ECCS Jacket

SAFETY EVALUATION REPORT

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1 PLANT

DOCKET NO. 50-313

DEC 2 7 1974

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## 1.0 INTRODUCTION

On January 4, 1974, the Commission published its acceptance criteria for emergency core cooling systems for light water power reactors (39 FR 1003). This rule included Appendix K to 10 CFR 50 which specifies analytical techniques to be employed for the evaluation of ECCS effectiveness. On August 5, 1974, Babcock and Wilcox officially submitted a five volume package of topical reports [1,2,3,4,5] 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 Babcock and Wilcox, starting just prior to the publication of the Acceptance Criteria in January, 1974. The Regulatory staff reviewed these documents and published a Status Report on October 15, 1974, [6] which addressed each item required by Appendix K and identified areas which were acceptable to the staff and areas of concern which were to be resolved.

On November 13, 1974, the Regulatory staff published a Supplement to the Status Report [7] 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 Babcock and Wilcox approach. For certain other items, the staff concluded that adequate justification had not been provided and that further modification of the August 5, 1974 model was required. Babcock and Wilcox will modify their model to reflect these staff requirements and has evaluated the effect of all changes upon the previous calculations.(9) Accordingly, the Babcock and Wilcox model with the modifications presented in Section 2.0 and 4.0 of this SER is acceptable and would conform to Appendix K.

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

On August 2, 1974, Arkansas Power & Light Company (the licensee) submitted an analysis of ECCS performance for the Arkansas Nuclear One-Unit 1 (the plant or facility) along with proposed Technical Specification changes to reflect the new ECCS evaluation model calculations.(8) These evaluations were based upon the Babcock and Wilcox August 5, 1974 Evaluation Model. Section 3.0 of this SER discusses the applicability of the generic evaluation model to the specific Arkansas Unit 1 plant.

As stated in the Status Report and its Supplement, the August 5th Babcock and Wilcox Evaluation Model was not completely acceptable and specific model changes noted in the Status Report and its Supplement were required. These changes are now being made to the generic Babcock and Wilcox evaluation model. Since the Arkansas Unit 1 evaluation was based upon a model which was not acceptable, it also requires some changes. A revised set of computations for the plant (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 model changes made to the August 5, 1974 Babcock and Wilcox Evaluation Model, the staff requested, and Babcock and Wilcox submitted, a series of generic plant sensitivity studies which quantified the effect of the model changes on the results of the previous calculations.[9] The staff followed the performance of these sensitivity studies while they were in progress and has reviewed the results. These results are presented in Section 4.0 along with a discussion of their effects on the evaluation submitted for Arkansas Unit 1.[8]

From these studies, it appears that certain operating restrictions are required to ensure that in the event of a postulated loss-of-coolant accident, ECCS cooling performance will not exceed the values for calculated peak clad temperature and oxidation and hydrogen generation limits set forth in 10 CFR 50.46(b). These restrictions on maximum heat generation rate are set forth in the proposed Technical Specifications submitted on August 2, 1974, and are set forth in Appendix A hereto along with the other appropriate operating limits. To verify the limitations contained in the licensee's submittal of August 2, 1974, a reevaluation of ECCS performance in conformity with 10 CFR 50.46 and Appendix K, and based upon an approved evaluation model should be submitted for Arkansas Unit 1, along with appropriate Technical Specifications based on such evaluation, as soon as practicable.

During the interim, before an evaluation in conformity with the requirements of 10 CFR 50.46 can be submitted and evaluated, the facility should continue to conform to the requirements of the Commission's Interim Acceptance Criteria as well as the limitations contained in the licensee's proposed Technical Specifications in the submittal dated August 2, 1974. These requirements will provide reasonable assurance that the public health and safety will not be endangered.

## 2.0 BABCOCK AND WILCOX ECCS EVALUATION MODEL

The staff Status Report[6] provides a complete evaluation of the Babcock and Wilcox ECCS Evaluation Model*. Each part of 10 CFR 50, Appendix K was addressed and appropriate comments regarding compliance to each aspect of the model were included. All phases of the Babcock and Wilcox analytical methods were concluded to meet Appendix K requirements with the exceptions noted in Supplement 1 of the Status Report.[7] Of the fourteen areas of concern addressed in Supplement 1 to the Status Report, five were identified as model deficiencies for Oconee Class reactors (177 fuel assembly plants with a lowered loop arrangement) requiring modification or additional data to justify conformance to Appendix K. These areas are briefly discussed below. Additional detail of each deficiency is presented in Section 4.0 of this SER and in the staff Status Report.[6,7]

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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. These listings, combined with the Babcock and Wilcox impact studies,[9] constitute the currently acceptable ECCS model.

## 2.1 Metal-Water Reaction

The staff required that the Babcock and Wilcox ECCS model be revised to account for thinning of the oxide layer on the inside and outside of the fuel cladding. In addition, an improved calculational technique for arriving at a predicted value for total core-wide metal-water reaction resulted from staff comments. Babcock and Wilcox is modifying its ECCS model to incorporate these features. See Section 4.0 for an assessment of impact upon the current plant operating restrictions.

## 2.2 Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

As noted in the Status Report[6] and Supplement 1,[7] the staff accepted the Babcock and Wilcox modeling of swelling and rupture with three limitations. As discussed in Section 4.0 of this SER, these limitations were satisfied in the Arkansas evaluation.

Babcock and Wilcox has proposed to modify its model to incorporate a plastic swelling model, discussed in the Status Report Supplement, and a transient pin pressure model, which would eliminate two of the staff limitations. These modifications have not yet been completed. At present, the existing swelling and rupture model is acceptable if the staff limitations are observed.

2.3 End-of-Blowdown

As indicated in the Status Report and Supplement 1, the staff accepted the modeling of end-of-blowdown with the conditions that the definition of end-of-bypass be changed and that the downcomer noding representation be changed to use a homogeneous noding. Babcock and Wilcox is modifying its model to incorporate these changes. Section 4.0 discusses the impact of these deficiencies on the licensee's calculations.

2.4 Containment Pressure

Page 4-41 of the Status Report [(6)] states that the containment backpressure calculation performed for the Oconee Class plants is conservative and acceptable. For plants of a different type, specific input assumptions must be justified on an individual plant basis.

Although the backpressure model is acceptable for the Oconee Class plants, the effect of the use of the conservatively assumed parameters should be assessed by comparison with actual as-built values. Accordingly, the licensee has been requested to provide as-built values and to discuss the methods used to determine the passive containment heat sinks for the Arkansas Unit 1. Also required is an identification of each sink by category (e.g., cable tray, equipment supports, floor grating, crane wall) and surface area, thickness, materials of construction, thermal conductivity and volumetric heat capacity by component category. Values of paint thickness, thermal conductivity and volumetric heat capacity for containment internal structures are also requested.

## 2.5 Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors

Two concerns discussed by the staff in Supplement 1 to the Status Report[7] are related to the effect of hot walls on the ECC water being injected in the downcomer and the appropriateness of the value used for vent valve resistance. Babcock and Wilcox will modify their model to incorporate the resolution of these concerns. Section 4.0 assesses the impact of these concerns upon the plant operating restrictions.

## 3.0 APPLICABILITY OF GENERIC EVALUATION MODEL

As noted in BAW-10091[1] and in the staff's Status Report,[6] the development of the generic Babcock and Wilcox Evaluation Model involved the utilization of a plant design appropriate to all Oconee Class reactors. The series of sensitivity studies described in BAW-10091, Section 5.0 were therefore directly applicable to Arkansas Unit 1. Also worthy of note are the actual key parameters utilized in the generic model calculations. Babcock and Wilcox stated that they bounded the variations in key parameters within the Oconee Class plants by choosing values in their generic calculations which conservatively include any plant-to-plant variations. Table 1 provides a list of such key parameters employed in the generic evaluation and compares each parameter to the actual values for Arkansas Unit 1. This list shows that the generic calculation sufficiently incorporated the differences in these key parameters found in this plant.

## 4.0 RESULTS OF LOCA CALCULATIONS

From a break spectrum analysis, the worst break examined by Babcock and Wilcox using the August 5, 1974 model was an 8.55 $ft^2$ double-ended rupture at the reactor coolant pump discharge.[1] This generic analysis was the basis for the licensee's submittal.[8] This calculation resulted in a peak clad temperature of 2062 $^{o}F$, 3.38% local metal-water reaction, and 0.14% whole core metal-water reaction. These values are within the criteria of 10 CFR 50.46 (2200 $^{o}F$, 17%, and 1%, respectively).

All of the model deficiencies noted in Section 2.0 of this SER were examined by Babcock and Wilcox with regard to an impact assessment on current operating reactors.[9] The following sections address each of the relevant model deficiencies and their effects on the August 5, 1974 LOCA analysis.

4.1 Metal-Water Reaction

As indicated in Section 2.1 of this SER, the staff has requested that the Babcock and Wilcox ECCS Evaluation Model be revised to account for thinning of the oxide layer on the inside and outside of the fuel cladding. The generic model LOCA limit calculations[1] assumed initial values of 0.0001 inches oxide layer thickness and 1800 psia internal pin pressure. An oxide thickness sensitivity study conducted by Babcock and Wilcox[7] yielded the conclusion that the value of internal pin pressure combined with the value of the oxide thickness used by Babcock and Wilcox in their generic calculations conservatively predicted the highest peak cladding temperature for fuel cycle 1 operations. The Babcock and Wilcox study thinned the oxide layers consistent with the degree of pin swelling predicted.

The staff also noted in the Status Report that further justification was required to support the Babcock and Wilcox calculational technique for predicting total core-wide metal-water reaction. In the Supplement, the staff reported that

Babcock and Wilcox had chosen to modify their model in a manner which the staff found would be adequate. These modifications are now being made to the Babcock and Wilcox model. To determine whether this modification would affect the calculations submitted by the licensee, the staff considered sensitivity studies performed using staff models previously developed for confirmation of analyses submitted under the IAC. Although these models do not fully incorporate all required evaluation features, they are adequate to demonstrate that the results will fall well within the hydrogen generation criteria of 10 CFR 50.46(b)(3).

Therefore, this modification has no impact on the licensee's calculations.

4.2 Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

As noted in the Status Report,[6] the staff accepted the generic calculation if three limitations were observed:

a) The internal pin pressure selected for the initial condition value must exceed the maximum predicted during normal operation for the design being analyzed.

b) If the rod with the highest peak clad temperature ruptures, then the time of rupture is restricted to a time period prior to the end of blowdown.

c) 70% circumferential swelling for certain rupture temperatures must be employed. It is permissible to increase linearly from 1700 °F (about 40% circumferential swelling) to the 70% plateau at 2000 °F.

The Arkansas analysis satisfies each of these limitations (maximum pin pressure was assumed, ruptures occurred prior to end of blowdown, and rupture temperatures were less than 1700 °F). Accordingly, there is no impact on the licensee's calculation.

4.3 End-of-Blowdown

Since the generic calculation [(1)] showed that end-of-bypass always occurred prior to, or at the same time as, end-of-blowdown, the model change regarding the definition of end-of-bypass has no effect on peak clad temperatures for this plant.

With regard to the staff concern that the downcomer model did not appear to be properly represented, Babcock and Wilcox has now changed the downcomer noding to a homogeneous noding representation as required in the Status Report Supplement. In connection with this change, a number of other areas previously modeled on a heterogeneous basis have also been changed to homogeneous noding. . This is acceptable. These modifications will require related changes to the generic model sensitivity studies. These are being performed by Babcock and Wilcox. However, in assessing the impact of this required change upon the calculations made using the August 5th model, Babcock and Wilcox found that two counteracting phenomena occur to result in an overall decrease in peak clad temperature at the 6-foot elevation of about 80 $^{o}$F. Although less water remains in the vessel at the end-of-bypass (leading to a longer adiabatic heatup), reduced water head in the downcomer allows a significantly higher negative flow through the core for a longer period of time. As previously indicated, the overall effect is to decrease the peak clad temperature, especially at the higher core elevations.

4.4 Containment Pressure

For the reasons stated in Section 2.4 of this SER, staff concerns in the area of the containment backpressure calculation have no effect on the licensee's calculations.

4.5 Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors

As noted in Supplement 1 (7) to the Status Report, the staff required that Babcock and Wilcox correct the vent valve resistance (K) for two-phase flow by applying a factor of 1.5 to the single phase value. With respect to the vent valve flow resistance factor used by Babcock and Wilcox (K = 3.9), the staff required correction of this factor for two-phase flow. As indicated in the Supplement, a correction factor of C = 1.5 based upon appropriate experimental data for gate valves was proper along with a further correction to account for the pressure dependence of C. In the Reynolds number range of interest during reflood (starting with a reference K of 3.3 based on single-phase data), a multiplier of 0.85 is acceptable to correct for pressure effects. Therefore, the required vent valve K-factor to be used in reflood calculations is:

$$K = 3.3 \times 1.5 \times 0.85 = 4.2$$

Babcock and Wilcox will modify its model to use this value. Various sensitivity studies were performed by Babcock and Wilcox to assess the impact of this change of assumed vent valve K. The results of these studies showed that an increase in vent valve resistance from the value of 3.9 used in the generic calculations[1] to 4.2 showed about a $20^{\circ}$F increase in peak clad temperature.

With regard to the effect of hot walls on the ECC water being injected in the downcomer, the staff has provided Babcock and Wilcox a description of an acceptable hot wall time delay model.[6] 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 the lower downcomer volume cannot handle all accumulator ECC water, some water will spill out the break. A storage volume is available in the upper downcomer which is determined by the elevation head above the bottom of the cold leg. The same elevation head should be used to determine the break flow rate.

3) Cold leg piping from the reactor coolant pump discharge to the vessel nozzle. A storage volume consistent with the upper downcomer water level is available.

Once the hot wall delay time has elasped 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 fall from the inlet elevation to the bottom of the downcomer (lower plenum). Once the hot wall delay time is ended and free fall starts, no further spillage of ECC water out the break would occur. Babcock and Wilcox has indicated that sufficient storage capacity exists to account for the volume of water which could be accumulated during the hot wall delay time. Therefore, there is no net change in the generic calculation due to hot wall effects.

4.6 Summary of Results

A review of preceding Sections 4.1 through 4.5 shows that the two model deficiencies which have an impact on the previous generic calculations are region noding (Section 4.3) and vent valve K-factor (Section 4.5). Table 2 shows a summary of the results of sensitivity studies by Babcock and Wilcox on peak clad temperature, local metal-water reaction, and whole core metal-water reaction. These calculations indicate that, while the model corrections could cause an increase in peak clad temperature, this increase would not be large enough to exceed the criteria of 10 CFR 50.46, provided that the LOCA limit curves submitted in the licensee's proposed Technical Specifications are observed in facility operation. These curves are set forth in Appendix A.

5.0 CONCLUSIONS

Based on the analysis set forth in this Safety Evaluation, the limitations contained in the licensee's submittals, particularly the LOCA limit curve set forth in Appendix A, will assure conformance with the peak clad temperature limit, and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46(b). However, these restrictions should be verified by a re-analysis based on the Babcock and Wilcox Evaluation Model, modified as described in this Safety Evaluation Report.

In addition, Arkansas Unit 1 satisfies the two remaining criteria, i.e., maintenance of a coolable geometry[6] and long-term cooling.[10] The heat removal system for long-term cooling of the plant as described in the FSAR 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. During the interim, until each evaluation is submitted and evaluated by the staff, operation should conform to the requirements of the Interim Acceptance Criteria, as well as to the requirements of the licensee's submittals[8] as indicated in Appendix A.

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*The Babcock and Wilcox ECCS Evaluation Model, which is wholly in conformance with Appendix K of 10 CFR 50.46, is described in a letter from Babcock and Wilcox dated December 18, 1974.[9]

TABLE 1

A COMPARISON OF ARKANSAS UNIT 1 TO KEY PARAMETERS EMPLOYED IN THE GENERIC EVALUATION MODEL

| PARAMETER | GENERIC MODEL | UNIT 1 |
|---|---|---|
| Rated Core Power, Mwt | 2,772 | 2,568 |
| Reactor Vessel Flow, lbm/sec (1) | 38,306 | 39,467 |
| Reactor Coolant System Pressure at Core Outlet, psig | 2,182 | 2,185 |
| Core Inlet Fluid Temperature, $^{o}F$ | 556 | 556 |
| Volume Average Fuel Temperature at 18 Kw/ft with a Sink Temperature of 580 $^{o}F$, $^{o}F$ | 3,105 | 3,050 |
| ECCS Delay Time, seconds | 35 | 25 |
| Reactor Building Free Volume $ft^3$ | $2.205 \times 10^6$ | $1.85 \times 10^6$ |

1. Flows are total systems flows because core flow is not measured.
2. These are estimates since full power has not yet been achieved. Other vessel flows and fluid temperatures are measured values.

TABLE 2. SUMMARY OF SENSITIVITY STUDIES

($8.55\ ft^2$ Double-Ended Rupture)

| Kw/ft | Axial Position, ft | *Peak Clad Temperature, F | *Local M-W Reaction, % | *Whole-Core M-W Reaction, % |
|---|---|---|---|---|
| 16.0 | 2 | 2167 | 3.77 | <0.5 |
| 17.5 | 4 | 2112 | 3.01 | <0.5 |
| 18.0 | 6 | 2122 | 3.53 | <0.5 |
| 17.1 | 8 | 2059 | 2.21 | <0.5 |
| 16.0 | 10 | 1877 | 1.68 | <0.5 |

*CRITERIA

Peak clad temperature.............2200 °F
Local Metal-Water Reaction........ 17%
Whole-Core Metal-Water Reaction... 1%

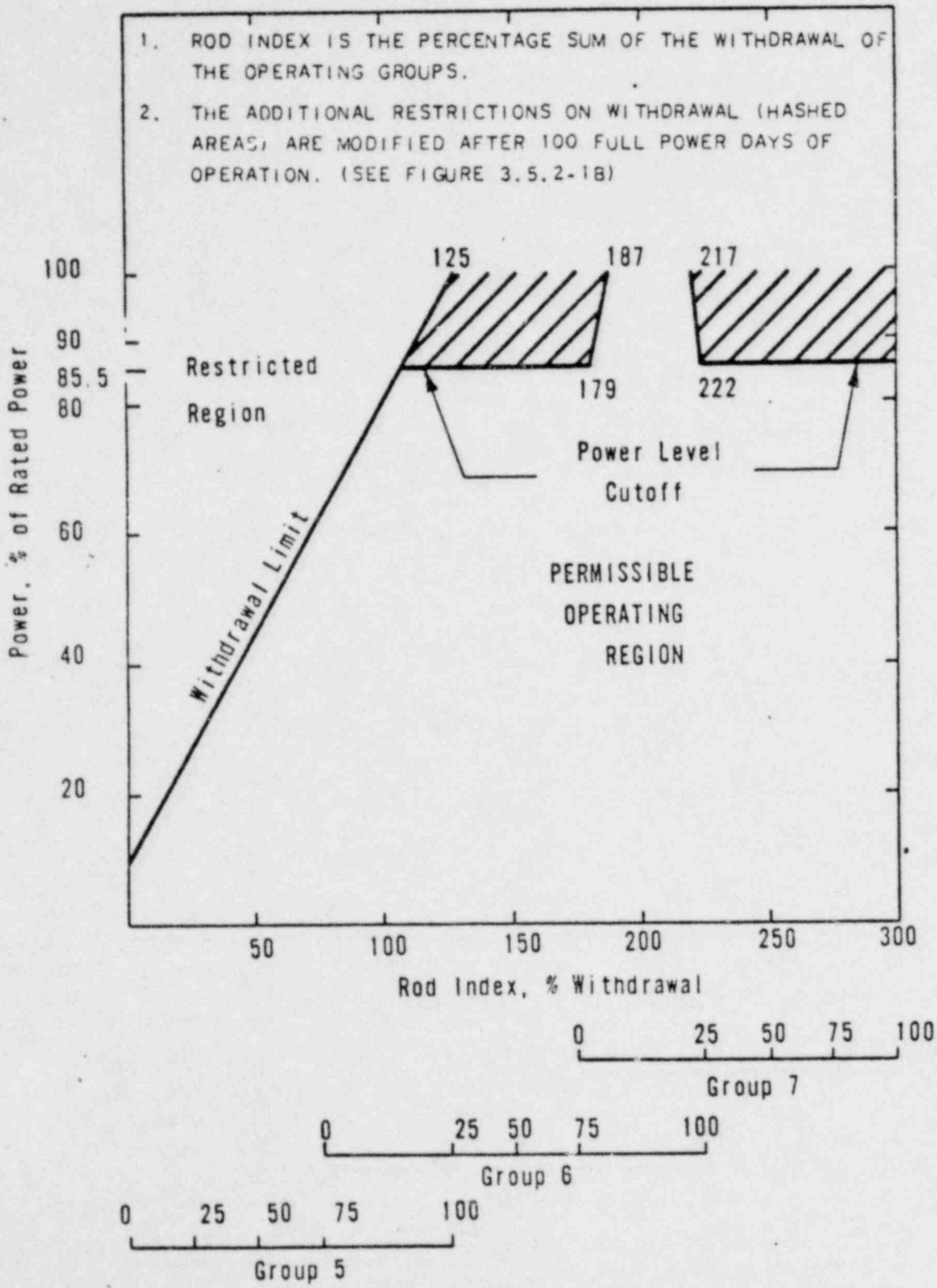
## 6.0 REFERENCES


1. BAW-10091, "B&W's ECCS Evaluation Model Report with Specific Application to 177 FA Class Plants with Lowered Loop Arrangement," August 1974.
2. BAW-10092, "CRAFT2-Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant," July 1974.
3. BAW-10093, "REFLOOD - Description of Model for Multinode Core Reflood Analysis," July 1974
4. BAW-10094, "Revisions to THETA 1-B, A Computer Code for Nuclear Reactor Core Thermal Analysis," IN-1445, July 1974.
5. BAW-10095, "CONTEMPT - Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," July 1974.
6. "Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," October 1974.
7. "Supplement 1 to the Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," November 13, 1974.
8. Letter from J. D. Phillips to Mr. A. Giambusso dated August 2, 1974.
9. Letter from James F. Mallay to T.M. Novak dated December 18, 1974.
10. Letter from James F. Mallay to T. M. Novak dated November 25, 1974.

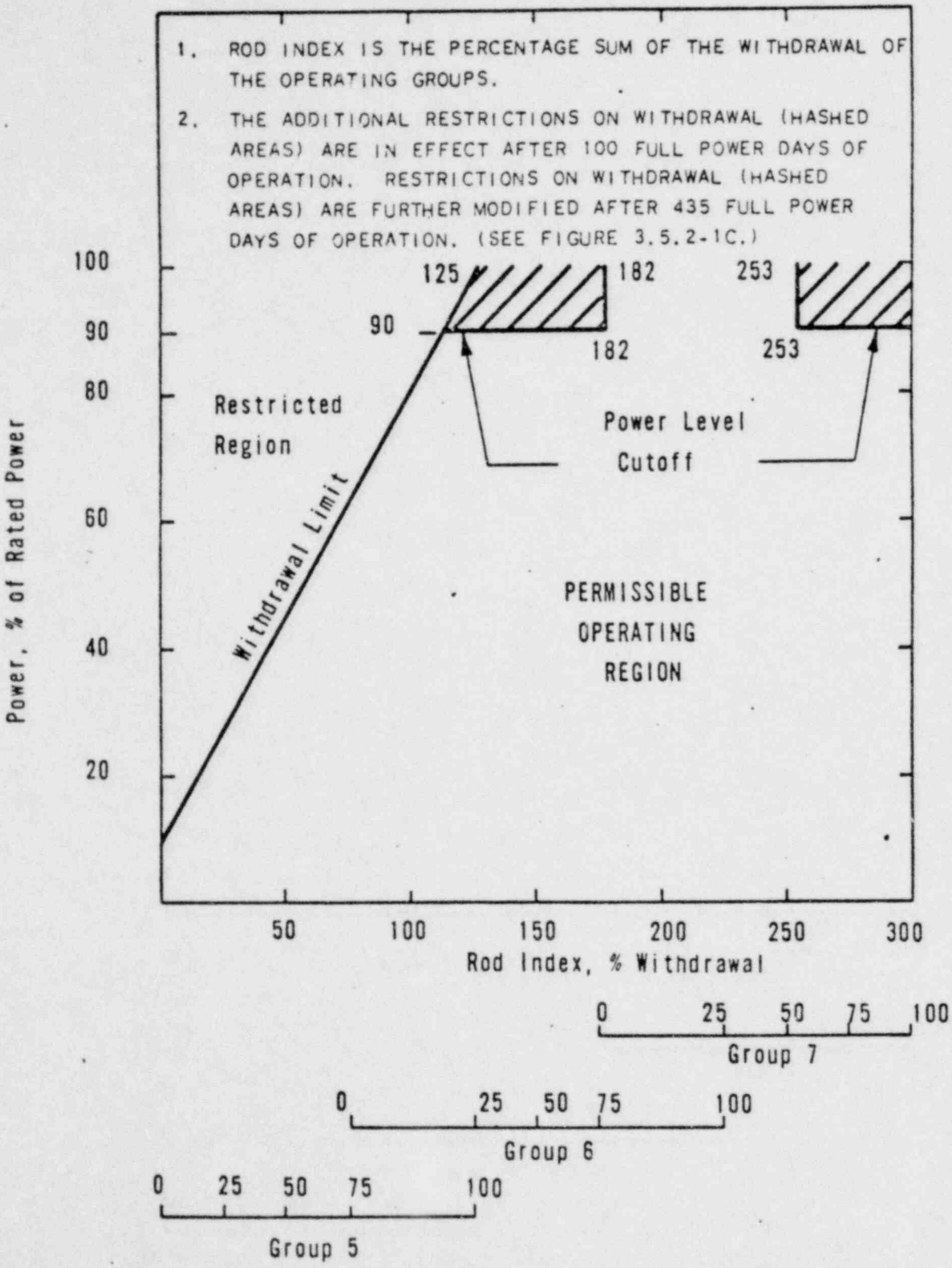
## APPENDIX A

## OPERATING RESTRICTIONS

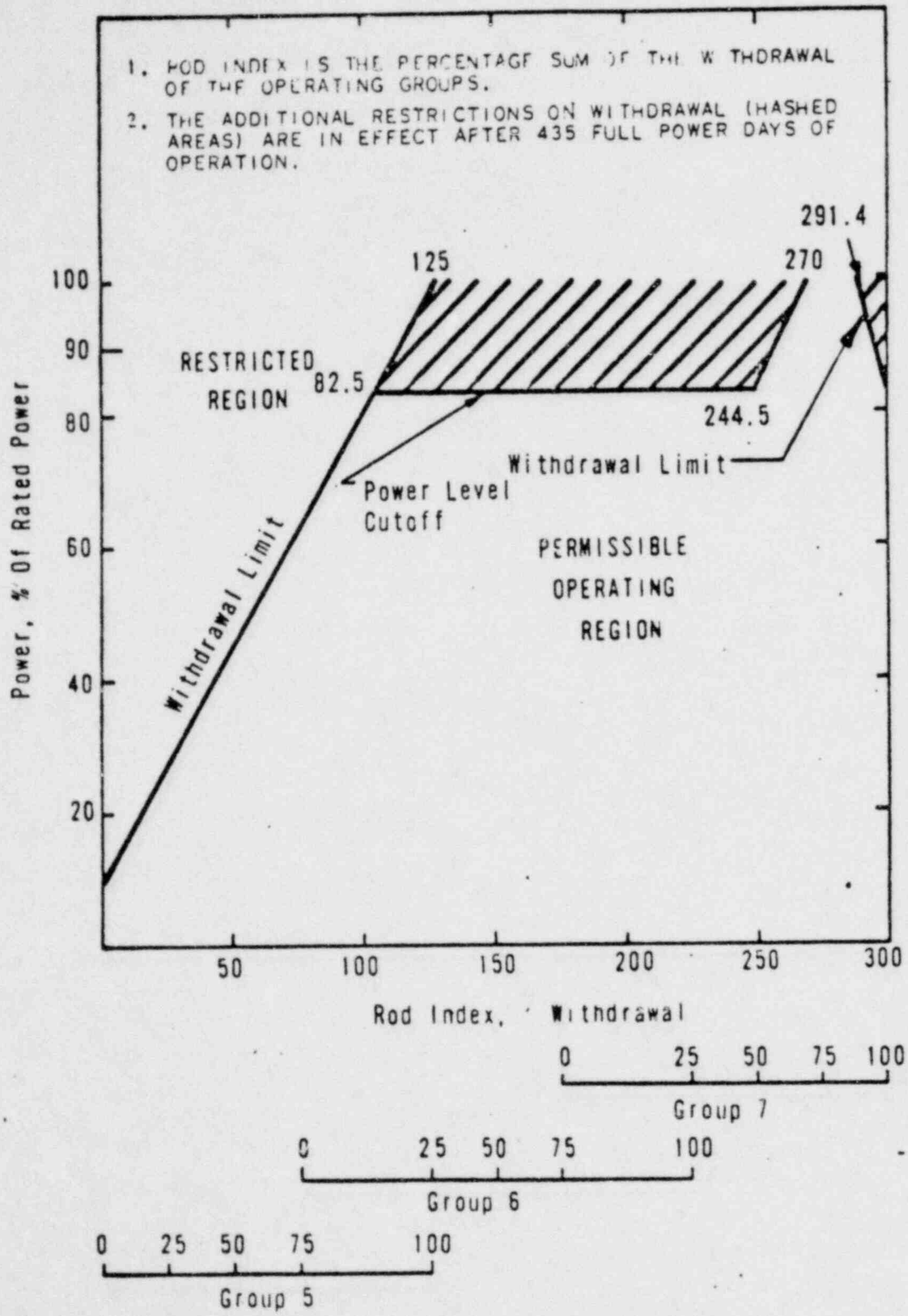
The Regulatory staff has reviewed the methods used by Babcock and Wilcox to derive the LOCA-related operating limits for its plants. The review considered the basic calculation method, the range of operating conditions calculated, the types of uncertainties and their magnitude, and the instrumentation provided to monitor plant operation. Based on this review, we conclude that sufficient monitoring instrumentation is present to provide assurance that the plant may be operated within LOCA-related operating restrictions. We further conclude that operation of Arkansas Unit 1, within the restrictions shown on Figures 3.5.2-1A through 3.5.2-1C, which were a part of the August 2, 1974 proposed Technical Specifications from the licensee, will assure that the heat generation limits of Figure 3.5.2-4 will not be exceeded. It should be understood that the operating restrictions for Arkansas Unit 1 presented in this SER are to be observed in addition to those operating restrictions in effect under the Interim Acceptance Criteria.



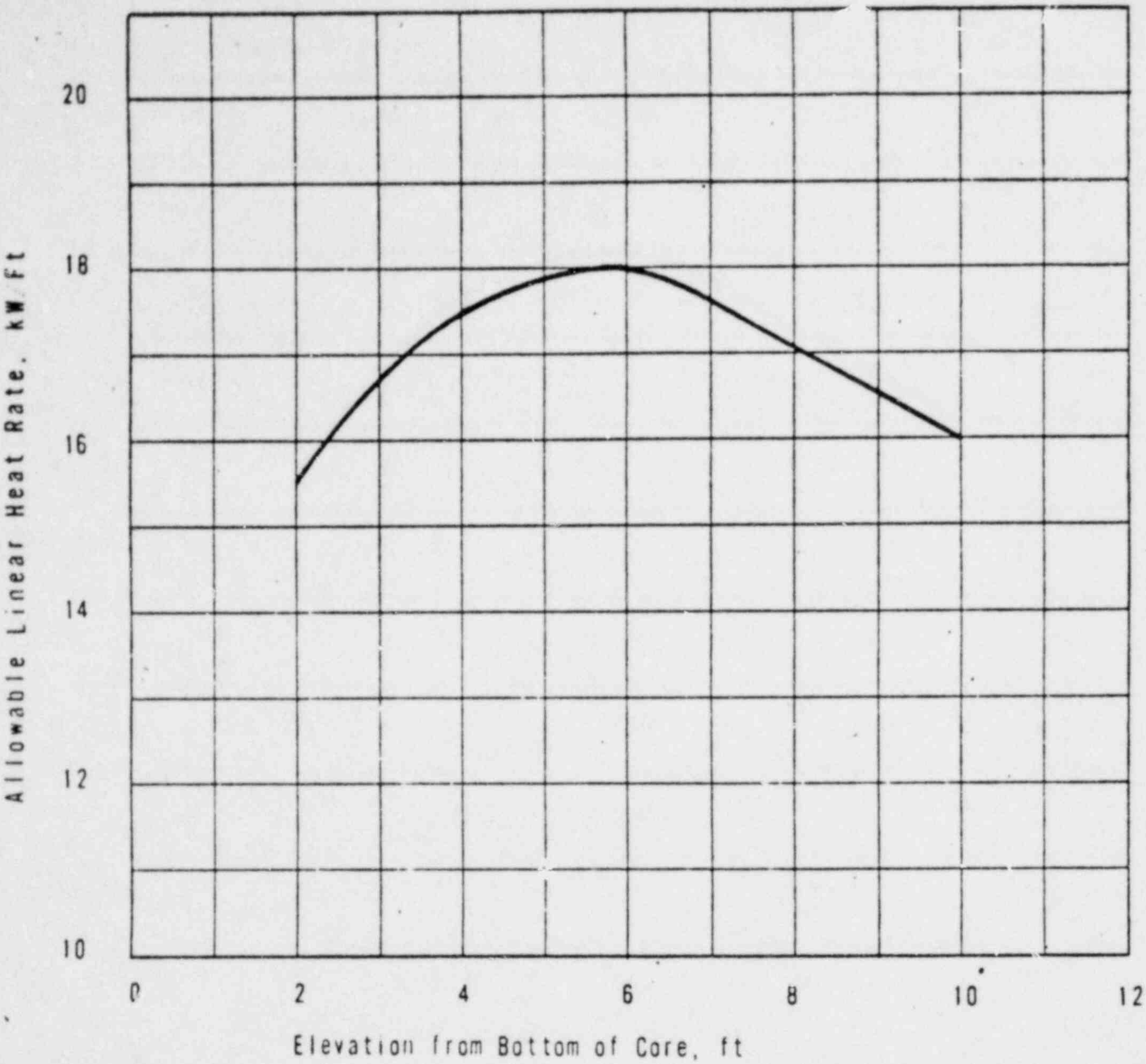

| | | |
|---|---|---|
| KANSAS POWER & LIGHT CO.<br>KANSAS NUCLEAR ONE-UNIT 1 | CONTROL ROD GROUP WITHDRAWAL<br>LIMITS FOR 4 PUMP OPERATION | FIG. NO.<br>3.5.2-1A |




| ARKANSAS POWER & LIGHT CO.<br>ARKANSAS NUCLEAR ONE-UNIT 1 | CONTROL ROD GROUP WITHDRAWAL<br>LIMITS FOR 4 PUMP OPERATION | FIG. NO.<br>3.5.2-1B |
|---|---|---|




| ARKANSAS POWER & LIGHT CO.<br>ARKANSAS NUCLEAR ONE-UNIT 1 | CONTROL ROD GROUP WITHDRAWAL<br>LIMITS FOR 4 PUMP OPERATION | FIG.NO.<br>3.5.2-1 C |
|---|---|---|




LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE

| ARKANSAS POWER & LIGHT CO.<br>ARKANSAS NUCLEAR ONE-UNIT 1 | LOCA LIMITED MAXIMUM ALLOWABLE<br>LINEAR HEAT RATE | FIG. NO.<br>3.5.2-4 |
|---|---|---|

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.

References


1) WCAP-8170 (P) dated June 1974, "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident"
2) WCAP-8200 (Rev. 2 (P)) dated June 1974, "WFLASH - A Fortran - IV Computer Program for Simulation of Transients in a Multi-Loop PWR"
3) WCAP-8301 (P) dated June 1974, "LOCTA - IV Program Loss-of-Coolant Transient Analysis"
4) WCAP-8302 (P) dated June 1974, "SATAN IV Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant"
5) WCAP-8327 (P) dated July 1974, "Containment Pressure Analysis"
6) WCAP-8339 (NP) dated June 1974, "Westinghouse Emergency Core Cooling System Evaluation Model, Summary"
7) WCAP-8340 dated July 1974, "Westinghouse ECCS - Plant Sensitivity Studies"
8) WCAP-8341 (P) dated July 1974, "Westinghouse Emergency Core Cooling System Evaluation Model Sensitivity Studies"
9) WCAP-8354 (P) dated July 1974, "Long Term Ice Condenser Containment Code - LOTIC Code"
10) BAW-10091, "B&W's ECCS Evaluation Model Report with Specific Application to 177 FA Class Plants with Lowered Loop Arrangement," August 1974
11) BAW-10092, CRAFT2 - Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant," July 1974
12) BAW-10093, "REFLOOD - Description of Model for Multinode Core Reflood Analysis," July 1974
13) BAW-10094, "Revisions to THETA 1-B, A Computer Code for Nuclear Reactor Core Thermal Analysis," IN-1445, July 1974
14) BAW-10095, "CONTEMPT - Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," July 1974
15) CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model," (P) August 1974
16) CENPD-134 (P) April 1974, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core"
17) CENPD-135 (P) April 1974, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program"
18) CENPD-136 (P) July 1974, "High Temperature Properties of Zircaloy and $UO_2$ for Use in LOCA Evaluation Models"
19) CENPD-137 (P) August 1974, "Calculative Methods for the CE Small Break LOCA Evaluation Model"
20) CENPD-138 (P) April 1974, "PARCH - A Fortran IV Digital Computer Program to Evaluate Pool-Boiling Axial Rod and Coolant Heatup"
21) CENPD-139, "CE Fuel Evaluation Model (FATES Thermal Performance and Densification Model)"
22) NEDO-20566 (DRAFT) (P) - "General Electric Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K"
23) Status Report By The Directorate of Licensing In The Matter Of Babcock and Wilcox ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K (undated)