, and modified by letter of January 28, 1977, Northern States Power Company (the licensee) requested an amendment to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The proposed amendments would provide additional DNB margin to account for the effects of rod bowing. During our review of the proposed changes, we found that certain modifications to the proposal were necessary to meet NRC requirements. These changes were discussed with the licensee's staff. The licensee has agreed with these changes and the changes will be incorporated into the amendment. In addition, certain minor administrative errors in previous amendments would be corrected.

DISCUSSION

On August 9, 1976, Westinghouse Electric Corporation presented data to the NRC staff which showed that previously developed methods for accounting for the effect of fuel rod bowing on departure from nucleate boiling may not contain adequate thermal margin when unheated rods (such as thimble tubes) are present. We have evaluated the impact of the Westinghouse data on all operating pressurized water reactors (PWR's). Models for treating the effects of fuel rod bowing on thermal-hydraulic performance have been derived for all PWR's. The models are based on the propensity of the individual fuel designs to bow and on the thermal analysis methods used to predict the coolant conditions for both normal operation and anticipated transients. As a result of these evaluations, the staff has concluded that for some facilities the current technical specification operating limits do not provide sufficient thermal margin. In these cases, additional thermal margin is required to assure, with high confidence, that departure from nucleate boiling (DNB) does not occur during anticipated transients.

To accommodate the loss of thermal margin for the Prairie Island facility, the licensee has proposed, by letter dated November 4, 1976, as modified by letter dated January 28, 1977, to change the Technical Specifications requirements.

BACKGROUND

In 1973 Westinghouse Electric presented to the staff the results of experiments in which a 4 x 4 bundle of electrically heated fuel rods was tested to determine the effect of fuel rod bowing to contact on the thermal margin. The departure from nucleate boiling ratio (DNBR) is a measure of the thermal margin available prior to the point at which DNB occurs. The tests were performed at conditions representative of PWR coolant conditions. The results of these experiments showed that, for the highest power density at the highest coolant pressure expected in a Westinghouse reactor, the DNBR reduction due to a heated rod bowed to the point of contact with adjacent heated rods was approximately 8%.

Fuel bundle coolant mixing and heat transfer computer programs such as COBRA IIIC and THINC-IV were able to predict the results of these experiments. Because the end point could be predicted, i.e., the DNBR reduction at contact, there was confidence that the DNBR reduction due to partial rod bow, that is, rod bow to a point less than contact with the adjacent rod, could also be correctly predicted.

On August 9, 1976, Westinghouse met with the NRC staff to discuss further experiments with the same configuration of fuel bundle (4 x 4) using electrically heated rods. However, for this set of experiments one of the center 4 fuel rods was replaced by an unheated tube of the same size as a Westinghouse thimble tube. This new test configuration was tested over the same range of power, flow and pressure as the earlier tests. However, with the unheated, larger diameter rod the reduction in DNBR was much larger than in the earlier (1973) tests.

The data consisted of points corresponding to no intentional bowing (that is, a certain amount of bowing due to tolerances cannot be prevented) and to contact. No data were taken at partial clearance reductions between rods.

The staff attempted to calculate the Westinghouse results with the COBRA IIIC computer code but could not obtain agreement with the new data. Westinghouse was also unable to obtain agreement between their experimental results and the THINC-IV computer code.

On August 19, 1976, Combustion Engineering (CE) presented results of similar experiments to the staff. These tests were performed using a 21-rod bundle of electrically heated rods and an unheated guide tube. Results were presented for not only the case of full contact, but also the case of partial bowing.

Both sets of data (Westinghouse and CE) showed similar effects due to variations in coolant conditions. For both cases, the DNBR reduction became greater as the coolant pressure and the rod power increased.

Because both sets of data showed that plant thermal margins might be less than those intended, the staff derived an interim model to conservatively predict the DNBR reduction. Since the data with unheated rods could not be predicted by existing analytical methods, empirical models were derived[1]. Using these empirical models, the staff calculated DNBR reductions to be applied to all operating pressurized water reactors. The staff has permitted the calculated reduction in DNBR to be offset by certain available thermal margins on a case-by-case basis. These "credits" may be either generic to a given fuel design or plant specific. As described in reference 1, the generic fuel design credits for PINGP Unit Nos. 1 and 2 are:

1. Design pitch reduction (1.8%)
2. Thermal diffusion coefficient (1.7%)
3. Critical heat flux correlation statistics (2.6%)
4. Densification power spike factor effect on DNB (3.9%)

EVALUATION

The licensee has proposed Technical Specification changes which would provide for additional DNBR margin to offset the reduction in DNBR due to rod bow. The credits which the licensee has taken to offset the DNBR penalty are:

1. Excess flow capacity - 7.2%

2. $F_{\Delta H}$ limit - 3.6%

The staff has evaluated the proposed Technical Specification changes using the procedure given in reference 1 and concluded that the reduction in $F_{\Delta H}$ limits and credits for excess flow are adequate to offset the loss of thermal margin indicated by the recent Westinghouse rod bow data; and, therefore, the proposed changes are acceptable.

(1) Revision 1 to Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations, dated February 16, 1977.

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 these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment:
Revision 1 to Interim Safety Evaluation Report
on Effects of Fuel Rod Bowing on Thermal
Margin Calculations, dated February 16, 1977

Date: March 11, 1977

ATTACHMENT

INTERIM SAFETY EVALUATION REPORT ON EFFECTS OF FUEL ROD BOWING ON THERMAL MARGIN CALCULATIONS FOR LIGHT WATER REACTORS

(REVISION I)

February 16, 1977

CONTENTS

1.0 Introduction

Data have recently been presented (Reference 1) to the staff which show that previously developed methods for accounting for the effect of fuel rod bowing on departure from nucleate boiling in a pressurized water reactor (PWR) may not contain adequate thermal margin when unheated rods, such as instrument tubes, are present. Further experimental verification of these data is in progress. However an interim measure is required pending a final decision on the validity of these new data.

The staff has evaluated the impact of these data on the performance of all operating pressurized water reactors. Models for treating the effects of fuel rod bowing on thermal-hydraulic performance have been derived. These models are based on the propensity of the individual fuel designs to bow and on the thermal analysis methods used to predict the coolant conditions for both normal operation and anticipated transients. As a result of these evaluations the staff has concluded that in some cases sufficient thermal margin does not now exist. In these cases, additional thermal margin will be required to assure, with high confidence, that departure from nucleate boiling (DNB) does not occur during anticipated transients. This report discusses how these conclusions were reached and identifies the amount of additional margin required.

The models and the required DNBR reductions which result from these models are meant to be only an interim measure until more data are available. Because the data base is rather sparse, an attempt was made to treat this problem in a conservative way. The required DNBR reductions will be revised as more data become available.

The staff review of the amount and consequences of fuel rod bowing in a boiling water reactor is now underway. At present no conclusions have been reached. When this review reaches a stage where either an interim or final conclusion can be reached, the results of this review will be published in a separate safety evaluation report.

It should be noted that throughout the remainder of this report, all discussion and conclusions apply only to pressurized water reactors.

2.0 DNBR Reduction Due To Rod Bow

2.1 Background

In 1973 Westinghouse Electric presented to the staff the results of experiments in which a 4x4 bundle of electrically heated fuel rods was tested to determine the effect of fuel rod bowing to contact on the thermal margin(DNBR reduction) (Reference 2). The tests were done at conditions representative of PWR coolant conditions. The results of these experiments showed that, for the highest power density at the highest coolant pressure expected in a Westinghouse reactor,the DNBR reduction due to heated rods bowed to contact was approximately 8%.

Fuel bundle coolant mixing and heat transfer computer programs such as COBRA IIIC and THINC-IV were able to accurately predict the results of these experiments. Because the end point could be predicted, i.e., the DNBR reduction at contact,there was confidence that the DNBR reduction due to partial bow, that is, bow to less than contact could also be correctly predicted.

On August 9, 1976 Westinghouse met with the staff to discuss further experiments with the same configuration of fuel bundle (4x4) using electrically heated rods. However, for this set of experiments one of the center 4 fuel rods was replaced by an unheated tube of the same size as a Westinghouse thimble tube. This new test configuration was tested over the same range of power, flow and pressure as the earlier tests. However, with the unheated, larger diameter rod the reduction in DNBR was much larger than in the earlier (1973) tests.

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The staff attempted to calculate the Westinghouse results with the COBRA IIIC computer code but could not obtain agreement with the new data. Westinghouse was also unable to obtain agreement between their experimental results and the THINCIV computer code.

On August 19, 1976 CE presented results of similar experiments to the staff. These tests were performed using a 21 rod bundle of electrically heated rods and an unheated guide tube. Results were presented for not only the case of full contact, but also the case of partial bowing.

Both sets of data (Westinghouse and CE) showed similar effects due to variations in coolant conditions. For both cases, the DNBR reduction became greater as the coolant pressure and the rod power increased.

Because both sets of data showed that plant thermal margins might be less than those intended, the staff derived an interim model to conservatively predict the DNBR reduction. Since the data with unheated rods could not be predicted by existing analytical methods, empirical models were derived. These models give the reduction in DNBR as a function of the clearance reduction between adjacent fuel rods. Two such models were derived, one based on the Westinghouse data and one based on the CE data.

2.2 Model Based on Westinghouse Data

As stated in Section 2.1, data were presented by Westinghouse for the DNBR reduction at full contact and with no bow. No data at partial gap closure were presented. Westinghouse proposed, and the staff accepted, a straight line interpolation between these two points as shown in Figure 2.1.

This approach is conservative if the DNBR reduction does not increase more rapidly than the straight line reduction shown in Figure 2.1. Although the data for DNBR reduction due to rod bowing in the presence of an unheated fuel rod cannot be predicted by existing analytical methods, one would nevertheless expect that the actual behavior would more nearly follow the curved line also shown in Figure 2.1. According to this curved line, the DNBR would be reduced gradually for small amounts of bow. As the fuel rods (or fuel rod and unheated rod) become close enough so that there is an interaction, the DNBR would decrease more rapidly. No physical mechanism has been postulated which would lead to sudden large decreases in the DNBR for small or moderate gap closures. Thus, the straight line approximation is believed to be an overestimate of the expected behavior.

Experience with critical heat flux tests also supports the assumption of a small reduction in DNBR for small amounts of fuel rod bowing. Experimental measurements of critical heat flux done on test assemblies always have some amount of rod bowing. This may be due simply to fabrication tolerances or to electromagnetic attraction forces set up between electrically resistance heated rods which simulate fuel rods.

It should be noted that this behavior (little or no reduction in DNBR for small amount of bowing) is shown by Combustion Engineering data which became available to the staff after the Westinghouse model was derived. The Combustion Engineering data is discussed in Section 2.3 and the model derived from this data is shown in Figure 2.2.

All manufacturers of reactor cores, including Westinghouse, include a factor in their initial core design to account for the reduction in DNBR that may result from pitch reduction from fabrication tolerances and initial rod bow. The amount of this pitch reduction factor varies with the fuel design and the analysis methods which are used. For any particular core this factor is not varied as a function of burnup.

In developing the interim rod bow penalties described in this report, it became apparent that the penalty should be a function of burnup since the magnitude of rod bow is a function of burnup. However, to maintain existing thermal margins early in core life when only a small amount of fuel rod bow is anticipated, the initial pitch reduction factor was included until such time as the rod bow DNBR reduction became greater. This is represented as the straight horizontal line on Figure 2.1.

2.3 Combustion Engineering Model

Combustion Engineering performed experiments to determine the effect of rod bowing on DNBR which included some cases in which the effect of partial bowing as well as bowing to contact was determined. Again, a straight line interpolation is used. However, the point of zero DNBR reduction is not at zero clearance reduction but rather, at an intermediate value of clearance reduction. This is shown schematically in Figure 2.2. The horizontal straight line, representing the initial pitch reduction factor is included as explained previously in Section 2.2

2.4 Models for Babcock and Wilcox and Exxon

On August 17, 1975 representatives of Babcock and Wilcox met with the staff to discuss this problem. Babcock and Wilcox did not present any data on the effects of rod bowing on DNBR. They had previously presented data to the staff on the amount of bowing to be expected in Babcock and Wilcox 15x15 fuel assemblies. Because Babcock and Wilcox had no data on the effect of rod bow on DNBR, the staff applied the Westinghouse model to calculate the effect of rod bowing on DNBR for Babcock and Wilcox fuel. This is acceptable since the conditions of operation are nearly the same in pressurized water reactors from both vendors and the fuel bundle designs are similar.

The amount of fuel rod bowing as a function of burnup was calculated using the Babcock and Wilcox 15x15 fuel bundle data.

Representatives of the Exxon Nuclear Corporation discussed the effects of fuel rod bowing in the presence of an unheated rod on DNBR with the staff on August 19, 1976. Exxon has not performed DNB tests with bowed rods and thus has no data pertinent to this problem. The first cycle of Exxon fuel has just been removed from H. B. Robinson and the results of measurements on the magnitude of rod bowing have not yet been presented to the staff. The effects of fuel rod bowing for Exxon fuel were evaluated on a plant by plant basis as discussed in Section 4.0

2.5 Application of the Rod Bow/DNBR Model

Using these empirical models, the staff derived DNBR reductions to be applied to both operating reactors and plants in the Operating License review stage. The procedure in applying these empirical models is as follows:

Step 1: Predict the clearance reduction due to rod bow as a function of burnup. An expression of the form

$$\frac{\Delta C}{C_o} = a+b\sqrt{BU}$$

is used where

$\frac{\Delta C}{C_o}$ = fractional clearance reduction due to rod bowing

a,b = empirical constants obtained for a given fuel design

BU = burnup (region average or bundle average, depending on the fuel designer).

Westinghouse showed in Reference 6 that an equation of the above form fit the rod bow data from 26 fuel regions. The constant a represents the initial bow of the fuel rods due to fabrication tolerance. The staff has approved the above equation (Reference 8).

Also included in the constants a and b is a factor of 1.2 to convert from the cold conditions at which the measurements were made to the hot operating conditions and a factor of 1.645 which, when multiplied by the standard deviation, gives an amount of bow greater than that expected from 95% of the fuel rods with a 95% confidence.

Step 2: Apply the previously discussed empirical models of DNBR reduction as a function of clearance reduction using the value of $\Delta C/c_o$ calculated from step 1.

Step 3: The staff has permitted the reduction in DNBR calculated in step 2 to be offset by certain available thermal margins. These may be either generic to a given fuel design or plant dependent.

An example of a generic thermal margin which would be used to offset the DNBR reduction due to rod bow is the fact that the DNBR limit of 1.30 is usually greater than the value of DNBR above which 95% of the data lie with a 95% confidence. The difference between 1.30 and this number may be used to offset the DNBR reduction.

For Westinghouse 15x15 fuel, the value of DNBR which is greater than 95% of the data at a 95% confidence level is 1.24 (Reference 1). For Westinghouse 17x17 fuel this number is 1.28 (Reference 1). A review of the data used to derive these numbers shows that the use of three significant figures is justified.

An example of a plant specific thermal margin would be core flow greater than the value given in the plant Technical Specifications.

A discussion of the application of this method to Construction Permit and Operating License reviews is given in Section 3.0. A discussion of the application and the results of this method to operating reactors is given in Section 4.0. The application to reactors using Exxon fuel is also discussed in Section 4.0.

3.0 Application to Plant in Construction Permit And Operating License Review Stage

3.1 CP Applications

No interim rod bow DNB penalties should be applied to CP applications. The rod bow data upon which the interim limits have been based should be considered preliminary. There is sufficient time available to review the data and assess a penalty, if any, prior to the OL stage. We will advise each CP applicant of the nature of interim penalties being applied to OL reviews and operating reactors.

As stated above, the data used to evaluate the effects of rod bow on DNBR are preliminary. They are also incomplete. In order to assess the conservatism of the straight line approximation and to obtain data on designs for which no data is now available we will require the applicant to (1) fully define the gap closure rate for prototypical bundles and (2) determine by an appropriate experiment the DNB effect that bounds the gap closure from part (1). Such requirements will be part of our CP review effort.

3.2 OL Applications

Plants which are in the operating license review stage should consider a rod bow penalty. This penalty should be as described in Section 2.2 for Westinghouse or Section 2.3 for Combustion Engineering. Babcock and Wilcox plants should use the rod bow vs. burnup curve appropriate to their fuel and the Westinghouse curve of DNBR reduction as a function of rod bow.

All applicants may propose appropriate thermal margins (as discussed in Section 2.4) to help offset the calculated DNBR reduction.

4.0 Application To Operating Reactors

This section divides the operating plants into distinct categories and lists them according to the fuel and/or reactor manufacturer. Operating plants which cannot be so categorized (such as plants with fuel supplied by more than one vendor) are placed in a separate category. The plants assigned to each category are listed in the appropriate subsection.

The conclusions reached in this section are in some cases dependent on conditions or analysis which are valid only for the present fuel cycle. Hence, the FΔH or DNBR reductions which are given (or the fact that no such reduction is concluded to be required) is valid only for the present operating cycle.

4.1 Westinghouse LOPAR Fuel

The designation LOPAR stands for low parasitic and refers to the fact that the guide tubes in the fuel bundle are made of Zircaloy. Table 4.1 gives a list of the operating plants which fall into this classification.

TABLE 4.1: PLANTS WHICH CURRENTLY USE THE WESTINGHOUSE LOPAR FUEL ASSEMBLY

15 x 15	17 x 17
Zion 1 Cycle 2	Trojan Cycle 1
Zion 2 Cycle 1	Beaver Valley 1 Cycle 1
Indian Point 3 Cycle 1	
Turkey Point 3 Cycle 4	
Turkey Point 4 Cycle 3	
Prairie Island 2 Cycle 2	
Prairie Island 1 Cycle 2	

TABLE 4.1 (cont.)

15 x 15

Surry 1 Cycle 4

Surry 2 Cycle 3

Kewaunee Cycle 2

Point Beach 1 Cycle 5

Point Beach 2 Cycle 3

The reduction in DNBR due to fuel rod bowing is assumed to vary linearly with the reduction in clearance between the fuel rods (or fuel rod and thimble rod) according to the model discussed in Section 2.2.

The maximum value of DNBR reduction (at contact), obtained from the experimental data was used to calculate the DNBR reduction vs. bow for the 15x15 LOPAR fuel. This DNBR contact reduction was adjusted for the lower heat flux in the 17x17 LOPAR fuel.

The clearance reduction is conservatively assumed to be given by the following equation for the 15x15 (and 14x14) fuel,

$$\frac{\Delta C}{Co} = a + b\sqrt{Bu}$$

where $\frac{\Delta C}{Co}$ is the reduction in clearance

Bu is the region average burnup

and a,b are empirical constants fitted to Westinghouse 15x15 rod bow data

For the 17x17 LOPAR fuel, the clearance reduction was calculated from the equation:

$$\Delta C/Co = \left(\frac{\Delta C}{Co}\right)_{15x15} \times \left(\frac{L}{I}\right)_{15x15} \times \left(\frac{I}{L}\right)_{17x17}$$

where L = the distance between grids

I = moment of inertia of fuel rod

On December 2, 1976, Westinghouse informally showed the staff new data pertaining to the magnitude of rod bow as a function of region average burnup in 17x17 fuel assemblies. This data show that the above correction is probably conservative and that the magnitude of fuel rod bowing in 17x17 fuel rods can better be represented by an empirical function. This review is now underway.

The calculated DNBR reduction is partially offset by existing thermal margins in the core design. For the Westinghouse LOPAR fuel design some or all of the following items were used in calculating the thermal margin for the operating plants:

- design pitch reduction
- conservatively chosen TDC used in design*
- Critical heat flux correlation statistics (assumed in thermal analysis safety calculations) are more conservative than required.
- Densification power spike factor included although no longer required (Reference 4)

After taking these factors into account, the reductions in $F_{\Delta H}$ shown in Table 4.2 were found necessary. All operating plants listed in Table 4.1 will be required to incorporate these reductions in $F_{\Delta H}$ into their present operating limits.

*TDC (thermal diffusion coefficient) is a measure of the amount of mixing between adjacent subchannels.

TABLE 4.2: F∆H REDUCTION FOR WESTINGHOUSE LOPAR FUEL

CYCLE	REDUCTION IN F∆H (%)		
	15x15	17x17	ZION 1&2
1st Cycle (0-15 Gwd*/MTU)	0-2 ramp	0-9.5	0-6 ramp
2nd Cycle (15-24 Gwd*/MTU)	4	12	8
3rd Cycle (24-33 Gwd*/MTU)	6	12	10

These reductions in F∆H may be treated on a region by region basis. If the licensee chooses, credit may be taken for the margin between the actual reactor coolant flow rate and the flow rate used in safety calculations. Credit may also be taken for a difference between the actual core coolant inlet temperature and that assumed in safety analyses. In taking credit for coolant flow or inlet temperature margin, the associated uncertainties in these quantities must be taken into account.

4.2 Westinghouse HIPAR and Stainless Steel Clad Fuel

The designation HIPAR stands for high parasitic and refers to the fact that the guide tubes in the fuel bundle are made of stainless steel. These two fuel types, HIPAR and Stainless Steel clad, are grouped together because the amount of bowing expected (and observed) is significantly less than that in the observed Westinghouse LOPAR fuel. The plants which fall under this classification are listed in Table 4.3.

$*\ \frac{Gwd}{MTU} = 1000\ \frac{Mwd}{MTU}$

TABLE 4.3: HIPAR AND STAINLESS STEEL PLANTS

Ginna	Indian Point 2
San Onofre	Connecticut Yankee

The model for the reduction in DNBR due to fuel rod bowing is assumed to be identical to that used for the LOPAR fuel. This is acceptable since cladding material should have no effect on CHF (critical heat flux) and the same DNB correlation applies to both HIPAR and LOPAR grids.

For reactors in this category, the peak reduction in DNBR (corresponding to 100% closure) was adjusted to correspond to the peak overpower heat flux of that particular reactor.

The amount of rod bowing for the plants listed in Table 4.3 which use HIPAR and stainless steel fuel, was calculated by means of an adjustment to the 15x15 LOPAR formula. This adjustment took the form of the ratio

$$\frac{\text{amount of bow for assembly type}}{\text{amount of bow for LOPAR fuel}} = \frac{(L/IE)\ \text{assy type}}{(L/IE)\ \text{LOPAR}}$$

where L is the span length between grids

I is the moment of inertia of the fuel rod

E is the modulus of elasticity of the fuel rod cladding

Ginna Cycle 6

The Ginna plant is fueled with 121 fuel assemblies. Two of these are Exxon assemblies, and two are B&W assemblies. The remainder are Westinghouse HIPAR fuel assemblies. The experimental value of DNBR reduction was adjusted for heat flux and pressure from peak experimental to actual plant conditions. Ginna took credit for the thermal margins due to pitch reduction, design vs. analysis values of TDC and fuel densification power spike. These thermal margins offset the calculated DNBR reduction so that no reduction in F∆H is required.

San Onofre Cycle 5

San Onofre is fueled with 157 bundles of 15x15 stainless steel clad fuel. An F∆H of 1.55 was used in thermal design and in the Technical Specifications. To offset the reduction in F∆H due to rod bowing San Onofre has proposed taking credit for margin available from the assumed worst case axial power distribution used in the thermal analysis for San Onofre and that which would be possible during operation. This proposal is now being reviewed by the staff.

Indian Point 2 Cycle 2

Indian Point 2 is fueled with HIPAR fuel bundles. The experimental value of DNBR reduction was adjusted for heat flux and pressure to actual plant conditions. Indian Point Unit 2 had thermal margin to offset this DNBR reduction in pitch reduction, design vs. analysis values of TDC, fuel densification power spike and a value of F∆H of 1.65 used in the design (vs. 1.55 in the Tech Spec). Therefore, no reduction of F∆H is required for Indian Point Unit 2.

Connecticut Yankee Cycle 7

Connecticut Yankee is fueled with 157 stainless steel clad fuel assemblies. The DNBR reduction at contact was assumed to be that used for the Westinghouse LOPAR 15x15 fuel. No adjustment was made for heat flux. The value of pressure was adjusted to the overpressure trip set point value of 2300 psi. Full closure will not occur in stainless steel fuel out to the design burnup.

Connecticut Yankee has sufficient thermal margin in variable overpressure and overpower trip set points to accommodate the calculated DNBR reduction. Therefore no penalty is required.

4.3 Babcock and Wilcox 15x15

The reactors listed in Table 4.4 are fueled with B&W fuel.

TABLE 4.4: REACTOR USING B&W FUEL

Oconee 1 Cycle 3
Oconee 2 Cycle 2
Oconee 3 Cycle 1
Rancho Seco
Three Mile Island 1 Cycle 2
Arkansas 1 Cycle 1

Babcock and Wilcox met with the staff on September 8, 1975 and presented data on the amount of rod bow in B&W fuel. The staff derived a model for B&W 15x15 fuel based on this data. This model has the form:

$$\frac{\Delta C}{Co} = a + b\sqrt{Bu}$$

where $\frac{\Delta C}{Co}$ is the fractional amount of closure

Bu is the bundle average burnup

and a,b are empirical constants fitted to B&W data

The reduction in DNBR due to fuel rod bowing is assumed to vary linearly with the reduction in clearance between the fuel rods (or fuel rod and thimble rod) but can never be lower than that due to the pitch reduction factor used in thermal analysis, as explained in Section 2.2.

Babcock and Wilcox claimed and the staff approved credit for the following thermal margins:

- Flow Area (Pitch) reduction
- Available Vent Valve credit
- Densification Power Spike removal
- Excess Flow over that used in safety analyses
- Higher than licensed power used for plant safety analyses

Based on this review and the thermal margins presented by B&W to offset the new Westinghouse data, Rancho Seco is the only plant for which a reduction in DNBR is required. Table 5 gives the values for the reduction of DNBR required at this time.

TABLE 5: DNBR REDUCTIONS FOR B&W PLANTS

Burnup	DNBR Reduction Rancho Seco
Cycle 1 (0-15 $\frac{Gwd}{MTU}$)	0
Cycle 2 (15-24 $\frac{Gwd}{MTU}$)	1.6%
Cycle 3 (24-33 $\frac{Gwd}{MTU}$)	3%

Plans must be submitted to the staff to establish how these reductions in DNBR will be accommodated.

4.4 Combustion Engineering 14x14

Combustion Engineering has presented data to the staff on the amount of rod bowing as a function of burnup. (Reference 5) The staff used this data to derive the following model for CE 14x14 fuel (Reference 7)

$$\frac{\Delta C}{Co} = a + b\sqrt{Bu},$$

$\Delta C/Co$ = fraction of closure for CE fuel

Bu is the bundle average burnup

and a,b are empirical constants fitted to CE data

CE was given credit for thermal margin due to a multiplier of 1.065 on the hot channel enthalpy rise used to account for pitch reduction due to manufacturing tolerances. Table 4.6 presents the required reduction in DNBR using the model described above, after accounting for this thermal margin. Table 4.7 is a list of the reactors to which it applies.

A licensee planning to operate at a burnup greater than 24000 Mwd/MTU should present to the staff an acceptable method of accommodating the thermal margin reduction shown in Table 4.6. This may be done as part of the reload submittal if this burnup will not be obtained during the current cycle.

TABLE 4.6: EFFECT OF ROD BOWING ON DNBR IN REACTORS WITH COMBUSTION ENGINEERING 14x14 FUEL

BURNUP	REDUCTION IN DNBR
Cycle 1 (0-15 $\frac{Gwd}{MTU}$)	0
Cycle 2 (15-24 $\frac{Gwd}{MTU}$)	0
Cycle 3 (24-33 $\frac{Gwd}{MTU}$)	3%

TABLE 4.7: PLANTS FUELED BY CE FUEL TO WHICH VALUES OF TABLE 4.6 APPLY

St. Lucie 1	Cycle 1
Ft. Calhoun	Cycle 3
Millstone 2	Cycle 2
Maine Yankee	Cycle 2
Calvert Cliffs 1	Cycle 1

4.5 Plants Fueled Partially With Exxon Fuel

Palisades, H. B. Robinson, Yankee Rowe and D. C. Cook are partially fueled with Exxon fuel. A discussion of these reactors follows:

Palisades Cycle 2

The Palisades reactor for Cycle 2 is fueled with 136 Exxon fuel assemblies and 68 Combustion Engineering fuel assemblies.

The Combustion Engineering fuel was treated according to the Combustion Engineering model for both extent of rod bow as a function of burnup and DNBR reduction due to clearance reduction.

The Exxon fuel was assumed to bow to the same extent as the Combustion Engineering fuel. This assumption is acceptable since the Exxon fuel has a thicker cladding and other design features which should render the amount of bowing no greater than in the Combustion Engineering fuel.

The DNBR reduction was assumed to be linear with clearance reduction according to the Westinghouse type curve of Figure 2.1. The DNBR reduction at contact was based on the Westinghouse experimental data adjusted for the peak rod average heat flux in Palisades and for the coolant pressure in Palisades.

The variation of the DNBR reduction with coolant pressure is given in Reference 1. The DNBR reduction decreases as the coolant pressure decreases. The overpressure trip set point in Palisades is set at 1950 psi. At this pressure, according to the data presented in Reference 1, the penalty is greatly reduced compared to the penalty at high pressures.

The limiting anticipated transient in the Palisades reactor results in a DNBR of 1.36. The thermal margin between this value and the DNBR limit of 1.3 results in adequate thermal margin to offset the rod bow penalty.

Yankee Rowe Cycle 12

Yankee Rowe is fueled with 40 Exxon fuel assemblies and 36 Gulf United Nuclear Corporation fuel assemblies. The fuel assemblies consist of 16x16 Zircaloy clad fuel rods.

The reduction in DNBR due to fuel rod bowing was assumed to vary linearly with the reduction in clearance between fuel rods. The peak experimental conditions used in the Westinghouse test were used to fix the penalty at full closure. The calculated reduction in DNBR is still less than that which would produce a DNBR less than 1.3 for the most limiting anticipated transient (two pump out of four pump loss-of-flow). Thus, no penalty is required.

H. B. Robinson Cycle 5

H. B. Robinson is fueled with 105 Westinghouse fuel assemblies and 52 Exxon Nuclear Corporation fuel assemblies. The Westinghouse 15x15 DNBR penalty model was applied to the Westinghouse fuel with a correction for the actual heat flux rather than the peak experimental values. The Exxon fuel was considered to bow to the same extent as the Westinghouse 15x15 fuel so that the Westinghouse bow vs. burnup equation was also applied to the Exxon fuel. This assumption is conservative since the Exxon fuel has a thicker cladding and other design features which should render the amount of bowing no greater than in the Westinghouse fuel.

The DNBR reduction calculated by this method was offset by the fact that the worst anticipated transient for H. B. Robinson results in a DNBR of 1.68.

D. C. Cook Cycle 2

D. C. Cook contains 128 Westinghouse fuel assemblies and 65 Exxon fuel assemblies. The limiting transient for D. C. Cook is the Loss of Flow (4 pump coastdown) which has a minimum DNBR of 2.01. This value of DNBR is sufficiently high to accommodate the rod bow penalty for Cycle 2 without reducing the DNBR below the safety limit value of 1.3.

5.0 References

1. Letter to V. Stello, Director, Division of Operating Reactors, USNRC from C. Eicheldinger, Manager, Nuclear Safety Department, Westinghouse Electric Corporation, NS-CE-N61, August 13, 1976.

2. Hill, K. W. et., al, "Effects of a Bowed Rod on DNB", Westinghouse Electric Corporation", WCAP 8176.

3. Standrad Review Plan - Section 4.4, II.1.A.

4. Letter to R. Salvatori, Manager, Nuclear Safety Department, Westinghouse Electric Corporation from D. Vassallo, Chief, Light Water Reactors Project Branch 1-1, Directorate of Licensing, December 4, 1974.

5. Letter to V. Stello, Director, Division of Operating Reactors, USNRC, from P. L. McGill, Combustion Engineering Company, December 15, 1975.

6. Reavis, J. R., et. al., "Fuel Rod Bowing" WCAP 8691 (Proprietary) Westinghouse Electric Corporation, December, 1975.

7. Letter to Mr. Ed Sherer, Combustion Engineering from D. F. Ross, Assistant Director, Reactor Safety, May 14, 1976.

8. Interim Safety Evaluation Report on Westinghouse Fuel Rod Bowing Division of System Safety, USNRC, April, 1976.

FIGURE 2.1

WESTINGHOUSE MODEL

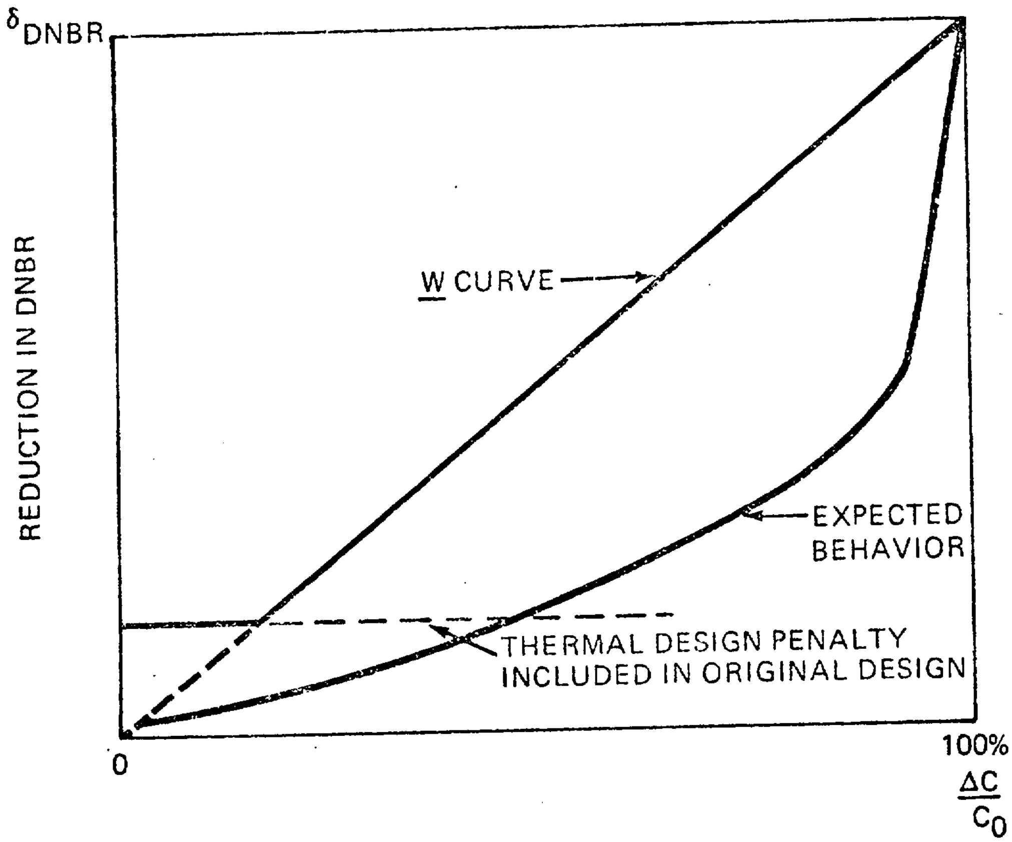
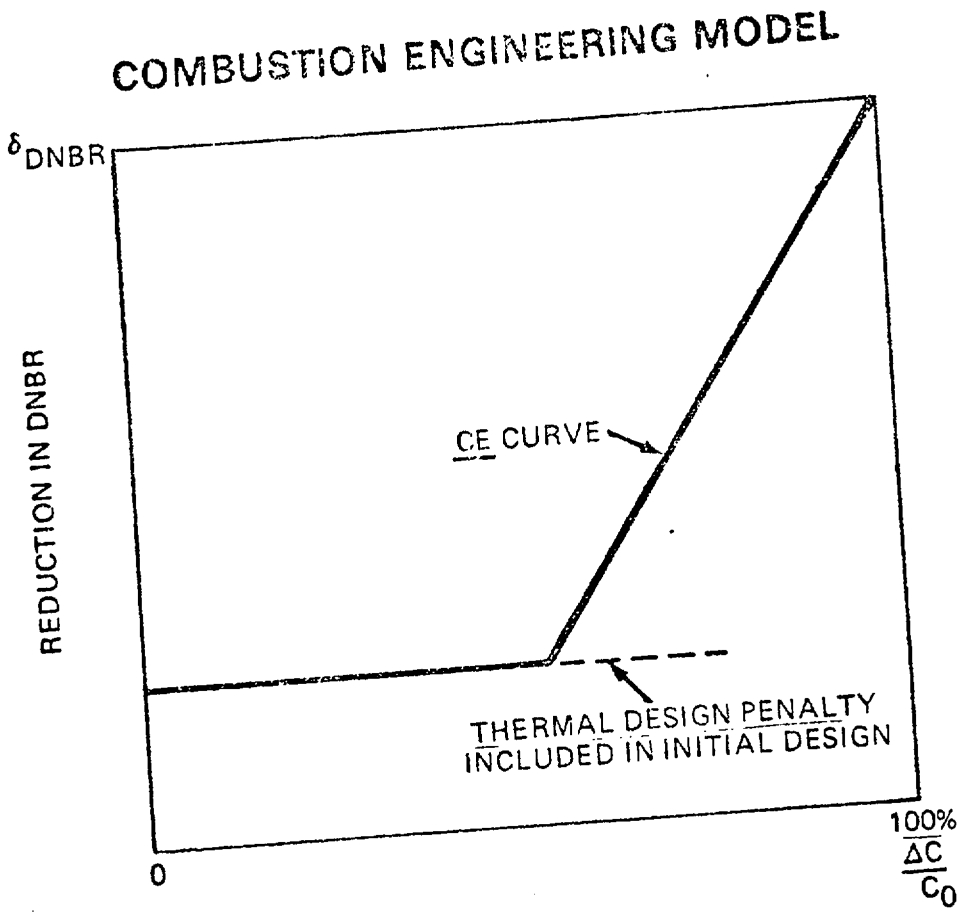


FIGURE 2.2


COMBUSTION ENGINEERING MODEL
δDNBR
REDUCTION IN DNBR
CE CURVE
THERMAL DESIGN PENALTY
INCLUDED IN INITIAL DESIGN
0
100%
ΔC
C0

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 19 and 13 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

These amendments provided for additional DNBR margin to account for the effects of fuel rod bowing. The amendments also include corrections to administrative errors in previously issued license amendments.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 4, 1976, and supplement thereto dated January 28, 1977, (2) Amendment Nos. 19 and 13 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of March, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors