ID/TS2-B

QUAD-CITIES NUCLEAR POWER STATION

UNIT 2 CYCLE 7

STARTUP TEST RESULTS

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1. Control Rod Scram Timing

Purpose

The purpose of this test is to demonstrate the scram capability of all of the operable control rods in compliance with Technical Specifications 4.3.C.1 and 4.3.C.2.

Criteria

A. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods during reactor power operation shall be no greater than:

% INSERTED FROM FULLY WITHDRAWN	AVG. SCRAM INSERTION TIMES (sec)
5	0.375
20	0.900
50	2.000
90	3.500

The average of the scram insertion times for the three fastest control rods of all groups of four rods in a two by two array shall be no greater than:

% INSERTED FROM FULLY WITHDRAWN	AVG. SCRAM INSERTION TIMES (sec)
5	0.398
20	0.954
50	2.120
90	3.800

If these times cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shutdown immediately upon determination that average scram time is deficient.

B. The maximum insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds. If this requirement cannot be met, the deficient control rods shall be considered inoperable, fully inserted into the core, and electrically disarmed.

Results and Discussion

All 177 control rods were scram tested. The results are presented in Table 1. The maximum 90% insertion time was 3.08 seconds for control rod H-7 (30-27). Both criteria A and B were met.

Table 1.

Control Rod Scram Results

NUMBER OF RODS	REACTOR CONDITIONS	AVERAGE TIMES FOR % INSERTED, SEC 5%	20%	50%	90%
177	Cold	0.26	0.49	0.97	1.69
177	Hot	0.31	0.69	1.47	2.59

2. Shutdown Margin Demonstration and Control Rod Functional Checks

Purpose

The purpose of this test is to demonstrate for this core loading in the most reactive condition during the operating cycle, that the reactor is subcritical with the strongest control rod full out and all other rods fully inserted.

Criteria

If a shutdown margin of 1.2347% ΔK (=0.25% + R + B_4C settling penalty) cannot be demonstrated with the strongest control rod fully withdrawn, the core loading must be altered to achieve this margin. The core reactivity has been calculated to be at a maximum 2500 MWD/T into the cycle and R is given as 0.934%ΔK. The control rod B_4C settling penalty for Unit Two is 0.05%ΔK.

Results and Discussion

On December 21, 1983, control rod D-8 (the rod which was calculated by General Electric to be of the highest worth) was fully withdrawn to demonstrate that the reactor would remain subcritical with the strongest rod full out. This maneuver was performed to allow cold control rod testing prior to the shutdown margin demonstration.

Control Rod functional subcritical checks were performed as part of the cold scram timing and control rod friction testing. No unexpected reactivity insertions were observed when any of the 177 control rods were withdrawn.

General Electric provided rod worth information for the two strongest diagonally adjacent rods E-7 and E-9 with rod D-8 full out. This method provided an adequate reactivity insertion to demonstrate the desired shutdown margin. On February 17, 1984, a diagonally adjacent shutdown margin demonstration was successfully performed. Using the G.E. supplied rod worth for D-8 (the strongest rod) and diagonally adjacent rods E-7 and E-9, it was determined that with D-8 and E-7 at position 48, and E-9 at position 18, a moderator temperature of 177°F, and the reactor critical on a 300 second period, a shutdown margin of 1.294% Δ K was demonstrated. The G.E. calculated shutdown margin with D-8 withdrawn and 68°F reactor water temperature was 2.287% Δ K at the beginning of cycle 7.

At approximately 2500 MWD/T into cycle 7 a minimum calculated shutdown margin of 1.353% Δ K will occur with M-8 fully withdrawn. Note that the minimum shutdown margin shifts from rod D-8 at beginning of cycle to rod M-8 at 2500 MWD/T.

The rod worth curves supplied by General Electric for D-8, E-7, and E-9 indicated that the reactor would be subcritical with D-8 and E-7 at 48 and E-9 at 18. As this was not the case, G.E. and Commonwealth Edison's Nuclear Fuel Services Department were consulted by the Station to determine a cause for this discrepancy. It is the position of General Electric that the discrepancy was caused by computer code modeling uncertainties concerning the four fuel bundles surrounding the highest worth rod. These four are all once-burned, higher reactivity fuel bundles that are to be a part of the end of cycle barrier fuel ramp demonstration. Due to this concentration of high reactivity fuel with a similar exposure history in a single control cell, the self correcting aspects of the normal mixed cell loading are not available to dampen exposure uncertainties.

The position presented by General Electric is supported by several observations: 1) a review of the core verification video tapes revealed no misloaded or misoriented bundles in the vicinity of the demonstration, 2) an examination of control rod exposures in the area of the demonstration divulged no excessive exposure relative to the current boron depletion limit, 3) a second local shutdown margin demonstration performed in a symmetric and identically loaded core location displayed results very similar to the first demonstration, and 4) the critical rod pattern predicted by the same General Electric computer codes for the dispersed, whole-core critical was only 0.0008ΔK different from the actual critical rod pattern. Therefore, the physical and experimental evidence supports the conclusion that the predicted to actual local critical discrepancy was not the result of a local or core-wide anomaly, but rather was the result of local computer model uncertainties.

3. Initial Critical Prediction

Purpose

The purpose of this test is to demonstrate General Electric's ability to calculate control rod worths and shutdown margin by predicting the insequence critical.

Criteria

General Electric's prediction for the critical rod pattern must agree within 1% Δ K to actual rod pattern. A discrepancy greater than 1% Δ K in the non-conservative direction will be cause for an On-Site Review and investigation by Nuclear Fuel Services.

Results and Discussion

On February 18, 1984, at 0303 hours the reactor was brought critical with a reactor water temperature at the time of criticality of 167°F. The ΔK difference between the expected critical rod pattern at 68°F and the actual critical rod pattern at 150°F was 0.0012 from rod worth tables supplied by General Electric. The temperature effect was -0.0014 ΔK from General Electric-supplied corrections. The excess reactivity yielding the 92 second positive period was 0.0006 ΔK. These reactivities sum to give 0.0008 ΔK difference (0.08% ΔK) between the expected critical rod pattern and the actual rod pattern. This is within the 1% ΔK required in the criteria of this test, and General Electric's ability to predict control rod worths is, therefore, successfully demonstrated.

4. Core Power Distribution Symmetry Analysis

Purpose

The purpose of this test was to determine the magnitude of indicated core power distribution asymmetries using data (TIP traces and OD-1) collected in conjunction with the P-1 update.

Criteria

A. The total TIP uncertainty (including random noise and geometric uncertainties obtained by averaging the uncertainties for all data sets) must be less than 9%.

B. The gross check of TIP signal symmetry should yield a maximum deviation between symmetrically located pairs of less than 25%.

Results and Discussion

Core power symmetry calculations were performed based upon computer program OD-1 data runs on March 13, 1984, at 99.8% power, and March 27, 1984, at 99.7% power. The average total TIP uncertainty from the two TIP sets was 6.020%. The random noise uncertainty was 2.867%. This yields a geometrical uncertainty of 5.292%. The total TIP uncertainty was well within the 9% limit.

Table 2 lists the symmetrical TIP pairs and their respective deviations. Figure 1 shows the core location of the TIP pairs and the average TIP readings. The maximum deviation between symmetrical TIP pairs was 15.29% for pair 18-39. Thus, the second criterion, mentioned above, was also met.

The method used to obtain the uncertainties consisted of calculating the average of the nodal ratio of TIP pairs by:

$$\bar{R} = \frac{1}{18n}\left[\sum_{j=1}^{n}\sum_{i=5}^{22} Rij\right]$$

where Rij is the ratio for the ith node of TIP pair j, there being n such pairs, where n=18.

Next the standard deviation of the ratios is calculated by:

$$\sigma_{\bar{R}} = \left[\frac{\sum_{j=1}^{n}\sum_{i=5}^{22} (Rij - \bar{R})^2}{(18n - 1)}\right]^{\frac{1}{2}}$$

σ_R is multiplied by 100 to express σ_R as a percentage of the ideal value of $_R$ of 1.0.

$$\%\ \sigma_R = \sigma_R \times 100$$

The total TIP uncertainty is calculated by dividing % σ_R by $\sqrt{2}$ in order to account for data being taken at 3 inch intervals and analyzed on a 6 inch nodal basis.

In order to calculate random noise uncertainty the average reading at each node for nodes 5 through 22 is calculated by:

$$\overline{\text{BASE}}\ (K) = \frac{1}{NT\ .\ MT}\left[\sum_{M=1}^{MT}\ \sum_{N=1}^{NT}\ \text{BASE}\ (N,\ M,\ K)\right]$$

where NT = number of runs per machine = 4
MT = number of machines = 5
$\underline{\text{BASE}}$ (K) = average reading at nodal level K,
K = 5 through 22

The random noise is derived from the average of the nodal variances by:

$$\%\sigma\ \text{noise} = \left[\frac{\sum_{K=5}^{22}\ \sum_{M=1}^{MT}\ \sum_{N=1}^{NT}\left[\frac{\text{BASE}\ (N,\ M,\ K) - \overline{\text{BASE}}\ (K)}{\text{BASE}\ (K)}\right]^2}{18\ (NT \times MT - 1)}\right]^{\frac{1}{2}} \times 100$$

Finally the TIP geometric uncertainty can be calculated by:

$$\%\ \sigma\ \text{geometric} = (\%\ \sigma\ \text{total}^2 - \%\ \sigma\ \text{noise}^2)^{\frac{1}{2}}$$

Table 2

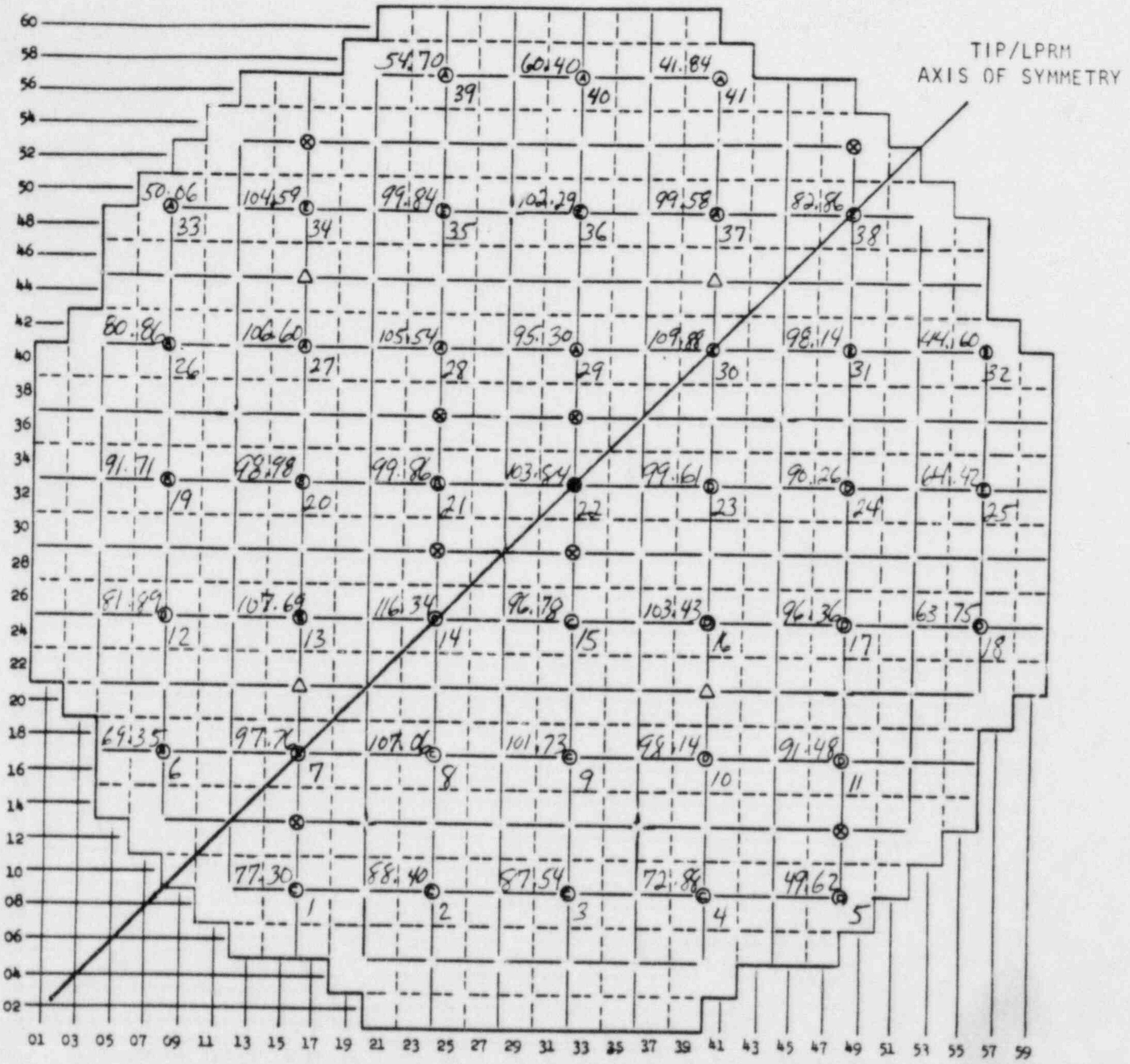
CORE SYMMETRY
Based on OD-1's From
03-13-84 (99.8% power), and 03-27-84 (99.7% power)

SYMMETRICAL TIP PAIR NUMBERS a	b	$T = T_a - T_b$ ABSOLUTE DIFFERENCE	$\% = 100 \times T / \frac{T_a + T_b}{2}$ % DEVIATION
1	6	7.958	10.868
2	12	6.505	7.646
3	19	4.173	4.634
4	26	7.979	10.362
5	33	1.200	2.390
8	13	0.624	0.580
9	20	2.744	2.742
10	27	8.457	8.255
11	34	13.114	13.384
15	21	3.082	3.136
16	28	2.098	1.981
17	35	3.470	3.560
18	39	9.054	15.289
23	29	4.307	4.438
24	36	12.023	12.480
25	40	1.160	1.750
31	37	1.787	1.779
32	41	2.762	6.386

$$T_i = \sum_{i=5}^{22} T_i(K) \quad /18$$

Average Deviation=
5.139%

FIGURE 1



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

△ SRM Locations

AVG. BASE

STRING NO.

UNIT TWO POWER SYMMETRY
AVERAGE BASE READINGS
(nodes 5 through 22)
from OD-1's on 03-13-84,
and 03-27-84

BW/BY

MAY 9, 1984

To:

C Reed/NA Kershaw
LO DelGeorge
NE Wandke
JS Abel
HE Bliss
WV Burkamper
TC Cilhar/GE Peterson
R Cosaro
LE Davis
JD Deress/JT Westermeier
EE Fitzpatrick
DP Galle
KL Graesser (NC ONLY)
JF Gudac
KJ Hansing
JH Hughes
PE Hull
RE Jortberg
NJ Kalivianakis (NC ONLY)
AW Kleinrath
DE Lindvall
TJ Maiman/BR Shelton
TE Quaka
RE Querio

VI Schlosser/RP Tuetken
DJ Scott
JS Scott
DL Shamblin
WJ Shewski
G Sorenson
BB Stephenson
HP Studtmann
CJ Tomashek
GP Wagner
MJ Wallace
EF Wilmere/GL Tanner
WR Bird-Consumers Power
\- (RIII ONLY)
MA Bowidowicz-S&L
J Gallo-IL&B Washington
J Gayley IL&B
WE Kortier-W
ZE Pate (50.55e ONLY)
G Wright-State of Illinois
\- (NRC/CECO LTRS ONLY)
NL Distribution
T. TRAMM

In the judgement of the Nuclear Licensing Administrator the attached document contains information that may be useful to you or your organization. No specific action or response by Commonwealth Edison is required at this time.

IDENTIFICATION OF ATTACHED DOCUMENT:

Byron - ASLAB DECISION dated May 7, 1984 reopening the hearings.

NOTE:

For your information.

ED Swartz
NL-84-0560

FILE: Byron - ASLAB

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

May 8, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Quad Cities Station Unit 2
Summary Startup Test
Report - Cycle 7
NRC Docket No. 50-265

Dear Mr. Denton:

Enclosed for your information and use is the Quad Cities Station Unit 2, Cycle 7 Startup Test Report Summary. This report is submitted in accordance with previous requests from the NRC Staff and our Technical Specifications.

Please address any questions concerning this matter to this office.

One (1) signed original and forty (40) copies of this letter and enclosure are provided for your use.

Very truly yours,

B Rybak

B. Rybak
Nuclear Licensing Administrator

lm

cc: NRC Resident Inspector - Quad Cities

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