ATTACHMENT A

LOCA ECCS ANALYSIS WITH
ZIRC/WATER REACTION CORRECTION

Indian Point Unit No. 2

Docket No. 50-247

April 17, 1978

LOCA ECCS ANALYSIS WITH ZIRC/WATER REACTION CORRECTION

Introduction

We have been informed by Westinghouse Electric Corporation that a logic inconsistency exists in two of the computer codes used in their LOCA ECCS Evaluation Model. The SATAN-VI [1] and LOCTA-IV [2] codes are the affected computer codes. This logic inconsistency involves the interface between the zirconium-water reaction heat generation calculation and the heat conduction equation. The inconsistency underestimates the volumetric heat flux due to zirconium-water reaction by a factor of 2.

Westinghouse Electric Corporation has studied the effect of correcting this error on calculated peak clad temperature. In addition to correcting this error, some beneficial model changes were also studied. The results of their studies indicate a net increase in peaking factor. Some details of these calculations follow.

Proposed Solution

Westinghouse Electric Corporation has proposed the use of the following improvements to the October 1975 version of their evaluation model:

1. Change the transition boiling correlation used during blowdown from the W Transition Boiling Correlation to the Dougall-Rohsenow. Both correlations have been documented by Westinghouse [3] and both are termed "acceptable" in Appendix K of 10CFR50.46 and the NRC SER for the Westinghouse evaluation model.

2. Use of an emissivity in the refill heat transfer model of 0.9 [4].

3. Multiply the volumetric heat flux from the zirconium-water reaction calculation by a factor of 2 to correct the logic inconsistency.

4. Use of maxi-convolution to improve the peaking factors being calculated [5].

5. Use of a new 15X15 FLECHT correlation [6].

All of the modifications were discussed with the staff on March 29, 1978. We understand that it will take the staff approximately 3 months to review all of these model changes. We will work with Westinghouse until then to arrive at a new approved LOCA ECCS Evaluation Model. At that time we will submit to the Nuclear Regulatory Commission a schedule for reanalysis of the present limiting break size with the new model. Until then we plan no further analyses.

Interim F_Q Basis

In addition, the Nuclear Regulatory Commission has requested the justification for operating at a peaking factor, F_Q = 2.24 until a new analysis is performed later this year. The justification for interim operation at F_Q = 2.24 is a Westinghouse estimate of F_Q = 2.41 which is greater than the staff's requested interim F_Q. The Westinghouse estimate was based on the use of the October 1975 evaluation model modified as described above. The estimates have been confirmed by many actual calculations.

We understand that we are also being requested to document the calculations involved in deriving the NRC estimate of F_Q = 2.24. Some portions of this calculation were impacted by input from Westinghouse. Other portions of the calculation were performed by the NRC staff and although we understand the calculations we cannot provide documentation, since neither Westinghouse nor ourselves developed those portions. Those portions will be noted. The attached Table 1 summarizes the various single effect changes to our peaking factor along with the basis for each.

Based on an F_Q of 2.24, we have evaluated the need for additional incore monitoring beyond presently required monitoring specified in the Technical Specifications. A subset of the "18 Case" total peaking factor calculation was re-evaluated. The values of radial peaking factors employed are the same as those previously employed. The results are shown on Figure 1. As can be seen, CAOC operation, as presently specified in the Technical Specifications, is sufficient to meet the imposed F_Q limit and justify full power operation for Cycle 3 without additional requirements.

Our current monitoring program (i.e., monthly flux maps) will be continuted and we will administratively meet the revised peaking factor limits, which are

$$F_Q(z) = \begin{cases} (2.24/P) \times K(z), & P \geq 0.5 \\ (4.48) \times K(z), & P < 0.5 \end{cases}$$

where P is the normalized core power level relative to the rated core power level and the K(z) function is defined by Figure 2.

References

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

2. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

3. "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339 Bordelon, F. M., Massie, H.W., and Zordan, T. A., July 1974.

4. "High Temperature Properties of Zircaloy - Oxygen Alloys," EPRI Report NP-524, March, 1977.

5. Little, C.C., et al., "Consideration of Uncertainties in the Specification of Core Hot Channel Factor Limits," WCAP-9180, September 1977.

6. "Westinghouse ECCS Evaluation Model, February, 1978 Version," WCAP-9220, February, 1978 (Proprietary), WCAP-9221, February, 1978 (Non-Proprietary).

TABLE 1

INDIAN POINT UNIT NO. 2

I. Current Analysis: $F_Q = 2.32$, Peak Clad Temperature = 2087°F.

II. Changes to Current Analysis:

Modification	F_Q Change	Justification/Basis
Zirc/Water Correction	-0.20	NRC
Current Analysis Margin to 2200°F	+0.11	NRC (25°F/%F_Q)
ESDR Power Used in Current Analysis	+0.01	Margin identified by Westinghouse
Net Change	-0.08	

Current F_Q	Plus	Net Change	=	New F_Q
2.32	+	-0.08	=	2.24

Figure 1 - Maximum $\left[F_Q^T \cdot P_{Rel}\right]$ Versus Axial Core Height

During Normal Operation

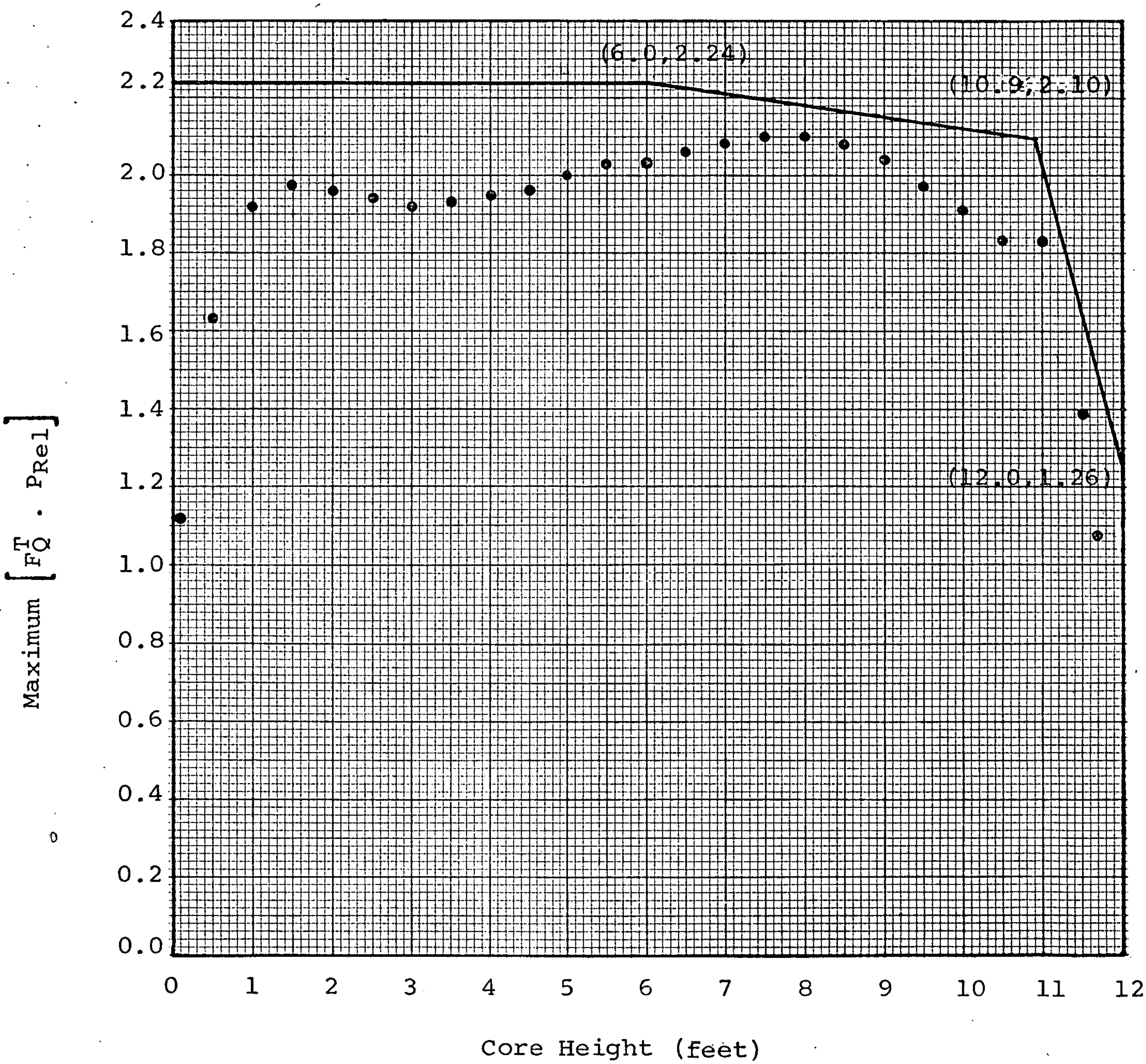
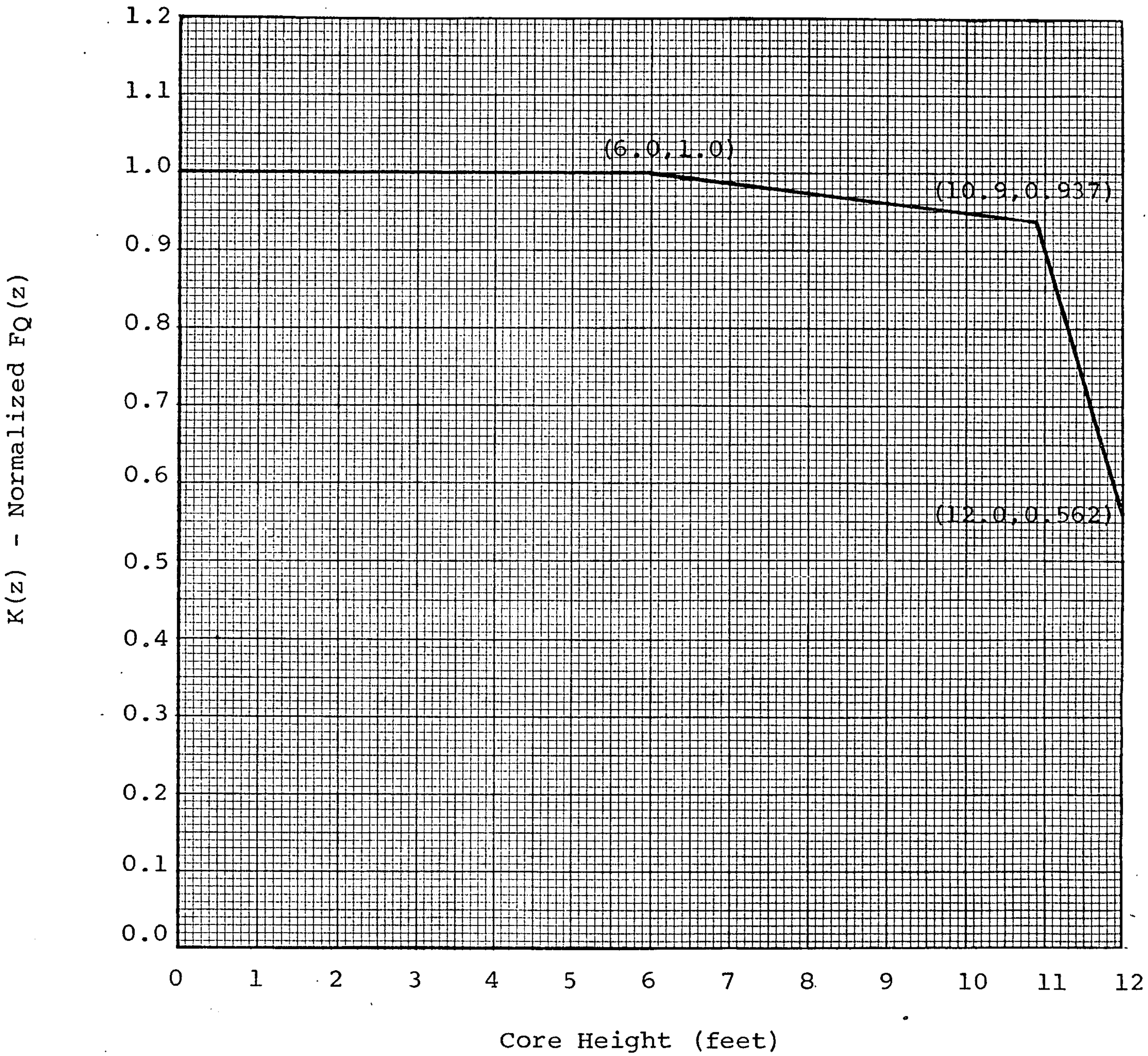


Figure 2 - Indian Point Unit No. 2 F_Q^T Normalized

Operating Function, K(z)



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

April 14, 1978

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To ~~All~~ Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

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Enclosures:
1. NRC Guidance
2. Notice

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