H. B. Robinson Unit 2 Cycle 14

CORE OPERATING LIMITS REPORT

Revision 0

November 16, 1989

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for H. B. Robinson Unit 2 Cycle 14 has been prepared in accordance with the requirements of Technical Specification 6.9.3.3.

The Technical Specifications affected by this report are listed below:

3.1.3.1 3.1.3.3	Moderator Temperature Coefficient
3.10.1.2	Shutdown Rod Insertion Limits
3.10.1.3 3.10.1.4	Control Rod Insertion Limits
3.10.2.1 3.10.2.2 3.10.2.2.1 3.10.2.2.2	Heat Flux Hot Channel Factor
3.10.2.1	Nuclear Enthalpy Rise Hot Channel Factor
3.10.2.2 3.10.2.2.1 3.10.2.2.2 3.10.2.7 3.10.2.9 3.10.2.11	Axial Flux Difference

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.3.3.

2.1 Moderator Temperature Coefficient

(Specifications 3.1.3.1 and 3.1.3.3)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

a) The MTC shall be less than or equal to +5.0 pcm/^{o}F at less than 50% of rated power, or

b) The MTC shall be less than or equal to 0.0 pcm/^{o}F at 50% of rated power and above.

2.2 Shutdown Rod Insertion Limits (Specification 3.10.1.2)

2.2.1 The shutdown rods shall be withdrawn to at least 225 steps.

2.3 Control Rod Insertion Limits

(Specifications 3.10.1.3 and 3.10.1.4)

2.3.1 The control rods shall be limited in physical insertion as shown in Figure 1.0.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$

(Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, and 3.10.2.2.2)

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

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

$$\text{where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

2.4.1 $F_Q^{RTP} = 2.32$

2.4.2 K(Z) is specified in Figure 2.0

2.5 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$

(Specification 3.10.2.1)

$$F_{\Delta H} \leq F_{\Delta H}^{RTP} (1 + PF_{\Delta H}(1-P))$$

$$\text{where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

2.5.1 $F_{\Delta H}^{RTP} = 1.65$

2.5.2 $PF_{\Delta H} = 0.2$

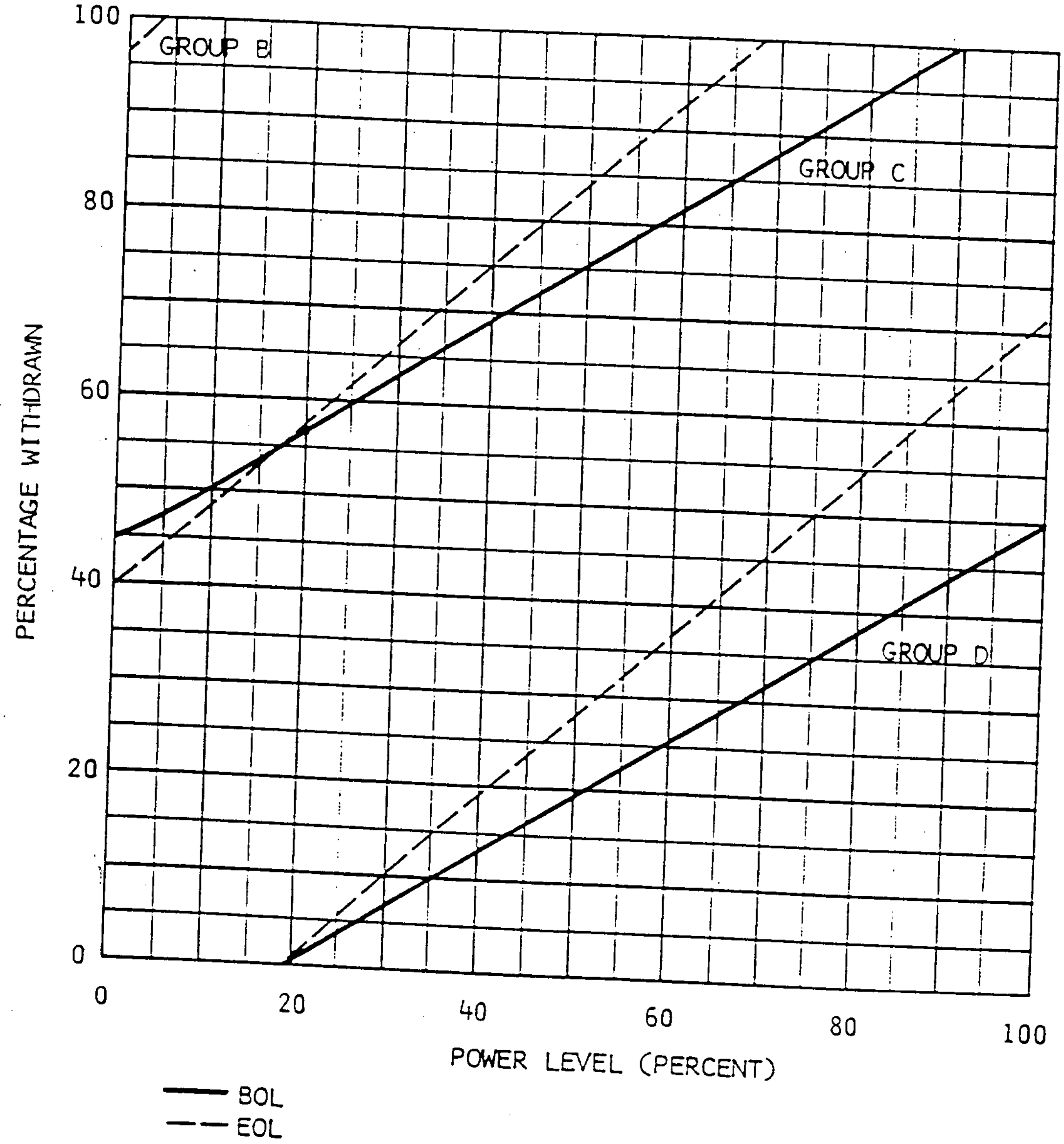
2.6 Axial Flux Difference

(Specifications 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, 3.10.2.11)

2.6.1 The axial flux difference target bands are ± 3% and ± 5% about the target AFD.

2.6.2 V(Z) values for the ± 3% and ± 5% target bands are specified in Figure 3.0.

2.6.3 The AFD Acceptable Operation Limits are specified in Figure 4.0.



H.B. ROBINSON Unit #2 Carolina Power & Light Company Technical Specifications	Control Group Insertion Limits For Three-Loop Operation	Figure 1.0 Page 5 of 8

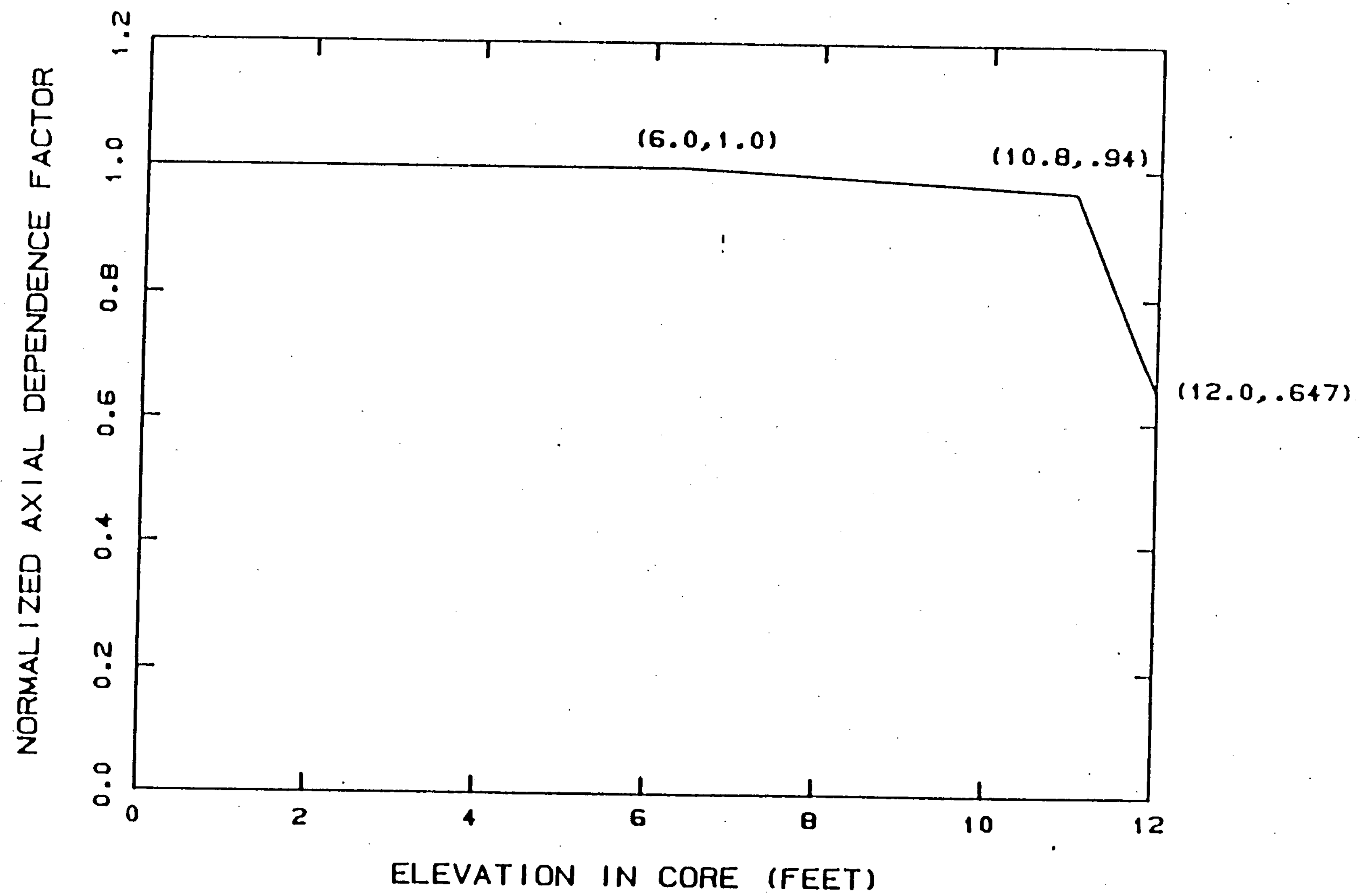
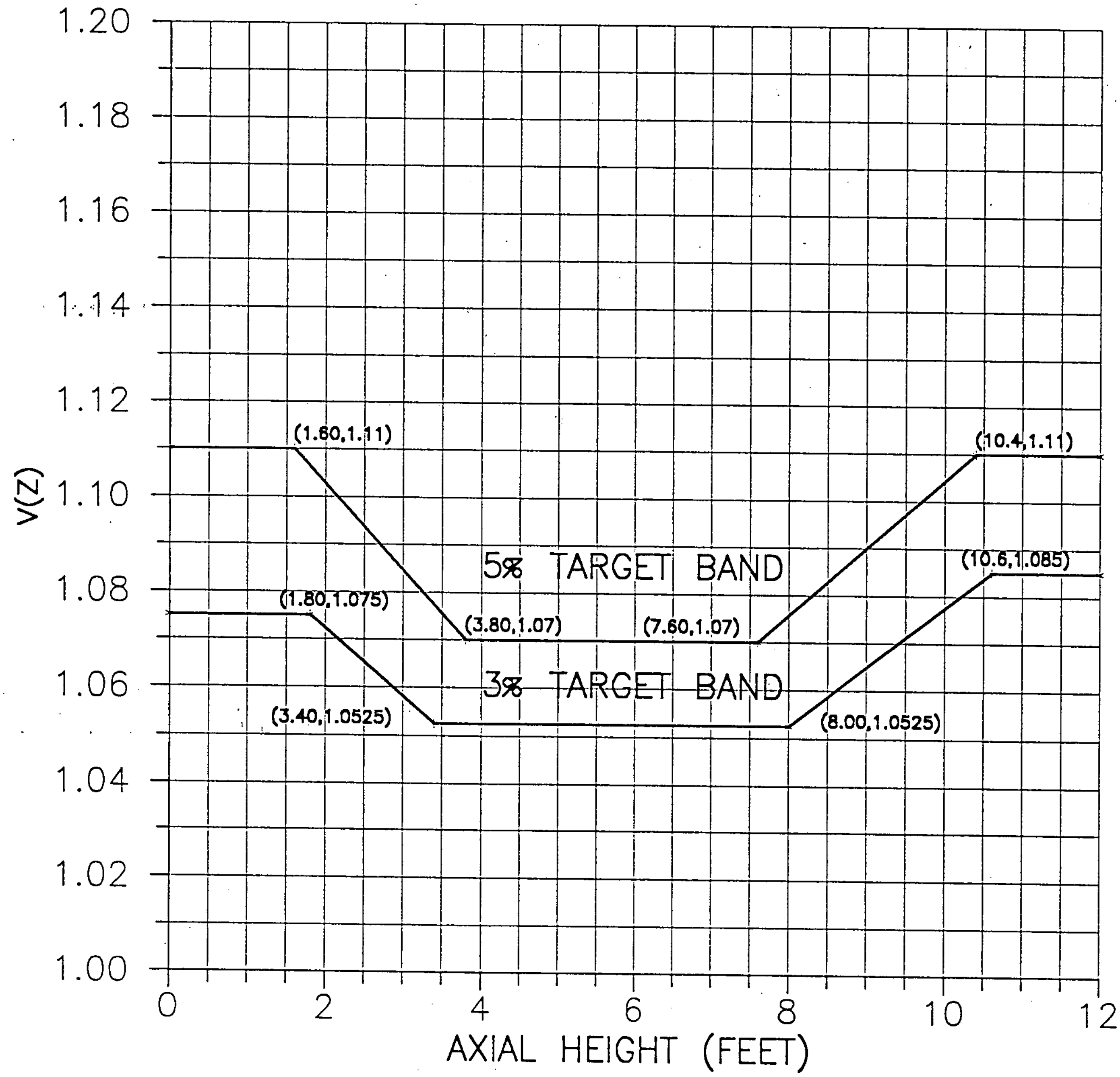


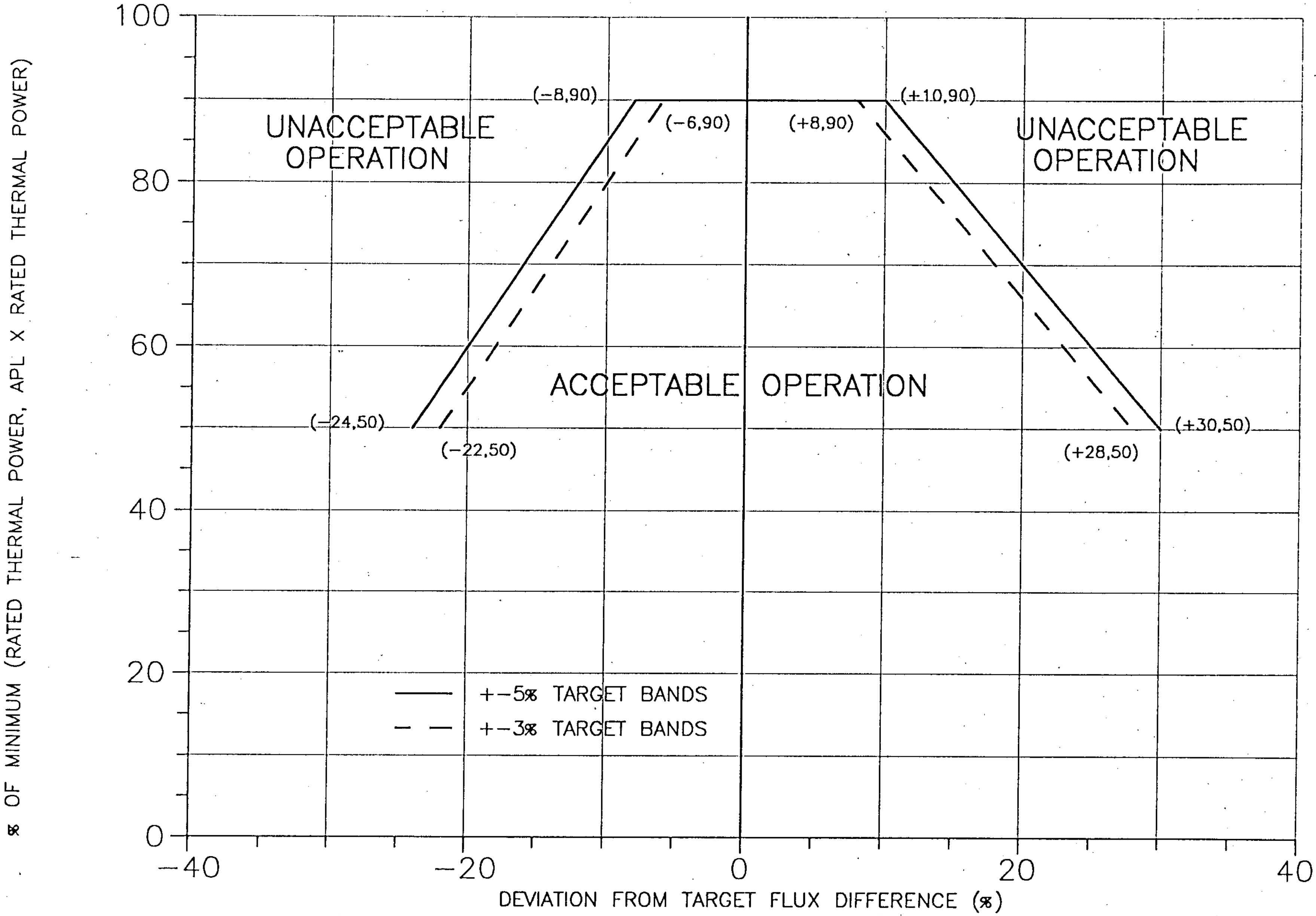
Figure 2.0

NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION
(PEAK Fq = 2.32)



V(Z) AS A FUNCTION OF CORE HEIGHT

Figure 3.0



ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE
Figure 4.0

ATTACHMENT 2

ENCLOSURE 1

NRC Staff Review of the Westinghouse Emergency Core Cooling System Evaluation Model

Background Information

Westinghouse submitted a description of the Westinghouse Emergency Core Cooling System (ECCS) evaluation model on August 5, 1974, in several reports (References 1 through 18 of Enclosure 2). The NRC staff reviewed the August submittal for conformance with the requirements of Appendix K to 10 CFR Part 50 entitled "ECCS Evaluation Models," and published its evaluation in "Status Report by the Directorate of Licensing in the Matter of the Westinghouse ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K" (Reference 21) dated October 15, 1974. This report addressed each requirement of Appendix K, discussed conformance by Westinghouse, indicated the acceptability of the analytical methods employed in the Westinghouse model, and assessed the impact of specific open items which were either unresolved or unacceptable.

Additional documentation was subsequently submitted by Westinghouse addressing these open items and the staff review entitled "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," was published on November 13, 1974 (Reference 22). We concluded that certain modifications, which were described in the above mentioned documents, were required to achieve conformance with Appendix K to 10 CFR Part 50.

On October 26 and November 14, 1974, the staff presented its assessment of the Westinghouse evaluation model to the Advisory Committee on Reactor Safeguards (ACRS). In its report to the Chairman of the AEC, dated November 20, 1974, the ACRS concluded that "the four light-water reactor vendors have developed evaluation models which, with additional modifications required by the staff, will conform to Appendix K to Part 50."

The required model changes which were subsequently implemented by Westinghouse into their evaluation model included the following:

1 - Deletion of rod-to-rod radiation during blowdown.
2 - Revision of the injection section pressure drop.
3 - Deletion of the 10% clad strain burst criterion.
4 - Inclusion of a hot wall time delay.
5 - Modifications to the steam cooling model for reflood rates less than 1 inch/second.

On December 27, 1974, the Commission, in response to licensee submittals of additional information from operating Westinghouse plants, pursuant to Section 50.46 and Appendix K to 10 CFR Part 50, issued a Safety Evaluation Report and Orders for Modification of Licenses pertaining to the latest proposed Technical Specifications. In addition, the Commission requested that the above modifications to the Westinghouse evaluation model be made and that a reanalysis be submitted within six months.

On April 17, 1974, Westinghouse formally submitted proprietary and non-proprietary versions of a topical report (References 19 and 20) which documented all of the modifications required by the staff in October and November 1974 (References 21 and 22). In addition to the modifications required by the staff, Westinghouse submitted supplemental information to complete the documentation requirements of Appendix K, including responses to questions raised by the staff in the course of reviewing the Westinghouse ECCS evaluation model, refinements to the steam cooling model, and documentation of minor modifications which had no significant effect on computational results.

Conclusions

The NRC staff has completed its review of the Westinghouse ECCS evaluation model which is comprised of References 1 through 20. We closely followed the development of the Westinghouse ECCS evaluation model and utilized the referenced reports to determine the compliance of the Westinghouse evaluation model with Appendix K to 10 CFR Part 50. The details of our review are summarized in References 21 and 22. We conclude:

1. That the Westinghouse evaluation model is an acceptable model to be used for ECCS performance evaluation for plants which satisfy the following plant classifications:

   a. Typical current Westinghouse ~~two~~, three, and four loop plants.

   b. Dry, subatmospheric or ice containments.

   c. Power ratings up to 3800 MWt.

   d. Plants utilizing only bottom flooding emergency core cooling systems.

2. That References 1 through 20, which constitute the consolidated description of the Westinghouse ECCS evaluation model, may be incorporated by reference in licensing applications as an accepted ECCS evaluation model. This acceptance applies to the Westinghouse ECCS model and does not constitute acceptance of the individual reports for any purpose other than for ECCS analyses.

3. The application of the LOTIC code for purposes of ECCS performance evaluation, in determining the minimum containment pressure response for all ice containments, will be evaluated on a plant-by-plant basis. Until such time that LOTIC is modified to resolve the staff concerns, noted in references 21 and 22, a conservative minimum containment pressure of zero psig must be assumed in ECCS analyses of plants using an ice condenser containment.

ENCLOSURE 2

REFERENCES

1. WCAP-8200, Rev. 2, "WFLASH - A Fortran IV Computer Program for Simula of Transients in a Multi-Loop PWR," (Proprietary), June 1974. 16.

2. WCAP-8261, Rev. 1, "WFLASH - A Fortran IV Computer Program for Simulat of Transients in a Multi-Loop PWR," (Non-proprietary), July 1974. 17.

3. WCAP-8359 - "Effects of Fuel Densification Power Spikes on Clad Therma Transients," (Non-proprietary), July 1974. 18.

4. WCAP-8354 - "Long-Term Ice Condenser Containment Code - LOTIC Code," (Proprietary), July 1974. 19

5. WCAP-8355 - "Long-Term Ice Condenser Containment Code - LOTIC Code," (Non-proprietary), July 1974. 20

6. WCAP-8340 - "Westinghouse ECCS - Plant Sensitivity Studies," (Proprieta July 1974. 2

7. WCAP-8356 - "Westinghouse ECCS - Plant Sensitivity Studies," (Non-proprietary), July 1974.

8. WCAP-8327 - "Containment Pressure Analysis Code (COCO)," (Proprietary), July 1974.

9. WCAP-8326 - "Containment Pressure Analysis Code (COCO)," (Non-proprieta July 1974.

10. WCAP-8302 - "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," (Proprietary), June 1974.

11. WCAP-8306 - "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," (Non-proprietary), June 1974.

12. WCAP-8170 - "Calculational Model for Core Reflooding After a LOCA (WREFLOOD Code)," (Proprietary), June 1974.

13. WCAP-8171 - "Calculational Model for Core Reflooding After a LOCA (WREFLOOD Code)," (Non-proprietary), June 1974.

14. WCAP-8301 - "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," (Proprietary), June 1974.

15. WCAP-8305 - "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," (Non-proprietary), June 1974.

16. WCAP-8339 - "Westinghouse ECCS Evaluation Model - Summary," (Non-proprietary), June 1974.

17. WCAP-8341 - 'Westinghouse ECCS Evaluation Model Sensitivity Studies," (Proprietary), July 1974.

18. WCAP-8342 - "Westinghouse ECCS Evaluation Model Sensitivity Studies," (Non-proprietary), July 1974.

19. WCAP-8471 - "Westinghouse ECCS Evaluation Model - Supplementary Information," (Proprietary), January 1975.

20. WCAP-8472 - 'Westinghouse ECCS Evaluation Model - Supplementary Information," (Non-proprietary), January 1975.

21. "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K,' October 15, 1974.

22. "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," November 13, 1974.

ATTACHMENT 4

6.5.1.6.5 A quorum of the PNSC shall consist of the Chairman, and four members, of which two may be alternates.

6.5.1.6.6 The PNSC activities shall include the following:

a) Perform an overview of Specifications 6.5.1.1 and 6.5.1.2 to assure that processes are effectively maintained.

b) Performance of special reviews, investigations, and reports thereon requested by the Manager - Nuclear Assessment Department.

c) Annual review of the Security Plan and Emergency Plan.

d) Perform reviews of Specifications 6.5.1.1.6, 6.5.1.2.4, 6.5.1.3.1, and 6.5.1.4.1.

e) Perform review of all reportable events.

f) Review of facility operations to detect potential nuclear safety hazards.

g) Review of every unplanned on site release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrences to the Vice President - Robinson Nuclear Project, Manager - Nuclear Assessment Department.

h) Review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

i) Review of major changes to radioactive liquid, gaseous, and solid waste treatment systems.

j) Review of changes to the CORE OPERATING LIMITS REPORT.

6.9.3.3.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a) XN-75-27(A), latest Revision and Supplements, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

b) XN-NF-84-73(P), latest Revision and Supplements, "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Corporation, Richland WA 99352 (Accepted by the NRC for H. B. Robinson Unit 2 in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 87 to Facility License No. DPR-23, 7 Nov. 84).

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

c) XN-NF-82-21(A), latest Revision, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specification 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

d) XN-NF-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's," Exxon Nuclear Corporation, Richland, WA 99352.

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor.)

e) XN-75-32(A), Supplements 1, 2, 3, 4, "Computational Procedure for Evaluating Rod Bow," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

f) XN-NF-82-49(A), latest Revision, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

g) EXEM PWR Large Break LOCA Evaluation Model as accepted in Letter, D. M. Crutchfield (NRC) to G. N. Ward (ENC), "Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," July 8, 1986.

EXEM PWR LBLOCA Model includes the following references:

XN-NF-82-20(P), latest Revision and Supplements, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Richland WA 99352.

XN-NF-82-07(A), latest Revision, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Richland WA 99352.

XN-NF-81-58(A), latest Revision, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Richland WA 99352.

XN-NF-85-16(P), Volume 1 and Supplements, Volume 2, latest Revision and Supplements, "PWR 17x17 Fuel Cooling Test Program," Exxon Nuclear Company, Richland WA 99352.

XN-NF-85-105(P), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

h) XN-NF-78-44(A), latest Revision, "Generic Control Rod Ejection Analysis," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor)

i) XN-NF-621(A), latest Revision, "XNB Critical Heat Flux Correlation," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specification 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

j) ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Advanced Nuclear Fuels Corporation, Richland WA 99352.

(Methodology for Specification 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

k) XN-NF-82-06(A), latest Revisions and Supplements, "Qualification of Exxon Nuclear Fuel for Extended Burnup", Exxon Nuclear Corporation, Richland, WA 99352.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

l) Meyer, P. E. and Kornfilt, J., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10080-A, August 1985.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

m) Lee, N., Tauche, W. D., Schwartz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," WCAP-10081-A, August 1985.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

n) Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301, (Proprietary) and WCAP-8305, (Nonproprietary), June 1974 (accepted by the NRC in the SER related to WCAP-8472-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information", April, 1975).

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

o) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power and Light Co., H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," USNRC, Washington D. C. 20555, 7 Nov. 84.

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

p) ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland WA 99352, latest revisions and supplements. (Accepted by the NRC for H. B. Robinson Steam Electric Plant, Unit 2, in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 128 to Facility License No. DPR-23, Docket No. 50-261, USNRC, Washington D.C., 20555, August 22, 1990).

(Methodology for Specifications 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

6.9.3.3.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.3.3.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

July 8, 1986

Mr. Gary M. Ward, Manager, Reload Licensing
Exxon Nuclear Company
2101 Horn Rapids Road
Richland, Washington 99352

Dear Mr. Ward:

SUBJECT: SAFETY EVALUATION OF EXXON NUCLEAR COMPANY'S LARGE BREAK ECCS EVALUATION MODEL EXEM/PWR AND ACCEPTANCE FOR REFERENCING OF RELATED LICENSING TOPICAL REPORTS

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Exxon Nuclear Company's large break ECCS Evaluation Model entitled EXEM/PWR. The EXEM/PWR Evaluation Model is described in the following topical reports:

XN-NF-82-20(P), Revision 1, August 1982
XN-NF-82-20(P), Revision 1, Supplement 1, June 1983
XN-NF-82-20(P)(A), Revision 1, Supplement 2, February 1985
XN-NF-82-20(P), Revision 1, Supplement 3, June 1985
XN-NF-82-20(P), Revision 1, Supplement 4, July 1984
XN-NF-82-07(P)(A), Revision 1, November 1982
XN-NF-81-58(P)(A), Revision 2, Supplements 1 & 2, March 1984
XN-NF-85-16(P), Volume 1, February 1985
XN-NF-85-16(P), Volume 1, Supplement 1, January 1986
XN-NF-85-16(P), Volume 1, Supplement 2, January 1986
XN-NF-85-16(P), Volume 1, Supplement 3, March 1986
XN-NF-85-16(P), Volume 2, Revision 1, January 1986
XN-NF-85-16(P), Volume 2, Revision 1, Supplement 1, January 1986
XN-NF-85-105(P), October 1985
XN-NF-85-105(P), Supplement 1, January 1986

The staff finds the reports to be acceptable for referencing in licence applications to the extent specified and under the limitations delineated in the reports and the associated NRC evaluation, which is enclosed. The evaluation defines an acceptable large break model in compliance with Appendix K to 10 CFR 50.

The staff does not intend to repeat our review of the matters described in the reports and found acceptable when the reports appear as references in license applications, except to assure that the material presented is applicable to the specific plant involved. This acceptance applies only to the matters described in the reports.

In accordance with procedures established in NUREG-0390, it is requested that ENC publish accepted versions of these reports within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation after the title page. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

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RAPIFAX NO. 365
PAGE 1 OF 2

The EXEM/PWR model becomes effective with the publication of the accepted versions of the aforementioned topical reports. All submittals demonstrating compliance with Appendix K to 10 CFR 50 using the Exxon ECCS Evaluation Model after that time must use the model described in the enclosed SER. During the ensuing 90 days, applicants or licensees requiring an Appendix K analysis may elect to submit an analysis based upon the previously approved ENC WREM-IIA model, with consideration of the effect of NUREG-0630, or submit analyses conforming to the enclosed SER. When the topical reports are referenced, the references must include both the proprietary and nonproprietary versions.

Should the criteria or regulations change such that the staff conclusions as to the acceptability of these reports are invalidated, ENC and/or the applicants referencing these topical reports will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of these topical reports without revision of their respective documentation.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
Division of PWR Licensing-B

Enclosure: Safety Evaluation

cc: W. Minners
J. Watt
C. Berlinger
R. Hodges
D. Wiggenton
R. Emch
C. Miller

ATTACHMENT 3

We consider the non-proprietary reports, listed in Enclosure 1, acceptable versions of the corresponding proprietary reports listed in Enclosure 1. When either a proprietary report or a non-proprietary version of a propriet. report is used as a reference, both the proprietary report and its non-proprietary version must be included in the reference.

If you have any questions about our review of the Westinghouse ECCS evaluation model, please contact us.

Sincerely,

D. B. Vassallo, Chief
Light Water Reactors Project Branch 1-1
Division of Reactor Licensing

Enclosures:
1. Summary of Staff Evaluation of ECCS Model
2. List of Westinghouse Reports Describing ECCS Model

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

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C. EICHELDINGER
PWR NUCLEAR SAFETY

Mr. C. Eicheldinger, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Eicheldinger:

The Nuclear Regulatory Commission staff has completed its review of the Westinghouse Electric Corporation Emergency Core Cooling System (ECCS) evaluation model. The model is defined in References 1 through 20 of Enclosure 2. A summary of our evaluation of Westinghouse's ECCS model is attached as Enclosure 1.

As a result of our review, we have concluded that the Westinghouse evaluation model meets the requirements of Appendix K to 10 CFR Part 50 and, therefore, is acceptable for use in the ECCS evaluation for the following classes of plants:

1 - Typical current Westinghouse two, three, and four loop plants.

2 - Plants with dry, subatmospheric or ice containments.

3 - Plants with power ratings up to 3800 MWt.

4 - Plants utilizing only bottom flooding emergency core cooling systems.

Therefore, the Westinghouse reports defining the ECCS evaluation model may be referenced in license applications as accepted for use in ECCS analyses. We find, however, that the LOTIC code (described in References 4 and 5) is not presently acceptable for determining the minimum containment pressure response for all plants using ice condenser containments. Until such time as the code is modified to resolve our concerns (stated in References 21 and 22), a conservative minimum containment pressure of zero psig must be assumed in the ECCS analyses of plants with ice condenser containments.

We do not intend to repeat our review of the Westinghouse reports listed in Enclosure 2 when they are used in support of the ECCS evaluation model. Our acceptance applies only to the use of the reports as part of the Westinghouse ECCS evaluation model and does not constitute acceptance of the individual reports for any purpose other than for ECCS analyses.

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