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SUBJECT: Requests review & approval of "AEP REactor Core Thermal-Hydraulic Analysis COBRA IIIC/MIT-2 Computer Code."

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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
REQUEST FOR REVIEW OF AEP REACTOR CORE THERMAL-HYDRAULIC
ANALYSIS USING THE COBRA IIIC/MIT-2 COMPUTER CODE

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

Dear Dr. Murley:

The purpose of this letter is to request your review and approval of the topical report entitled "AEP Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT-2 Computer Code." To facilitate your review, we have enclosed 10 copies of the report. The thermal-hydraulic methodology contained in this report is applicable to both units of the Donald C. Cook Nuclear Plant. We intend to use this methodology for licensing applications beginning with Cycle 12 of Unit 1. Cycle 12 operation is scheduled for early 1991; therefore, your approval of this report is requested by July 1990.

Development of the topical report is consistent with the NRC goals as outlined in Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions." In that letter, the NRC acknowledged and encouraged the efforts of utilities to develop the capability to perform their own safety analyses using large, complex thermal-hydraulic computer codes. Generic Letter 83-11 stated that licensees should demonstrate their proficiency in using the code by performing code verification in-house and submitting a report to the NRC for review.

The topical report contains seven sections. Section 1 is an introduction. Section 2 describes briefly the COBRA IIIC/MIT-2 computer code and various correlations/options used in the analysis. The hydraulic and thermal models are described in Sections 3 and 4, respectively. The engineering uncertainties applied to the thermal-hydraulic model are described in Section 5. Section 6 describes the specific analyses performed and comparisons of American Electric Power (AEP) results with those given in the licensing documents. Conclusions are provided in Section 7. The references cited in the topical report are available in the open literature.

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The COBRA IIIC/MIT-2 computer code is a public domain code and was obtained from the Massachusetts Institute of Technology. The code was installed, verified, and validated on the AEP computer system following our corporate software Quality Assurance Program. The code manual (Reference 5 of the topical report) includes the input preparation method and sample problems for the pressurized water reactor core. These sample problems were followed to model the core of Unit 1 of the Cook Nuclear Plant. The limitations applicable to this methodology are delineated in the text of the topical report.

The thermal-hydraulic methodology described in the topical report is based on a single stage method, which is described in Reference 1 of the report. This method has previously been used by the Virginia Electric Power Company (VEPCO) for departure from nucleate boiling (DNB) analyses for the Surry Nuclear Power Station. VEPCO submitted their topical report to the NRC on September 28, 1979, and it was accepted by your staff on August 26, 1983, for licensing application.

The accuracy of the thermal-hydraulic methodology contained in this report has been verified using data from Cycle 1 of the Cook Nuclear Plant Unit 1. Three steady state and three transient DNB analyses were performed using this methodology. For the transients analyzed, the minimum DNB results obtained are within ±1.5 percent of those given in the original Final Safety Analysis Report (FSAR) for Unit 1 of the Cook Nuclear Plant. The Unit 1 Cycle 1 FSAR analyses were performed using Westinghouse Electric Corporation's THINC computer code.

The methodology presented in this report will be initially used for plant operational support, licensee event report evaluations, and other FSAR type DNB analyses. Our ultimate use of the methodology will be for reload evaluations and licensing submittals to the NRC.

A check in the amount of $150.00 is enclosed with this letter for NRC processing of the aforementioned request.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

MPAlexich

M. P. Alexich
Vice President

ldp

Attachments

cc: D. H. Williams, Jr.
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
A. B. Davis
NRC Resident Inspector - Bridgman
G. Bruchmann