ATTACHMENT I

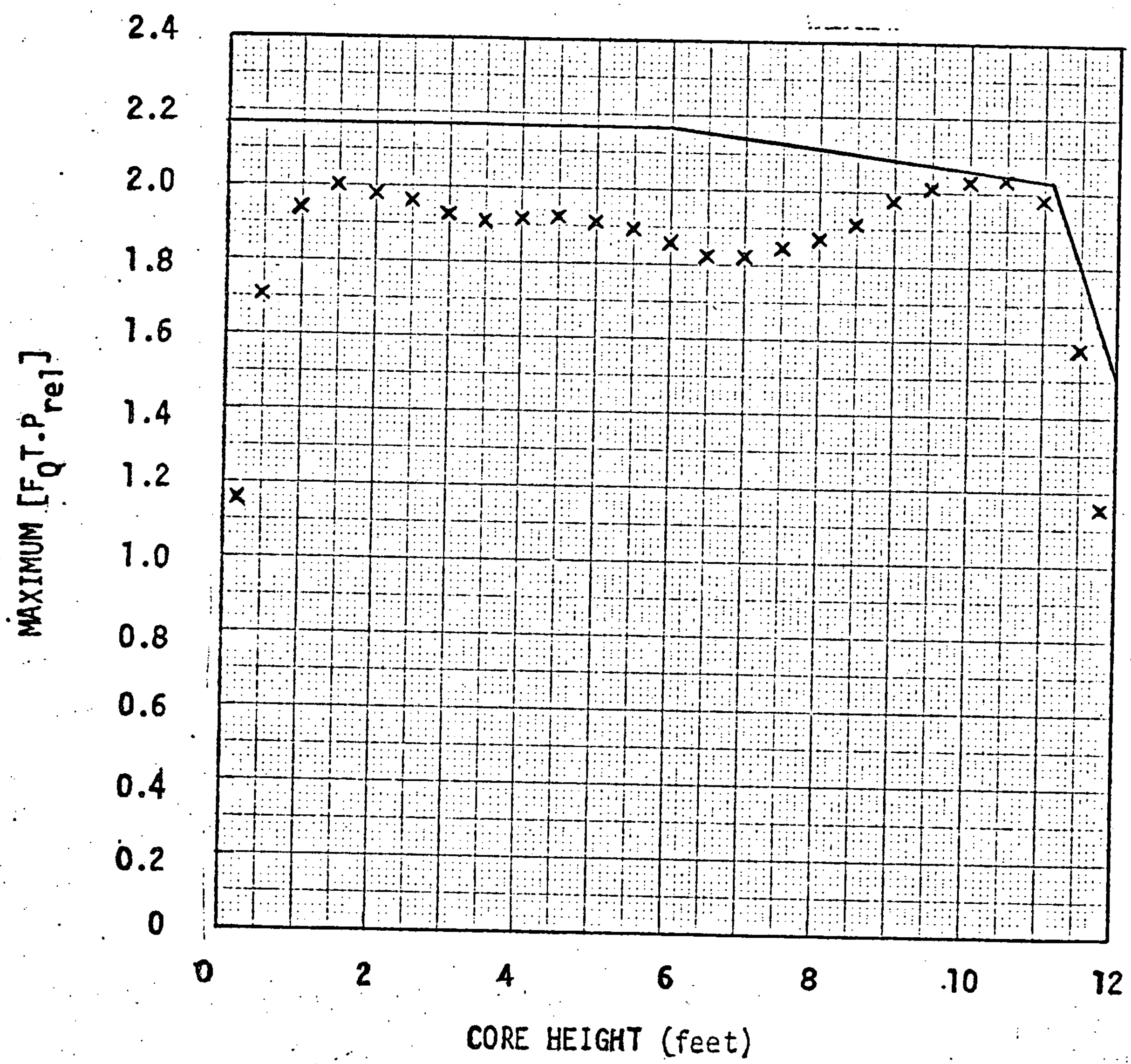
Indian Point 3
Cycle 2 Eighteen (18) Case
FAC Reanalysis

Power Authority of the State of New York

Indian Point Unit No. 3
Docket No. 50-286
May , 1978

MAXIMUM [F_QT.P_{rel}] vs AXIAL CORE HEIGHT

DURING NORMAL CORE OPERATION



INDIAN POINT NO. 3 NUCLEAR POWER PLANT

INCOMING CORRESPONDENCE - COVER AND ACTION SHEET

PASNY
RECEIVED

APR 19 1978

INDIAN POINT
NUCLEAR GENERATING
UNIT NO. 3

DOCUMENT NO.:	
DATE:	4/17/78

EXTERNAL ANS. REQ'D	YES NO
RESPONSIBILITY - PROJECT	
EXTERNAL RESP. - DUE DATE	
INTERNAL RESP. - BY	
FILE NO.	IP3 1.3.1.7
CENTRAL FILES:	

INDIAN POINT #3 PLANT	INFOR. COPY	ACT. COPY WITH ATTACH.
Bayne	/	
Becker		
Blake		
Burnett		
Byrne		
Cherico		
Clabby	/	
Davis	/	
Deeg		
Early	/	
Gidwani		
Gronberg		
Halama	/	
Huang		
Keshishian		
Klausmann	/	
Litchfield		
Lyon	/	
Mallon		
McNeill		
Schmieder		
Schomer	/	
Sunder Raj	/	
Wilverding	/	
Woods	/	
Teraszkiewicz		
Leopold		
Gramatges	/	
Khan	/	
Grochowski	/	

ACTION INSTRUCTIONS

NRR Supplement 1 to SER to ACRS concerning Full Power License (ECCS).

R. Gramatges (x6335)

CORRESPONDENCE CONTROL CW 4/20/78

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

APRIL 17 1978

NOTEL APR 19 1978 WILVERDING

Dr. Stephen Lawroski, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Lawroski:

Enclosed is Supplement 1 to the Safety Evaluation Report which was sent to you on April 6, 1978, regarding the proposed power increase to rated design power of the Indian Point 3 reactor. This supplement presents our evaluation of the submittal dated April 13, 1978, by the licensee, Power Authority State of New York, of the reanalysis of the ECCS performance with the corrections made to the recently discovered error in the volumetric heat generation of the zirc-water reaction.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosure:
Supplement 1 to SER

SUPPLEMENT 1 TO THE SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

UNITED STATES NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF THE POWER AUTHORITY OF THE STATE OF NEW YORK

REMOVAL OF LICENSE CONDITION LIMITING OPERATION

TO 91% OF RATED THERMAL POWER

FOR

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

APRIL 17 1978

I. Introduction

In Supplement No. 3 to the Safety Evaluation Report issued April 1976 and the SER dated April 6, 1978, for removal of the license condition limiting operation to 91% of rated thermal power, we concluded that the emergency core cooling performance for Indian Point Unit 3 conforms to the acceptance criteria of Section 50.46 of 10 CFR 50. These analyses performed in accordance with Appendix K to 10 CFR 50 identified the worst break as the double-ended cold leg break with a discharge coefficient (Moody multiplier) of 1.0.

On March 28, 1978, the staff met with the Westinghouse Electric Corporation to discuss a computational error discovered in the Westinghouse Evaluation Model for calculating loss-of-coolant accident in conformance with Appendix K to 10 CFR 50. The error involved a geometric error which resulted in only half of the volumetric heat generation due to metal-water reaction being used in the calculation of cladding temperature. The error was determined to be present in both the blowdown code (SATAN) and the fuel rod heatup code (LOCTA). We requested that the corrections be made to the evaluation model and that a reanalysis of the emergency core cooling system (ECCS) performance be performed for the Indian Point Unit 3.

II. Discussion

In a letter dated April 13, 1978, the licensee provided a reanalysis of the most limiting break using the previously approved Westinghouse Evaluation Model maintaining the same assumption as the previous analyses provided in the licensee's letters dated January 26 and April 20, 1977, but with the inclusion of the correction for the metal-water reaction heat release. Table 1 below summaries a comparison of pertinent input and results of the calculations provided in the licensee's letters of January 26 and April 13, 1978.

TABLE 1

INITIAL CORE CONDITIONS AND RESULTS FOR THE DOUBLE-ENDED COLD LEG BREAK ($C_D = 1.0$)

Initial Core Conditions	CONED Letter dated January 26, 1977	PASNY Letter dated April 13, 1978
Core Power (Mwt, 102% of Licensed Rating of)	3025	3025
Peak Linear Power (kw/ft, 102% of)	14.5	13.55
Heat Flux Hot Channel Factor ($F_Q(Z)$)	2.32	2.17
Radial Peaking Factor (F_{xy}, including uncertainties)	1.55	1.55
Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	1.55	1.55
Accumulator Water Volume (ft^3, each)	800	800

Results of Calculation	CONED Letter dated January 26, 1977	PASNY Letter dated April 13, 1978
Peak Clad Temp, °F	2125	2199
Peak Clad Location, ft	6.25	6.0
Local Zr/H_2O RXN (max, %)	7.59	11
Local Zr/H_2O Location, ft	6.0	6.0
Total Zr/H_2O RXN, %	<0.3	<0.3
Hot Rod Burst Time, sec	26.8	31.0
Hot Rod Burst Location, ft	6.0	6.0

III. Evaluation

We have reviewed the results of these analyses and also conclude that the worst break continues to be the double-ended cold leg break with a discharge coefficient (C_D) of 1.0. For this case, the recalculated peak clad temperature of the fuel rod was 2199 degrees Fahrenheit, which is below the acceptable limit of 2200 degrees Fahrenheit as specified in Section 50.46 of 10 CFR 50. In addition, the calculated maximum local metal-water reaction of 11 percent and a total core-wide metal-water reaction of less than 0.3 percent are well below the allowable limits of 17 percent and 1 percent, respectively. These analyses were performed with a total peaking factor (F_Q) of 2.17 at 102 percent of nuclear steam supply system power level of 3025 megawatts thermal.

We will require the licensee to either provide a plant specific constant axial offset control analysis of eighteen cases of load following which would ensure that the F_q limit of 2.17 would not be exceeded in normal operation of the plant or institute procedures for axial power distribution monitoring using manual procedures as indicated in Standard Technical Specifications 3/4 2.6 and ancillary specifications.

IV. Conclusion

Based on this review and previous supplements of the Safety Evaluation Report describing our review of the ECCS for Indian Point Unit 3, we conclude that the ECCS performance conforms to the acceptance criteria of Section 50.46 of 10 CFR 50 provided that the licensee completes the plant specific analysis which demonstrates that the F_q limit of 2.17 would not be exceeded in normal operation of the plant or institutes procedures for axial power distribution monitoring.

Westinghouse
Electric Corporation

Water Reactor
Divisions

Box 355
Pittsburgh Pennsylvania 15230

May 1, 1978

FP-CE-415

Mr. J. Clabby
Principal Fuels Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Clabby:

INDIAN POINT UNIT 3
Cycle 2 "18 Case" FAC Reanalysis Results

This letter formally transmits the Indian Point Unit 3 Cycle 2 "18 Case" FAC reanalysis results that were telecopied on Friday, April 21, 1978 to Dr. A. M. Khan.

We were pleased to learn that this information was very useful to PASNY during the ACRS meeting held on April 24, 1978.

Very truly yours,

R. N. Stanutz
Project Manager
NFD Projects

lrn

cc: J. Clabby 1L, 1A
G. A. Wilverding 1L, 1A
A. M. Khan 1L, 1A
M. L. Lee - Con Edison 1L, 1A

ATTACHMENT TO FP-CE-415

INDIAN POINT UNIT 3

Cycle 2 "18 Case" FAC Reanalysis

The "18 Case" FAC power distribution reanalysis was based on a Cycle 1 power rating limited to 91%. This analysis was requested by PASNY in order to minimize, or if possible, eliminate power rating restrictions due to a recent reduction in the LOCA limit (from 2.32 to 2.17). Sufficient advantages in power distribution parameters; e.g., Fxy(Z), Axial Offset, are available when a 91% power restriction is assumed for all of Cycle 1, so that Cycle 2 may be operated at 100% power with no restrictions except to limit the most positive axial flux difference pertinent to less than +10% at 100% power.

The results of this analysis are attached. Updated pages 5 and 10 of the Indian Point Unit 3 Cycle 2 RSE are given along with the new K(Z) function (Figure 2 of the RSE) appropriate to the 2.17 LOCA limit. The results of the "18 Case" power distribution analysis are given in the same format as Figure 4-1 of the Indian Point Unit 3 Cycle 2 Design Report. It should be emphasized that the validity of this analysis is predicted on restricting Indian Point Unit 3 Cycle 1 to power operation at or below 91% of full rated power (3025 MWt).

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability is evaluated considering the consequences of those incidents examined in the FSAR using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 2. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin during Cycle 2. The time dependent densification model was used for fuel temperature evaluations. The LOCA limit is met by maintaining $F_Q \times P$ at or below $2.17 \times K(Z)$ with K(Z) given in Figure 2. This limit is satisfied for the power control maneuvers allowed by the technical specifications, which assures that the final acceptance criteria (FAC) limits are met for a spectrum of small and large LOCAs.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR(1) and, fuel densification report(5) were examined. In most cases it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis. For those incidents which were reanalyzed, it was determined that the applicable design basis limits are not exceeded, and therefore, the conclusions presented in the FSAR and fuel densification report are still valid.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 2 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

4.0 TECHNICAL SPECIFICATIONS

This section contains the technical content of proposed changes to the Indian Point Unit 3 Technical Specifications. These changes are consistent with the plant operation necessary for the design and safety evaluation conclusions stated previously to remain valid.

4.1 SPECIFICATION 3.10.2 - POWER DISTRIBUTION LIMITS

Replace Figure 3.10-2.

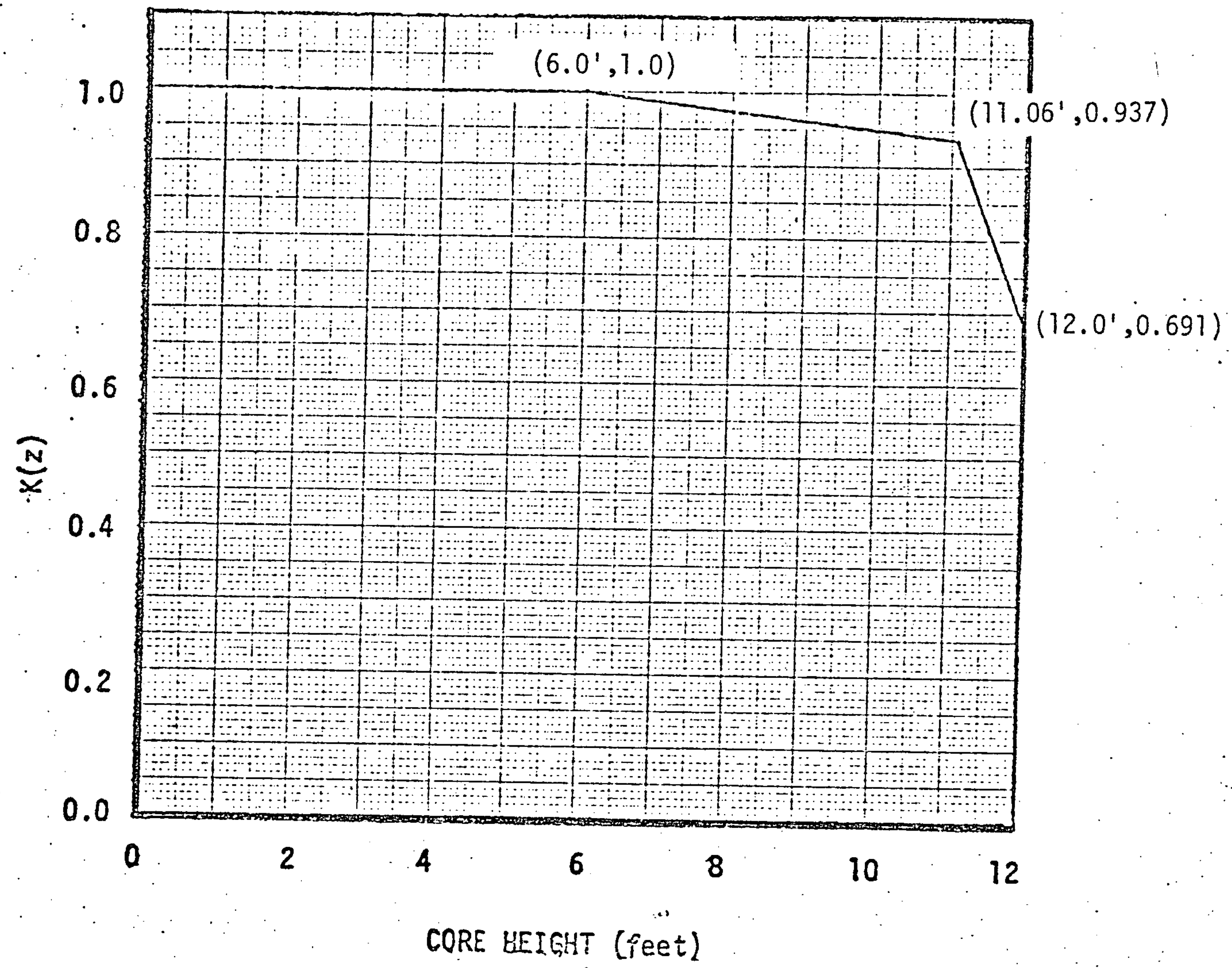
The increase in the K(Z) third line coordinate in Figure 2 from (12.0, 0.431) to (12.0, 0.691) assures that the Cycle 2 power control maneuvers allowed by the Technical Specifications will be satisfied. For this modified third line K(Z) segment, the small break LOCA was reanalyzed and was found to satisfy the FAC criteria.

Add to Section 3.10.2.4, "The indicated axial flux difference will be maintained less than + 10.0% at 100% power with the allowed axial flux difference increasing by 0.65% for each 1% reduction in power." This limit was used when verifying core peaking factor limits are met.

4.2 SPECIFICATION 3.10.4 - ROD INSERTION LIMITS

Revision: Replace Figure 3.10-4 with the attached Figure 3. This assures that core peaking factor limits are not exceeded during power control maneuvers allowed by the Technical Specifications.

F_Q^T NORMALIZED OPERATING
FUNCTION, K(z)



MAXIMUM [F_QT.P_{rel}] vs AXIAL CORE HEIGHT
DURING NORMAL CORE OPERATION