docket File

DEC 27 1974

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Order for Modification of License for the H. B. Robinson Steam Electric Plant Unit No. 2 pertaining to your proposed Technical Specifications which were submitted pursuant to Section 50.46 and Appendix K of 10 CFR Part 50. The enclosed Safety Evaluation Report contains the bases for our action.

A copy of the Order is being filed with the Office of the Federal Register for publication. This order effective immediately.

Within 30 days after receipt of this letter, we request that you inform us of your submittal date for the reevaluation required by paragraph 1 of Section III of the Order.

You will note that the order requires that a reanalysis in conformance with the order must be submitted along with any request for authorization for any core reloading. In order to provide for sufficient time for our review of your reanalysis, you should assure that such submittal is provided at least 45 days prior to your schedule of initiation of operation following approval of such reloading. You should also note, that since your current analysis was based upon your presently authorized fuel loading patterns, any modification of fuel design or core configuration which affects the basis for the analysis will require staff approval.

For enclosures 3 & 4 (combined) see Dockets Nos. 50-295/304.

Sincerely,

/S/

G. Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

ECCS R.

DEC 27 1974

Enclosures:
1. Order
2. Safety Evaluation Report
3. Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company Model Conformance to 10 CFR 50, Appendix K
4. Supplement 1 to the Status Report

cc: Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
910 - 17th Street, N. W.
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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

DEC 27 1974

Docket No. 50-261

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Senior Vice President
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Raleigh, North Carolina 27602

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Sincerely,

G. Lear

G. Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

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Columbia, South Carolina 29201

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-261
	)	
(H. B. Robinson Steam Electric Plant Unit No. 2)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Carolina Power & Light Company (the licensee) is the holder of facility license DPR-23, which authorizes operation of the H. B. Robinson Steam Electric Plant Unit No. 2 in Darlington County, South Carolina. This license provides, among other things, that it is subject to all rules, regulations and orders of the Commission now or hereafter in effect.

II.

Pursuant to the requirements of the Commission's regulations in 10 CFR § 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", on October 2, 1974, the licensee submitted an evaluation of ECCS cooling performance calculated in accordance with an evaluation model developed by the Westinghouse Corporation ("the vendor"), along with certain proposed technical specifications necessary to bring reactor operation into conformity with the results of the evaluation.

The evaluation model developed by the vendor has been analyzed by the Regulatory staff for conformity with the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models". The Regulatory staff's evaluation of the vendor's model is described in two previously published documents: Status Report by the Directorate of Licensing in the Matter of the Westinghouse ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K, issued October 15, 1974, and a Supplement to the Status Report, issued November 13, 1974. Based on its evaluation, the Regulatory staff has concluded that the vendor's evaluation model was not in complete conformity with the requirements of Appendix K and that certain modifications described in the above-mentioned documents were required in order to achieve such conformity. The Regulatory staff assessments were reviewed by the Commission's Advisory Committee on Reactor Safeguards in meetings held on October 26, 1974, and November 14, 1974.

In its Report to the Chairman of the AEC, dated November 20, 1974, the Advisory Committee has concluded that "the four light-water reactor vendors have developed Evaluation Models which, with additional modifications required by the Regulatory staff, will conform to Appendix K to Part 50".

Since the licensee's evaluation of ECCS cooling performance is based upon the vendor's evaluation model, the licensee's evaluation is similarly deficient. The Regulatory staff has assessed the effect of the changes required in the evaluation model upon the results of the evaluation of ECCS performance for the Robinson 2 facility submitted on October 2, 1974. This is described in the staff Safety Evaluation Report of the H. B. Robinson Steam Electric Plant Unit No. 2 dated December 27, 1974. On the basis of its review, the Regulatory staff has determined that changes in operating conditions for the plant, in addition to those proposed in the licensee's submittal of October 2, 1974, are necessary to assure that the criteria set forth in § 50.46(b) are satisfied. These additional changes, are set forth in Appendix A to the Safety Evaluation Report. These further restrictions will assure that ECCS cooling performance will conform to all of the criteria contained in 10 CFR § 50.46(b), which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling.

These further restrictions were established on the basis of studies of the effect of model changes on the previously submitted evaluations. The Regulatory staff believes that these restrictions should be verified by a re-analysis based upon an approved evaluation model, in conformity with 10 CFR § 50.46 and Appendix K. During the interim, before an evaluation in conformity with the requirements of 10 CFR § 50.46 can be submitted and evaluated, the Regulatory staff has concluded that continued conformance to the requirements of the Commission's Interim Acceptance Criteria,* and conformance to the restrictions contained in the licensee's October 2, 1974 submittal, together with the additional limitations set forth in Appendix A of the Staff Safety Evaluation Report, will provide reasonable assurance that the public health and safety will not be endangered. These additional restrictions are set forth as Appendix A to this Order.

III.

In view of the foregoing and, in accordance with the provisions of § 50.46(a)(2)(v), the Acting Director of Licensing has found that the evaluation of ECCS cooling performance submitted by the licensee is not consistent with the requirements of 10 CFR § 50.46(a)(1) and that the further restrictions set forth in this Order are required to protect the public health and safety. The Acting Director of Licensing has also found that the public health, safety, and interest require that the following Order be made effective immediately. Pursuant to the Atomic Energy Act of 1954, as amended, the Commission's regulations in 10 CFR §§ 2.204, 50.46, and 50.54.

*Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors, 36 F.R. 12247, June 29, 1971, as amended 36 F.R. 24082, December 18, 1971

IT IS ORDERED THAT:

1. As soon as practicable, but in no event later than six months from the date of publication of this order in the FEDERAL REGISTER, or prior to any license amendment authorizing any core reloading, whichever occurs first, the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, § 50.46. Such evaluation may be based upon the vendor's evaluation model as modified in accordance with the changes described in the Staff Safety Evaluation Report of the Robinson Steam Electric Plant, dated December 27, 1974. The evaluation shall be accompanied by such proposed changes in the Technical Specifications or license amendments as may be necessary to implement the evaluation results.

2. Effective immediately, reactor operation shall continue only within the limits:

(a) The requirements of the Interim Acceptance Criteria, the Technical Specifications, and license conditions imposed by the Commission in accordance with the requirements of the Interim Acceptance Criteria, and

(b) The limits of the proposed Technical Specifications submitted by the licensee on October 2, 1974, as modified by the further restrictions set forth in Appendix A, attached hereto.

The licensee shall conform operation to the foregoing limitations until such time as the proposed Technical Specifications required to be submitted in accordance with paragraph 1 above are approved or modified and issued by the Commission. Subsequent notice and opportunity for hearing will be provided in connection with such action.

IV.

Within thirty (30) days from the date of publication of this Order in the FEDERAL REGISTER the licensee may file a request for a hearing with respect to this Order. Within the same thirty (30) day period any other person whose interest may be affected may file a request for a hearing with respect to this Order in accordance with the provisions of 10 CFR § 2.714 of the Commission's Rules of Practice. If a request for a hearing is filed within the time prescribed herein, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this action, see (1) the licensee's submittal dated October 2, 1974 and vendor's topical reports referenced in the licensee's submittal, which describe the vendor's evaluation model, (2) the Status Report by the Directorate of Licensing in the Matter of Westinghouse ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K, (3) Supplement 1 thereto dated November 13, 1974, (4) The Safety Evaluation Report dated December 27, 1974, and (5) Report of the

Advisory Committee on Reactor Safeguards dated November 20, 1974. All of these items are available at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Hartsville Memorial Library, Hartsville, South Carolina.

A single copy each of items (2) through (5) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C., 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing, Regulation.

Dated at Bethesda, Maryland this 27th day of December, 1974.

FOR THE ATOMIC ENERGY COMMISSION

/S/

Edson G. Case, Acting Director
Directorate of Licensing

APPENDIX A

OPERATING LIMITS

1. The limiting F_Q without uncertainty in Specification 3.10.2.1, page 3.10.2 of the proposed Technical Specifications submitted on October 2, 1974, shall be 2.09 instead of 2.103 in the numerator of the second equation and at the end of item 3.10.2.1.b. The bases shall be changed accordingly.

2. In the proposed Technical Specifications, submitted on October 2, 1974, if the quadrant tilt is greater than 1.09, the action required by Specification 3.10.3.2.a, page 3.10-3, shall be revised to eliminate the words "and power range high flux setpoint".

3. If the APDMS is out of service as described in Specification 14.11.2 on page 4.11-2 of the proposed Technical Specifications, submitted on October 2, 1974, a log shall be kept of accumulated rod motion and time of manual traverses. Alternatively, if constant axial offset control procedures are used, conformance with the applicable limit and the flux difference shall be logged hourly for the first 24 hours, and half-hourly thereafter.

SAFETY EVALUATION REPORT

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

ISSUED: December 27, 1974

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APPENDIX A - OPERATING LIMITS

APPENDIX B - LETTER FROM ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 20, 1974

1.0 INTRODUCTION

On January 4, 1974, the Commission published its acceptance criteria for emergency core cooling systems (ECCS) for light water power reactors (39 FR 1003).[1] This rule includes Appendix K to 10 CFR Part 50 which specifies analytical techniques to be employed for the evaluation of the ECCS effectiveness. On August 5, 1974, Westinghouse officially submitted a seventeen volume package of topical reports[2-18] constituting their proposed ECCS evaluation model. The information contained in these reports had been the subject of a number of informal conferences and discussions between the staff and Westinghouse, starting shortly after the publication of the Acceptance Criteria in January, 1974. The Regulatory staff reviewed these documents and published a Status Report on October 15, 1974,[19] which addressed each item required by Appendix K and identified areas which had been acceptable to the staff and areas of staff concern which were to be resolved. On November 13, 1974,[20] the Regulatory staff published a Supplement to the Status Report which addressed each of these areas of concern. As reflected in the Supplement, for some items adequate additional information was provided to enable the staff to accept the Westinghouse approach. For certain other items, the staff concluded that adequate justification had not been provided and that further modification of the model was required. Westinghouse agreed to modify its model in accordance with the staff's comments. Since that time Westinghouse has made the model adjustments required, which are discussed in Section 2.0 of this SER and has evaluated the impact of all model changes upon previously submitted analyses. Accordingly, the Westinghouse evaluation model with the modifications described in Section 2.0 of this Safety Evaluation is acceptable and would conform to Appendix K.

A report of the Advisory Committee on Reactor Safeguards regarding the generic review and the acceptability of the Westinghouse ECCS Evaluation Model was issued on November 20, 1974, attached as Appendix B.

On October 2, 1974, Carolina Light and Power Company (the Licensee) submitted an analysis of ECCS performance for the Robinson Unit 2 Power Station along with proposed Technical Specification changes to reflect the impact of the new ECCS evaluation model calculations.[(21)] This evaluation was based upon the Westinghouse evaluation model submitted on August 5, 1974. The applicability of the generic evaluation model to the specific Robinson 2 plant analyses is discussed in Section 3.0 of this SER.

As stated in the Status Report and its Supplement, the August 5th Westinghouse evaluation model was not completely acceptable and specific model changes noted in the Status Report and its Supplement were required. These changes have been made to the generic Westinghouse evaluation model. Since the Robinson evaluation was based upon a model which was not acceptable, it also will require some changes. A revised set of computations for

Robinson (and for other facilities in a like position), using the newly revised and acceptable evaluation model, cannot be submitted for a number of months.

To determine the effect of the changes made to the August 5,1974 Westinghouse evaluation model, the staff requested and Westinghouse submitted a series of generic plant sensitivity studies which quantified the effect of the model changes on the results of previously performed calculations. The staff closely followed the performance of these sensitivity studies while they were in progress and has reviewed the results upon completion. These results are presented in Section 4.0 along with a discussion of the effects of these results on the evaluation submitted for Robinson on October 2, 1974.

From a review of the October 2nd submittal and these studies it appears that certain operating restrictions, in addition to those set forth in the proposed Technical Specifications submitted by the licensee on October 2, 1974, are required in order to be certain that in the event of a postulated loss-of-coolant accident, ECCS cooling performance will not exceed the values for calculated peak clad temperature, oxidation, and hydrogen generation limits set forth in 10 CFR 50.46(b). These further restrictions are set forth in Appendix A hereto. Although these further restrictions were established on the basis of applicable generic sensitivity studies of the effect of model changes, the staff believes that in conformity with the requirements of 10 CFR 50.46 these restrictions should be verified by a re-evaluation based upon the Westinghouse evaluation model as corrected. An evaluation of ECCS performance, wholly in conformity with 10 CFR 50.46 and Appendix K, and based on an approved evaluation model, should be submitted for the Robinson facility, as promptly as it can reasonably be performed, but within six months, along with proposed Technical Specifications based upon such re-evaluation.

During the interim, before an evaluation wholly in conformity with the requirements of 10 CFR 50.46 can be submitted and evaluated, continued conformance to the requirements of the Commission's Interim Acceptance Criteria (IAC) and the restrictions contained in the licensee's October 2, 1974, submittal as modified by the additional limitations set forth in Appendix A hereto will provide reasonable assurance that the public health and safety will not be endangered.

2.0 WESTINGHOUSE ECCS EVALUATION MODEL

The Regulatory staff has published a Status Report and a Supplement which addressed each requirement of Appendix K of 10 CFR 50, discussed conformance by Westinghouse, and the acceptability of the analytical methods. The staff identified specific aspects of the evaluation model* which were not in conformance with Appendix K and required additional modifications which have been made by Westinghouse.

The following sections discuss the required modifications to the Westinghouse evaluation model. Additional detail is presented in the staff Status Report and Supplement.

2.1 Swelling and Rupture of the Cladding

Westinghouse had proposed an additional criterion for predicting the incidence of rupture based on an arbitrary value of hoop strain prior to rupture. The staff required that this additional rupture criterion be removed from the Westinghouse ECCS evaluation model. Westinghouse has complied with this requirement.

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* A complete listing of each computer program, in the same form as used in the evaluation model, was furnished to the Regulatory staff, with the understanding that it be stored at a Westinghouse Nuclear Energy Systems location, accessible only to the AEC Regulatory staff for review.

2.2 Post-CHF Heat Transfer

Westinghouse had proposed to use the Bishop, Sandberg, and Tong correlation for subcooled film boiling. The staff indicated that this correlation is inappropriate for subcooled conditions and indicated that it should not be included in the Westinghouse ECCS evaluation model. Westinghouse has made the appropriate modification to its model.

During the blowdown transient Westinghouse had included a rod-to-rod radiation model. Since a two-phase mixture may exist during a portion of the blowdown transient, the staff concluded that the presence of water droplets would reduce the transmission of rod-to-rod radiation during this phase of the LOCA and therefore required Westinghouse to remove the rod-to-rod radiation model from the blowdown phase of the calculation. Westinghouse has complied with this requirement.

2.3 Steam Interaction with ECC Water in PWR's

Westinghouse had not fully addressed the delay time required for ECC water to fall from the cold leg inlet to the bottom of the downcomer under the influence of gravity and steam drag. The staff required that both transport time and hot wall holdup time be considered and referenced the Block and Wallis correlation for hot wall delay time, as adequately reflecting available experimental data.

The staff provided a description of an acceptable hot wall time delay model, which Westinghouse has incorporated into their ECCS evaluation model. During the hot wall delay period, ECC water, which is delayed in passing through the downcomer, accumulates in available storage volumes in the following manner:

1) Lower downcomer - region between the bottom of the downcomer and the lower lip of the cold leg. A maximum of 1/3 of this volume will become available linearly over the hot wall delay period.

2) Upper downcomer - region of downcomer above the lower lip of cold leg pipe. If lower downcomer volume cannot accommodate all accumulator ECC water, some water will spill out the break. A storage volume is available in the upper downcomer region which is determined by the elevation head above the bottom of the cold leg and the break flow rate.

3) Cold leg piping between the injection point and the downcomer inlet is always available for storage.

Once the hot wall delay time has elapsed, and flow through the downcomer begins, a further period of time is required for the ECC water to flow from the available storage volumes to the lower plenum. To reflect this period a downcomer transport (free fall) delay time is calculated which is added to the hot wall delay time to yield the total time required for ECC water to travel from the cold leg inlet elevation to the bottom of the downcomer (lower plenum). The free fall delay is the time required for the ECC water to fall from the lower downcomer storage volume to the bottom of the downcomer. Once the hot wall delay time is ended and free fall starts, no further spillage of ECC water out the break would occur.

During accumulator injection the effect of non-condensibles on the $90^{o}F$ injection section pressure drop was not considered by Westinghouse in a manner which covered all data presently available to the staff. Westinghouse has complied with the staff requirement to incorporate values of injection section differential pressures of +1.8 psid for $90^{o}F$ injection in the presence of non-condensibles.

2.4 Refill and Reflood Heat Transfer

For flooding rates less than one inch per second the proposed steam cooling model was non-conservative relative to FLECHT data. Westinghouse has proposed to modify their steam cooling heat transfer model by adjusting the steam cooling film coefficient such that the integrated heat flux above the quench front is conservative relative to the FLECHT test data. The staff has reviewed the modified model, found it conservative relative to the FLECHT data, and has concluded that the proposed model change is acceptable. Westinghouse has incorporated the modification into their ECCS evaluation model.

3.0 APPLICABILITY OF GENERIC EVALUATION MODEL

The Westinghouse ECCS evaluation model, as submitted on August 5, 1974, was used to analyze the ECCS performance for the Robinson Power Station, Unit 2. Westinghouse has performed plant sensitivity studies for two-, three-, and four-loop plant designs (WCAP-8356)[3] and generic sensitivity studies (WCAP-8342)[17] which demonstrated the applicability of their model to a three-loop plant such as Robinson. The sensitivity studies were performed for both large and small breaks. The large break analyses were performed utilizing a double-ended cold leg guillotine break with various discharge coefficients and a range of split-type break sizes ranging from 1.0 ft^2 area to the full double-ended area of the cold leg. The small break spectrum was performed for cold leg split breaks ranging from an equivalent 2-inch pipe up to a 1.0 ft^2 break. Sensitivity studies for three-loop plant designs included:

(a) Break discharge coefficient and break location

(b) Reactor coolant pumps--tripped/running

(c) Burn-up sensitivity

(d) Skewed axial power profiles

(e) Worst single failure

(f) Small-large break interface

The staff reviewed these generic plant sensitivity studies and concluded that the generic evaluation model was appropriate and applicable for use in the evaluation of the ECCS performance for the Robinson Power Station.

4.0 RESULTS OF LOCA CALCULATIONS

As reported in the October 2nd submittal and in the generic plant sensitivity studies in WCAP-8356, the worst break was identified as the Double-Ended Cold Leg Guillotine type break in the pump discharge (DECLG) with a discharge coefficient of 0.4 ($C_D = 0.4$). This calculation resulted in a peak clad temperature of 2167°F, local metal-water reaction of 9.00% of the cladding thickness, and whole core metal-water reaction of less than 0.3%. These results were within the acceptable limits of the criteria of 10 CFR 50.46 (2200°F, 17.0%, and 1.0%, respectively), as shown in Table 4.1.

As stated in the Supplement to the staff Status Report on the Westinghouse ECCS evaluation model, for each plant analysis submitted, the applicant must provide and justify the plant dependent input assumptions used in the containment backpressure calculations. A letter was sent to CP&L (licensee) on November 4, 1974, requesting the submittal of additional information for purposes of further evaluation of the Robinson Unit 2 compliance with the Emergency Core Cooling System Acceptance

Criteria. The licensee responded to our request on December 4, 1974.[24] The staff examined the submittal information and concluded that no adjustment to the licensee's reported peak clad temperature due to the effect of containment backpressure was necessary.

All of the evaluation model deficiencies noted in Section 2.0 of this SER Supplement were rectified by Westinghouse in a manner acceptable to the staff. Westinghouse performed generic sensitivity studies to assess the impact of these required model changes upon the calculated peak clad temperature local metal-water reactions and whole-core metal-water reaction. The generic three-loop sensitivity studies were found to be applicable to the Robinson Unit 2 Power Station and the staff has utilized these studies to make an appropriate adjustment on the reported peak clad temperature.

Table 4.1 compares the Acceptance Criteria to the October 2nd licensee submittal and shows the effect of staff required model changes to the August 5th Westinghouse evaluation model. With regard to the staff adjustments, the calculated peak clad temperature (2167°F) was increased 40°F to reflect the modifications to the evaluation model. The sensitivity studies resulted in an increase in the maximum local metal-water reaction of less than one percent of clad thickness and the whole core metal-water reaction remained below 0.3 percent. In order to reduce the peak clad temperature to the 2200°F criteria, the allowable peaking factors are appropriately reduced by the requirements of Appendix A.

5.0 CONCLUSION

Based on the analysis set forth in this Safety Evaluation, ECCS cooling performance for the Robinson 2 facility will conform to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46(b) provided that the total peaking factor does not exceed a value of 2.306. Further restrictions to assure that operation will conform to the requirements are set forth as Appendix A hereto. These restrictions should be verified by a reanalysis based on the Westinghouse evaluation model, modified as described in this Safety Evaluation Report.

As described in the Status Report, the Robinson Unit also satisfies the two remaining criteria, i.e., maintenance of coolable geometry and long-term cooling. The residual heat removal system for the Robinson facility described in the Robinson SAR is satisfactory for these requirements.

An evaluation of ECCS performance wholly in conformance with 10 CFR 50.46 and Appendix K, based on an approved evaluation model*, should be submitted for this facility as soon as practicable, but within six months or before any refueling is authorized. In the interim, operation should conform to the requirements of the Interim Acceptance Criteria and the previously approved Technical Specifications, as well as the requirements of the licensee's submittal (21) and the requirements of Appendix A.

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*The model, which is wholly in conformance with Appendix K of 10 CFR 50.46, is described in a letter from Westinghouse dated December 6, 1974, from F. Bordelon to C. Berlinger (22) and in a letter from Westinghouse dated December 16, 1974 from F. M. Bordelon and W. J. Johnson to V. Stello.(23)

TABLE 4.1

	LOCA Analysis (10/2/74)	Criteria (1/4/74)	Adjusted Results (12/15/74)
Peak Clad Temperature ($^{\circ}F$)	2167	2200	2207 *
Max. Local Zr/H_2O Reaction (%)	9.00	17.0	<10.0
Total Zr/H_2O Reaction (%)	<0.3	1.0	<0.3

* This temperature is reduced to $2200^{\circ}F$ by the peaking factor limits given in Appendix A.

6.0 REFERENCES

1. Federal Register, "Docket No. RM-50-1, Part 50 - Licensing of Production and Utilization Facilities - Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Reactors," Vol. 39, No. 3, p. 1001 ff, January 4, 1974.
2. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, June 1974.
3. "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies," WCAP-8356, July 1974.
4. "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies," WCAP-8340, July 1974 (Proprietary).
5. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.
6. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301, June 1974 (Proprietary).
7. Bordelon, F. M., et al., "Containment Pressure Analysis Code (COCO)," WCAP-8326, July 1974.
8. Bordelon, F. M., et al., "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974 (Proprietary).
9. Grimm, N. P., et al., "Long Term Ice Condenser Containment Code - LOTIC Code," WCAP-8355, July 1974.
10. Grimm, N. P., et al., "Long Term Ice Condenser Containment Code - LOTIC Code," WCAP-8354, July 1974 (Proprietary).

11. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974.

12. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302, June 1974 (Proprietary).

13. Collier, G., et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8171, June 1974.

14. Collier, G., et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170, June 1974 (Proprietary).

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APPENDIX A

OPERATING LIMITS

1. The limiting F_Q without uncertainty in Specification 3.10.2.1, page 3.10.2 of the proposed Technical Specifications submitted on October 2, 1974, shall be 2.09 instead of 2.103 in the numerator of the second equation and at the end of item 3.10.2.1.b. The bases shall be changed accordingly.
2. In the proposed Technical Specifications, submitted on October 2, 1974, if the quadrant tilt is greater than 1.09, the action required by Specification 3.10.3.2.a, page 3.10-3, shall be revised to eliminate the words "and power range high flux setpoint".
3. If the APDMS is out of service as described in Specification 14.11.2 on page 4.11-2 of the proposed Technical Specifications, submitted on October 2, 1974, a log shall be kept of accumulated rod motion and time of manual traverses. Alternatively, if constant axial offset control procedures are used conformance with the applicable limit and the flux difference shall be logged hourly for the first 24 hours, and half-hourly thereafter.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 20, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON EVALUATION MODELS FOR COMMISSION CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

Dear Dr. Ray:

At its 175th meeting, November 14-16, 1974, the Advisory Committee on Reactor Safeguards completed a review of Evaluation Models which have been submitted in accordance with the Commission criteria set forth in 10 CFR 50.46. The following subcommittee meetings with reactor vendors were held in Washington, D. C.: March 28, 1974, Babcock and Wilcox; April 25, 1974, General Electric Company; April 26, 1974, Westinghouse Electric Corporation; and May 18, 1974, Combustion Engineering, Inc. Subcommittee meetings were held with the Regulatory Staff and their consultants in Washington, D. C., on August 6, 1974, September 28, 1974 and October 26, 1974. The Committee also had the benefit of the documents listed below. Previous reports to the Commission on interim acceptance criteria were made on January 7, 1972, and on the proposed changes on September 10, 1973. The Committee has also addressed the safety research programs and the latest report is on November 20, 1974.

The ACRS believes that the four light-water reactor vendors have developed Evaluation Models which, with the additional modifications required by the Regulatory Staff, will conform to Appendix K to Part 50.

Approved Evaluation Models will aid in conducting the licensing reviews, but a variety of specifics must be evaluated on a case-by-case basis. Items such as the particular features of a containment, sequencing of operations, single failure analysis and special features of the reactor design, are noted in the Staff's review of the vendor models. Additional items involving peaking factors and treatment of the uncertainties in the power distributions and monitoring of the power levels remain to be incorporated, case-by-case, in the Technical Specifications with appropriate conservatism.

The generic review of the vendor models proposed for Appendix K, like the reviews of the Interim Acceptance Criteria models, has contributed to improved understanding of the modeling techniques, including the applicability and limitations on current knowledge of thermal and hydraulic phenomena, and the need for more definitive safety research programs and code developments. The implementation of safety research programs, noted in the Committee's (November 20, 1974) report, and their results should have impact on the future evaluation methods and ECC systems.

The ACRS remains mindful that the Evaluation Models, in themselves are not the desired end products, but that effective, reliable emergency core cooling systems are the objective. The Committee acknowledges the contribution to reduced peak clad temperatures resulting from recent core design changes but reaffirms its position stated in the September 10, 1973 report that improved ECCS reliability and capability should continue to be sought and, to the extent practical, employed.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References Attached.

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