SAFETY EVALUATION REPORT

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-247

LICENSE NO. DPR-26

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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 September 6, 1974, Consolidated Edison (the Licensee) submitted an analysis of ECCS performance for the Indian Point Nuclear Plant Unit No. 2 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 Indian Point Unit No. 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 Indian Point 2 evaluation was based upon a model which was not acceptable, it also will require some changes. A revised set of computations for

Indian Point Unit 2 (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 Indian Point Unit 2 on September 6, 1974.

From a review of the September 6th 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 September 6, 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 Indian Point Unit 2 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 September 6, 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 Indian Point Unit No. 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 four-loop plant such as Indian Point Unit 2. 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 four-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 Indian Point Unit 2.

4.0 RESULTS OF LOCA CALCULATIONS

As reported in the September 6th 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 1.0 (C_D = 1.0). This calculation resulted in a peak clad temperature of 2110°F, local metal-water reaction of 7.40% 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 the licensee on November 4, 1974, requesting the submittal of additional information for purposes of further evaluation of Indian Point Unit 2 compliance with the Emergency Core Cooling System Acceptance

Criteria. The licensee responded to our request on December 2 and 6, 1974. 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 four-loop sensitivity studies were found to be applicable to the Indian Point Nuclear Plant Unit 2 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 September 6th 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 (2110°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.

5.0 CONCLUSION

Based on the analysis set forth in this Safety Evaluation, ECCS cooling performance for Indian Point Unit 2 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 2.32. 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, Indian Point Unit 2 also satisfies the two remaining criteria, i.e., maintenance of coolable geometry and long-term cooling. The residual heat removal system for the plant, as described in the Indian Point 2 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

*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 (9/6/74)	Criteria (1/4/74)	Adjusted Results (12/15/74)
Peak Clad Temperature (^{o}F)	2110	2200	2150
Max. Local Zr/H_2O Reaction (%)	7.4	17.0	<8.4
Total Zr/H_2O Reaction (%)	<0.3	1.0	<0.3

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).

15. Esposito, V. J., Kesavan, K., and Maul, B., "WFLASH - A Fortran-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8261, Revision 1, June 1974.

16. Esposito, V. J., Kesavan, K., and Maul, B., "WFLASH - A Fortran-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2, June 1974 (Proprietary).

17. Salvatori, R., "Westinghouse Emergency Core Cooling System Evaluation Model - Sensitivity Studies," WCAP-8342, June 1974.

18. Salvatori, R., "Westinghouse Emergency Core Cooling System Evaluation Model - Sensitivity Studies," WCAP-8341, June 1974 (Proprietary).

19. Federal Register, "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 1974.

20. Federal Register, "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 1974.

21. "ECCS Acceptance Criteria Analysis Indian Point Nuclear Generating Station Unit 2", Consolidated Edison Company, September 6, 1974.

22. Letter from F.M. Bordelon (WNES) to Carl Berlinger (AEC) dated December 6, 1974, "Modifications for Incorporation into the Westinghouse Appendix K ECCS Computer Models," #249-5097 (WNES).

23. Letter from F. M. Bordelon and W. J. Johnson (WNES) to V. Stello (AEC) dated December 16, 1974, regarding the Modification to the Westinghouse ECCS Computer Models and the Evaluation Model Modifications on Calculated Peak Clad Temperature.

APPENDIX A

INDIAN POINT UNIT 2

OPERATING LIMITS

1. To assure conformance with the flux difference band operating limits in the proposed Technical Specifications, submitted on September 6, 1974, two separate alarms are required on constant axial offset control procedures. One is an alarm to indicate nonconformance with the $\pm$ 5% flux difference band around the target value for operation at power levels greater than 90% of rated power. The other is an alarm to indicate nonconformance with the limit on time (one hour in twenty-four) that the 5% flux difference band may be exceeded for operation at or below 90% of rated power. If the alarms are temporarily out of service, and during the period before they are installed, conformance with the applicable limit and the flux difference shall be logged hourly for the first 24 hours, and half-hourly thereafter. The alarms shall be installed as soon as practicable but not later than March 1, 1975.

2. The operating limit on $F_q(Z)$ in Specification 3.10.2.1 on page 3.10-1 of the proposed Technical Specifications submitted on September 6, 1974, shall be $\leq (2.32/P) \times K(Z)$ for $P>0.5$ and $\leq (4.64) \times K(Z)$ for $P <0.5$. The bases shall be changed accordingly.

3. The time specified in the last line of Specification 3.10.2.8.1 on page 3.10-4 of the proposed Technical Specifications submitted on September 6, 1974, shall be 24 hours instead of 2 hours.

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4. The limit on $[F_j(Z)]_s$ in Specification 3.10.2.8.5 on page 3.10-4 of the proposed Technical Specifications submitted on September 6, 1974, shall employ the value 2.32 instead of 2.63. The bases shall be changed accordingly.