Vice President

Consolidated Edison Company of New York, Inc.
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Telephone (212) 460-3819

(3)

August 18, 1976

Re: Indian Point Unit No. 3
Docket No. 50-286
R.O.-76-3-28(A)

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
Region 1
U.S. Nuclear Regulatory Commission
King of Prussia, PA 19406

Dear Mr. O'Reilly:

In accordance with the requirements of the Technical Specifications to Facility Operating License DPR-64, the attached report of Reportable Occurrence R.O.-76-3-28(A) is submitted. This report fulfills the requirement for a written report within 14 days of a Reportable Occurrence and is in accordance with the format set forth in Regulatory Guide 1.16, Revision 4.

Three copies of this letter and the attachment are enclosed as required.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

mw
Enclosure

cc: Dr. Ernst Volgenau, Director (40 copies)
Office of Inspection and Enforcement

Mr. Robert W. Reid, Chief (3 copies)
Operating Reactors Branch No. 4
Office of Nuclear Reactor Regulation

Mr. William G. McDonald, Director (3 copies)
Office of Management Information and
Program Control

Mr. George T. Berry,
General Manager and Chief Engineer
Power Authority of the State of New York

LICENSEE EVENT REPORT

R.O.-76-3-28(A)

CONTROL BLOCK: [| | | | |] (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

Line	Field	Entry	Columns
01	LICENSEE NAME	N Y I P S 3	9–14
01	LICENSE NUMBER	0 0 – 0 0 0 0 0 – 0 0	15–25
01	LICENSE TYPE	4 1 1 1 1	26–30
01	EVENT TYPE	0 1	31–32
01 CON'T	CATEGORY		57–58
01 CON'T	REPORT TYPE	T	59
01 CON'T	REPORT SOURCE	L	60
01 CON'T	DOCKET NUMBER	0 5 0 – 0 2 8 6	61–68
01 CON'T	EVENT DATE	0 8 0 5 7 6	69–74
01 CON'T	REPORT DATE	0 8 1 8 7 6	75–80

EVENT DESCRIPTION

Line	Entry
02	
03	(SEE ATTACHED SHEETS)
04	
05	
06	

Line	Field	Entry	Columns
07	SYSTEM CODE	Z Z	9–10
07	CAUSE CODE	F	11
07	COMPONENT CODE	Z Z Z Z Z Z	12–17
07	PRIME COMPONENT SUPPLIER	Z	43
07	COMPONENT MANUFACTURER	Z 9 9 9	44–47
07	VIOLATION	Z	48

CAUSE DESCRIPTION

Line	Entry
08	
09	(SEE ATTACHED SHEETS)
10	

Line	Field	Entry	Columns
11	FACILITY STATUS	B	9
11	% POWER	0 5 5	10–12
11	OTHER STATUS	NA	13–44
11	METHOD OF DISCOVERY	D	45
11	DISCOVERY DESCRIPTION	VENDOR NOTIFICATION	46–80
12	FORM OF ACTIVITY RELEASED	Z	9
12	CONTENT OF RELEASE	Z	10
12	AMOUNT OF ACTIVITY	NA	11–44
12	LOCATION OF RELEASE	NA	45–80

PERSONNEL EXPOSURES

Line	NUMBER (9–11)	TYPE (12)	DESCRIPTION (13–80)
13	0 0 0	Z	NA

PERSONNEL INJURIES

Line	NUMBER (9–11)	DESCRIPTION (12–80)
14	0 0 0	NA

PROBABLE CONSEQUENCES

Line	Entry
15	NA

LOSS OR DAMAGE TO FACILITY

Line	TYPE (9)	DESCRIPTION (10–80)
16	Z	NA

PUBLICITY

Line	Entry
17	NA

ADDITIONAL FACTORS

Line	Entry
18	NA
19	

NAME: CHARLES W. JACKSON PHONE: 212-460-4681

EVENT DESCRIPTION:

On August 5, 1976, we were advised by Westinghouse Electric Corporation that they have identified a generic error in the non-conservative direction in their Emergency Core Cooling System (ECCS) analyses for compliance with the Final Acceptance Criteria (FAC) of Appendix K to 10CFR Part 50. Westinghouse has determined that the analytical effect of error correction results in higher calculated peak clad temperatures for most Westinghouse plants.

Westinghouse has also informed us that recent tests have indicated that with respect to plants which have identified a rod bowing problem, such as Indian Point Unit No. 3, DNB penalties due to rod bowing should be higher than previously reported. [R.O.-76-3-28(A)].

CAUSE DESCRIPTION:

In accordance with Section C.2.a(8) of Reg. Guide 1.16, Rev. 4, a reportable occurrence report is required for "errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses."

The ECCS analyses for most Westinghouse plants have been performed assuming that the primary coolant in the upper head region above the reactor core is at the cold leg temperature. On August 13, 1976, the NRC notified us that as a result of the generic error, the conservative assumption must be made that the temperature of the primary coolant in the upper head region of the reactor is the hot leg temperature. Westinghouse has determined that this results in a 40^{o}F increase in the calculated maximum peak clad temperature for Indian Point Unit No. 3.

The ECCS analysis performed for Indian Point Unit No. 3 (see Table 14C-1 of Indian Point Unit No. 3 FSAR) yielded a calculated maximum peak clad temperature of 2168^{o}F. Applying the penalty described above (no steam generator tubes are plugged in Indian Point Unit No. 3), the calculated maximum peak clad temperature becomes 2208^{o}F.

Westinghouse has evaluated the effect of a change in F_Q on peak clad temperature. They have determined that a decrement of 0.01 in F_Q results in a 10^{o}F reduction in peak clad temperature. Thus, for a new F_Q limit of 2.312 (a negligible change in F_Q of 0.008) the peak clad temperature for Indian Point Unit No. 3 would be reduced to 2200^{o}F.

This new F_Q limit is sufficiently large to require no additional analysis, or surveillance, for the remainder of the present cycle. The present burnup (1628 MWD/MTU as of August 10, 1976) is sufficiently large to ensure the maximum F_Q will be below the 2.312 value, since F_Q decreases with burnup. To quantify the effect for

Indian Point Unit No. 3 the measured F_z and F_Q^N values, versus burnup, have been plotted (see Figures 1 and 2). This data (representing steady operation at power levels between 5 and 90% of full power, Bank D at 189 to 215 steps, and essentially equilibrium xenon conditions) exhibits a negative slope (decreasing F_z and F_Q^N) and the estimated decrease due to the present cycle burnup (see Figures 1 and/or 2) is 0.08 in F_Q. This is an order of magnitude larger than the required decrease in F_Q (i.e. 0.008). A similar trend is expected to apply for allowable operating configurations during the remainder of the cycle.

Scoping calculations performed by Westinghouse Electric Corporation indicate that the closest approach of calculated peaking factors to the 2.32 / power x K(Z) total peaking factor boundary occurs about the core midplane.

In addition, these scoping calculations show that the calculated peaking factors associated with these limiting elevations are all produced early in cycle life. Reevaluating the calculated peaking factors at these limiting elevations at a core burnup of 1,000 MWD/MTU, thereby accounting for burnup effects on the axial power shape, indicates peaking factor decreases of about 5% and 2% from scoping calculations, for base load and load follow (transient Xenon) conditions, respectively.

A 5% decrease in axial peaking factor from 0 to 1,000 MWD/MTU for base load operation is consistent with plant data presented in Figures 1 and 2.

Results for Indian Point Unit No. 2 (see Figure 3, previously filed with the NRC on January 12, 1976 via letter from LeBoeuf, Lamb, Leiby & MacRae to Mr. Ben C. Rusche) exhibit a similar trend; however, as expected, the slope is less negative than shown on Figures 1 and 2 (by a factor of 2 to 3) due to the greater burnup (range of data is about 4200 to 14,000 MWD/MTU).

Additionally, as part of the augmented test program for Indian Point Unit No. 3, it is expected that further data justifying the conservatism of the 2.312 F_Q value will be obtained. A detailed report containing the results of these tests will be submitted to the NRC at the conclusion of the augmented test program.

Based on these considerations no additional action, either analytical or administrative, is required, for the remainder of the first cycle, to ensure that the revised F_Q limit of 2.312 (as opposed to 2.32) is not exceeded.

Fuel rod bowing has been under generic review by the NRC Staff and has focused on the information supplied in the Westinghouse Electric Corp. topical reports WCAP-8691 (Proprietary) and WCAP-8692 (Non-proprietary) entitled, "Fuel Rod Bowing" submitted December, 1975.

These reports describe an empirical model for conservatively projecting the magnitude and frequency of rod bow with burnup and identifies generic design margins available to offset corresponding DNB and power spike effects.

Westinghouse has informed Con Edison that the NRC has reached and issued an interim licensing position (D.B. Vassallo to C. Eicheldinger letter dated May 11, 1976) concluding that the generic design margins are adequate to offset bow effects for 15 X 15 fuel design with linear core power densities exceeding that corresponding to 100% rated power operation of the Indian Point Unit No. 3 plant. Thus, no additional penalty or information is required to account for the power spike effect on F_Q (for LOCA), and the extent of fuel rod bow, as a function of burnup.

However, recent data on the effect of fuel rod bow on DNB, results in larger DNB penalties. The NRC requires that $F_{\Delta H}$ be reduced to account for loss of DNB margins, or that existing margins be used to demonstrate that the $F_{\Delta H}$ penalty is not required. These margins were specifically identified to be above design reactor coolant flow and/or decreased reactor core inlet temperature (and an associated overtemperature trip setpoint modification).

Indian Point Unit No. 3 has considerably greater reactor coolant flow than design (about 9% greater) and a lower reactor coolant inlet temperature (about 2 to 3°F lower than design); however, for first cycle operation, administrative action will be taken to apply the $F_{\Delta H}$ penalty, with no benefit taken for the demonstrated margins present. Thus, the maximum $F_{\Delta H}$, for first cycle operation, will include a penalty directly proportional to the core average burnup, which varies from 0% at 0 MWD/MTU to 4% at 15,000 MWD/MTU.

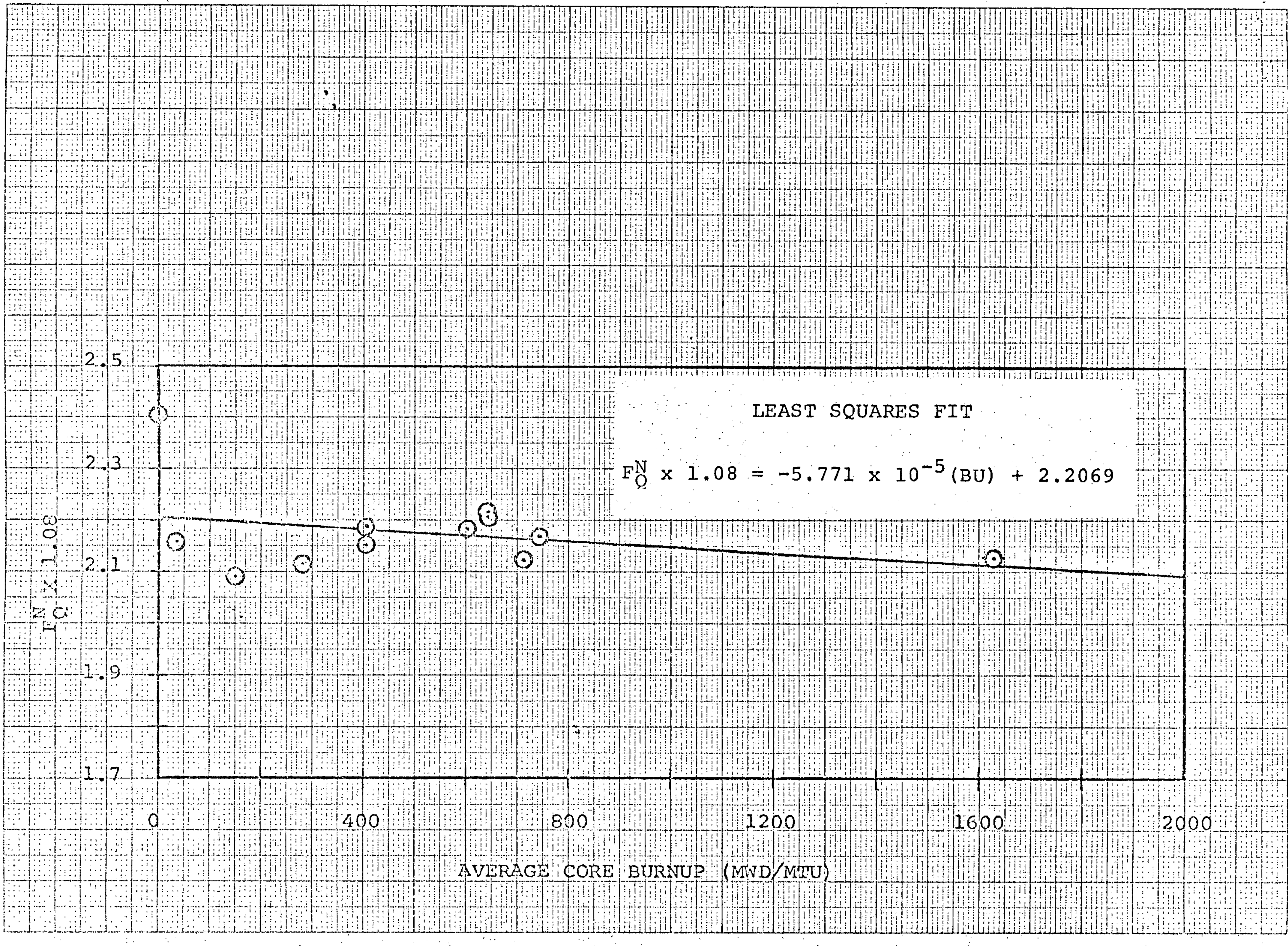

Figure 1. Indian Point Unit No. 3, F_Q^N versus burnup.

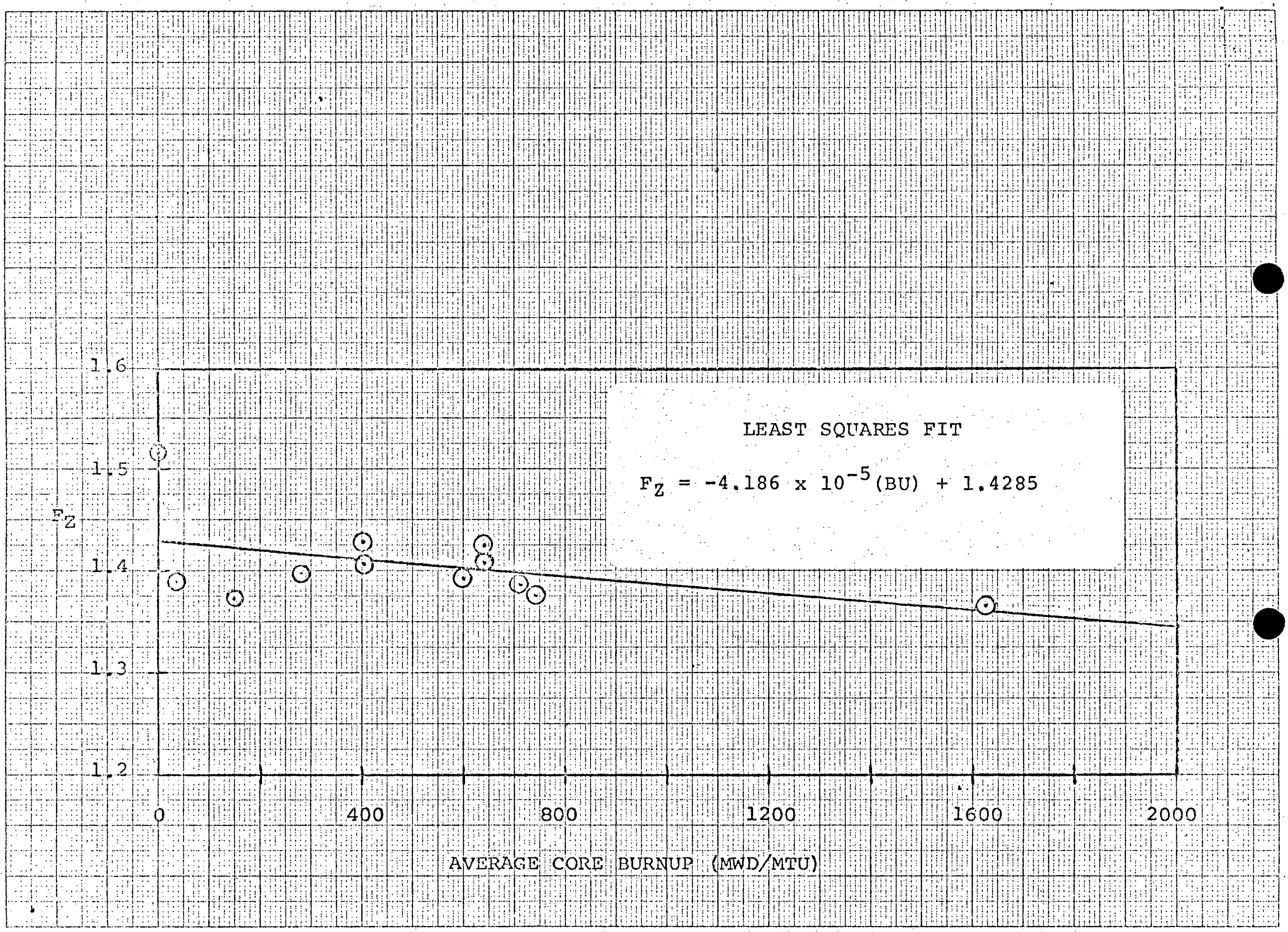

Figure 2. Indian Point Unit No. 3, F_Z versus burnup.

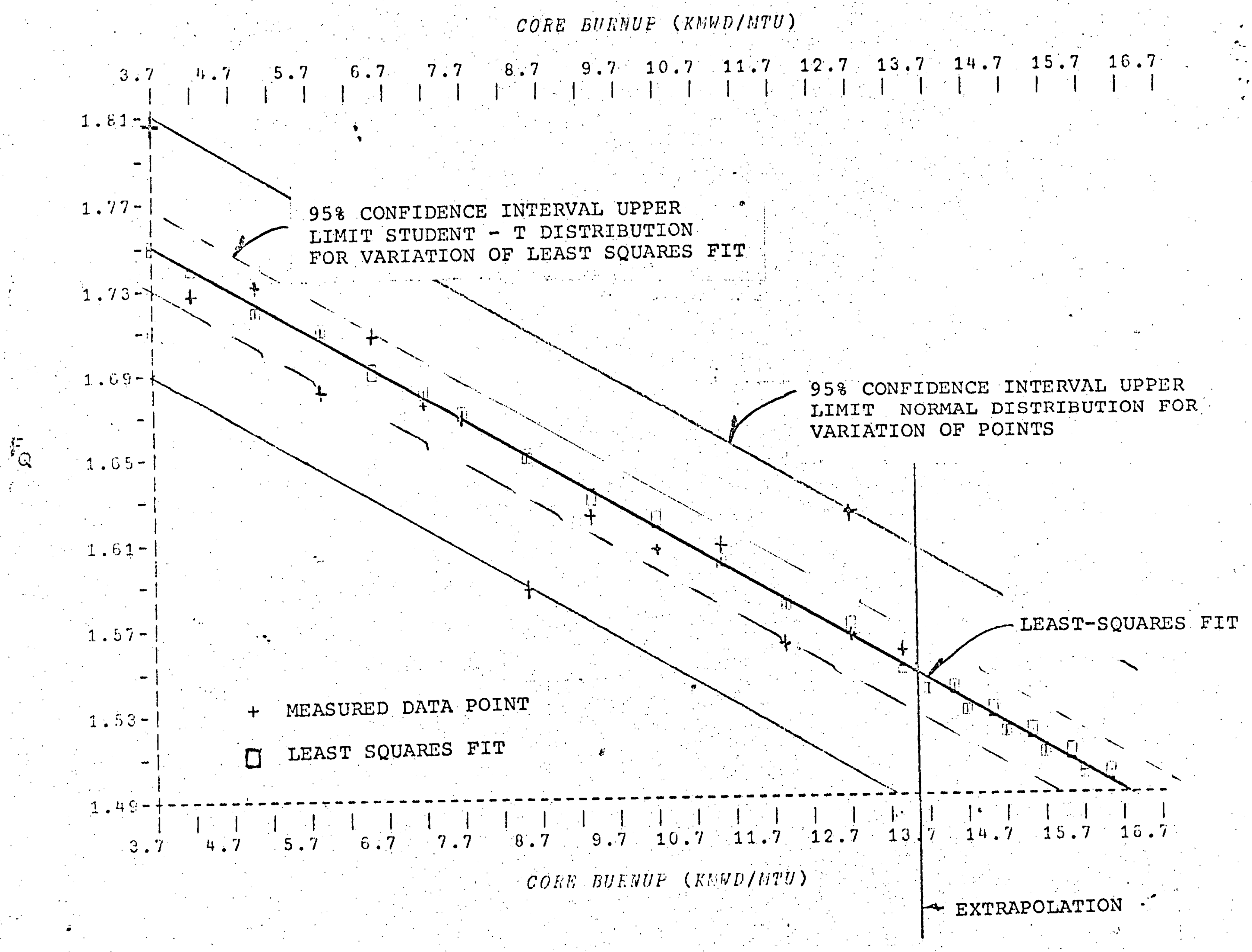

Figure 3. Indian Point Unit No. 2, F_Q^N versus burnup.