Commonwealth Edison
72 West Adams Street, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

June 12, 1986

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

Subject: Dresden Station Unit 3
Additional Information Regarding
Cycle 10 Reload
NRC Docket No. 50-249

References (a): Letter from J. A. Zwolinski to D. L. Farrar dated June 3, 1986.

(b): Letter from J. R. Wojnarowski to H. R. Denton dated April 18, 1986.

Dear Mr. Denton:

The reference (a) memo requested additional information regarding the Dresden Unit 3 Cycle 10 reload. The questions transmitted with reference (a) were discussed in a number of recent telecons. This transmittal provides our response.

Attachment 1 responds to questions 1, 2 and 3 of the reference (a) letter and includes a revised Technical Specification Figure 3.5-1A indicating linear heat generation rate limits. This figure supercedes the one transmitted with reference (b) and should be incorporated in our proposed Cycle 10 Technical Specifications. This attachment also includes a clarification of the proposed MCPR operating limits for Cycle 10 in response to questions raised during telecons with your staff.

Attachment B responds to question 4 from reference (a) regarding Exxon's MCPR Safety Limit determination.

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If you have any additional questions regarding this matter, please contact this office.

Three (3) signed originals and thirty-seven (37) copies of this letter and the attachments are provided for your use.

Very truly yours,

J. R. Wojnarowski
Nuclear Licensing Administrator

lm

Attachment

cc: R. Gilbert - NRR
Resident Inspector - Dresden
M. C. Parker - IDNS

SUBSCRIBED AND SWORN to
before me this 12th day
of June, 1986

Notary Public

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ATTACHMENT 1

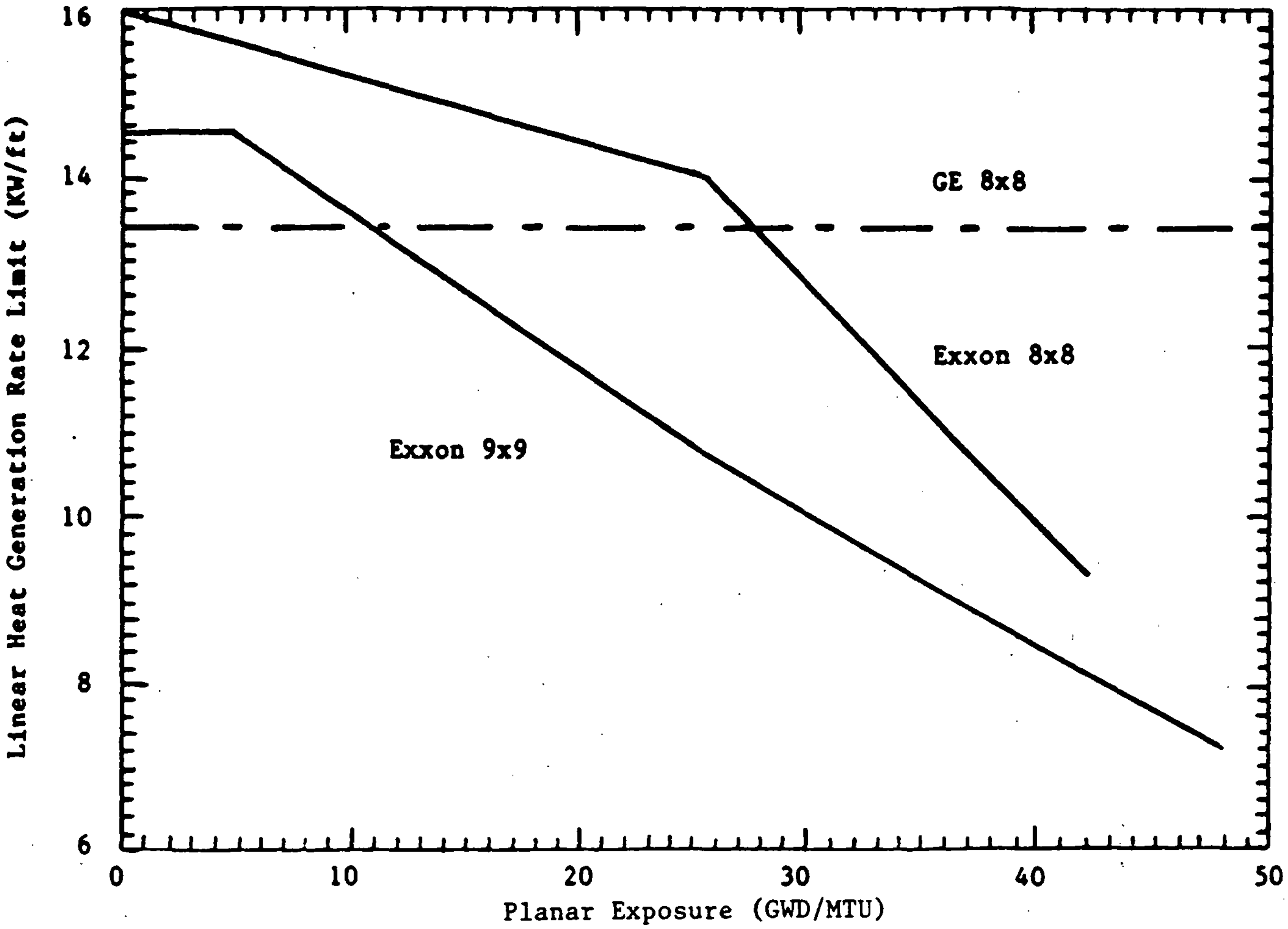
RESPONSE TO NRC QUESTIONS ON D3C10 RELOAD

QUESTIONS 1, 2 AND 3

The NRC has previously requested that Dresden incorporate Linear Heat Generation Rate (LHGR) limits into its Technical Specification. CECo submitted proposed limits in Reference (b). When the NRC reviewed the LHGR submittal, several questions arose concerning which of several Exxon documents on fuel mechanical design were the proper source of the data.

The most recent ENC generic mechanical design document which is currently undergoing NRC review is XN-NF-85-67, Revision 1. NRC approval of this topical report is expected within the next few weeks (before the D3C10 startup). During recent conference calls between CECo, Exxon and the NRC, it was agreed that CECo would submit LHGR limit curves for both ENC 8x8 and ENC 9x9 fuel based on Figures 3.1 and 3.2, respectively of XN-NF-85-67, Revision 1 for D3C10. These curves express fuel burnup in terms of planar exposure as monitored by the site software (POWERPLEX). The attached Techncial Specification Figure 3.5-1A reflects the curves, also in terms of planar average exposure.

During the previously mentioned conference calls, the staff requested clarification of how the proposed MCPR operating limits of 1.31 for ENC/GE 8x8 fuel and 1.35 for ENC 9x9 fuel were determined. Table 2.2 of XN-NF-85-62 (D3C10 Transient Analysis Report) indicates MCPR operating limits for GE 8x8, ENC 8x8 and ENC 9x9 fuel which were determined by adding the transient CPRs from Table 2.1 to the 1.05 MCPR Safety Limit. For simplicity, Commonwealth Edison applied the more conservative ENC 8x8 limit to the GE fuel resulting in MCPR operating limits of 1.29 for GE/ENC 8x8 fuel and 1.33 for ENC 9x9 fuel. As indicated in Section II.A of our previously submitted safety evaluation, we increased these valves by 0.02 to provide additional conservatism in the Technical Specification operating limits to facilitate future 50.59 reload reviews. This yields our proposed operating limits of 1.31 for GE/ENC 8x8 fuel and 1.35 for ENC 9x9 fuel.



Exxon 8x8 Fuel

Exposure	LHGR
0.00	16.00
25.40	14.10
42.00	9.30

Exxon 9x9 Fuel

Exposure	LHGR
0.00	14.50
5.00	14.50
25.20	10.80
48.00	7.20

Figure 3.5-1A

LINEAR HEAT GENERATION RATE VS.
NODAL EXPOSURE

ATTACHMENT 2

SATISFACTION OF SER CONDITIONS

DRESDEN UNIT 3 CYCLE 10 MCPR SAFETY LIMIT ANALYSIS

In the Safety Evaluation prepared by the Core Performance Branch covering Revision 1 to XN-NF-524(P), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," the NRC Staff identified four conditions to be met during the application of the subject methodology under the generic approval granted by the SER. The steps taken during the analysis to assure compliance with these conditions are described below.

CONDITION 1: Each plant specific application must contain the data used to generate the uncertainties employed in the methodology.

The uncertainties used in the Dresden Unit 3 MCPR safety limit calculations are the same as the uncertainties which have been used in previous Dresden analyses.

Plant measurement uncertainties which are not fuel-dependent were taken from approved NSSS supplier generic documents applicable to Dresden. As identified in XN-NF-524(A), Revision 1, specific uncertainty values used in the analysis were a feedwater flow rate uncertainty of 1.76%, a feedwater temperature uncertainty of 0.76%, a core pressure uncertainty of 0.5%, a total core flow rate uncertainty of 2.5%. The generic core inlet temperature uncertainty of 2.0% was conservatively replaced with an uncertainty of 2.4% on the core inlet enthalpy. These approved values were used as one-sigma uncertainties consistent with NEDO-24011. The nominal values, uncertainties, and statistical treatment of these measured plant parameters are summarized in Table 1.

The uncertainties associated with the XN-3 Critical Power Correlation are based on data contained in XN-NF-512(A), Revision 1, and XN-NF-734(A). The safety limit analysis was based on a one-sigma uncertainty value of 4.11% for the XN-3 correlation, consistent with the source documents noted above and with XN-NF-80-19(A), Volume 4, Revision 1, which provides a generic description of the overall reload analysis. The correlation statistics were developed from the XN-3 data base, which includes test geometries which encompass the Dresden 8x8 and 9x9 fuel designs. XN-NF-734(A) was issued explicitly to validate the XN-3 statistics for application to 9x9 fuel.

Power distribution measurement uncertainties are based on data contained in XN-NF-80-19(A), Volume 1. The safety limit calculation was based on one-sigma uncertainties of 5.28% on radial peaking factor and 2.46% on local peaking factor consistent with the reference report. These uncertainties were developed based on analytical predictions of measured data for BWR fuel. The same methods used for the analytical predictions were used for the nuclear design analyses for Dresden; hence the generic uncertainty values are applicable to Dresden.

The correlation and power distribution measurement uncertainties and their statistical treatment for the Dresden analysis are summarized in Table 2.

CONDITION 2: All plant parameters that are not statistically convoluted must be placed at their limiting value.

In the performance of plant transient analyses, ENC uses design values for major process parameters for consistence with the FSAR analyses which are superseded by the ENC transient analyses. Design values are established by the plant designer as conservative predictions of the boundaries of the plant operating envelope, and may not be accurate predictions of actual plant operation. These values are used to assure a conservative calculation of the transient effects. Nominal values are best-estimate predictions of plant operating conditions. The use of nominal conditions is appropriate for the statistically treated parameters in the Monte Carlo analysis.

Input to the Monte Carlo calculation consists of three major classifications of data: heat balance information, power distribution information, and fuel geometric information.

Heat balance information consists of feedwater temperature and flow rate, core pressure and total flow rate, and core inlet enthalpy. All of these variables are considered statistically in the Monte Carlo analysis.

Power distribution information is taken from the fuel management analysis and consists of radial, axial, and local peaking factors. Radial and local peaking factors are considered statistically in the Monte Carlo analysis. For power distributions characterized by bottom-peaked core average axial power shapes, a limiting center-peaked axial distribution is used.

Fuel geometric information consists of fuel dimensions and hydraulic demand curves. Small variations in fuel dimensions within manufacturing tolerances are considered in the ENC pressure drop methodology and contribute to the flow distribution uncertainty. Hydraulic demand curves are used to determine fuel assembly flow rates as a function of bundle power; individual assembly flow rates are treated statistically in the Monte Carlo analysis.

CONDITION 3: Each application should demonstrate that the uncertainties in plant parameters are treated with at least a 95% probability at a 95% confidence level in accordance with Acceptance Criterion 1.0 of Standard Review Plan Section 4.4.

The magnitude and nature of the uncertainties used in the Monte Carlo analysis have been established generically during the Staff review of ENC topical report XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors." A detailed review of the XN-3 correlation statistics was included in the review of ENC topical report XN-NF-512(A), Revision 1, "The XN-3 Correlation." A detailed review of power distribution measurement uncertainties was included in the review of ENC topical report XN-NF-80-19(A), Volume 1, "Neutronics Methods for Design and Analysis." The conclusion that these uncertainties may be conservatively treated as normally distributed was addressed during the generic review.

Uncertainties in the measurement of plant parameters were taken from the NSSS supplier's generic reload submittal. Based on the Staff's approval of these uncertainties for use in the MCPR safety limit calculation, the ENC analyses used the published values as 95% confidence statistics. Process measurement uncertainties are generally characterized by a normal distribution; therefore, a normal distribution was used in the ENC analysis.

The Monte Carlo analysis was performed to demonstrate that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition at a confidence level of 95%. This conclusion conservatively assures that the boiling transition limitation will be protected during anticipated operational occurrences in which the MCPR safety limit is protected. The referenced Standard Review Plan section identifies this method as an acceptable approach to the 95/95 treatment of uncertainties.

CONDITION 4: Each application must present a goodness-of-fit analysis for the fitting of the Pearson curve in order to assure that the number of Monte Carlo trials used in establishing the safety limit MCPR are sufficient.

In the original ENC MCPR safety limit methodology, the first four statistical moments of the Monte Carlo output were used to define an output frequency distribution through fitting of Pearson functions. This approach was taken to minimize the number of trials necessary in the Monte Carlo analysis. Revision 1 to XN-NF-524 abandoned this approach in favor of a distribution-independent method of assigning tolerance limits. The new approach required a larger number of Monte Carlo trials, but the end result was a conclusion which was independent of the Pearson functions.

Since the statistical analysis involved no fitting of standard functions to the Monte Carlo output, no goodness-of-fit analysis was provided. In the case of the Dresden analysis, 500 Monte Carlo trials were performed. In the non-parametric tables, an expected value may be established at a confidence level of 95% with as few as 50 trials.

TABLE 1

PLANT MEASUREMENT UNCERTAINTIES

PARAMETER	UNITS	NOMINAL VALUE	UNCERTAINTY ~~PCT~~ PERCENT NOMINAL	STATISTICAL TREATMENT
Feedwater flowrate	Mlbm/hr	12.56*	1.76	Convoluted
Feedwater temperature	deg F	345	0.76	Convoluted
Core pressure	psia	1015	0.50	Convoluted
Total core flow	Mlbm/hr	98.0	2.50	Convoluted
Core inlet temperature			2.00	Replaced by core inlet enthalpy
Core inlet enthalpy	BTU/lbm	521.9	2.40	Convoluted
Core power	MW	3222*		Allowed to vary with heat balance

*Feedwater flowrate and core power were increased above design values to attain desired core MCPR for safety limit evaluation, consistent with XN-NF-524(A), Revision 1.

Source: XN-NF-524(A), Revision 1

TABLE 2

FUEL-RELATED UNCERTAINTIES

PARAMETER	SOURCE DOCUMENT	UNCERTAINTY ~~PCT~~ PERCENT NOMINAL	STATISTICAL TREATMENT
XN-3 Correlation	XN-NF-512(A) XN-NF-734(A)	4.11	Convoluted
Radial Peaking Factor	XN-NF-80-19(A) Volume 1	5.28	Convoluted
Local Peaking Factor	XN-NF-80-19(A) Volume 1	2.46	Convoluted
Axial Peaking Factor	XN-NF-80-19(A) Volume 1	2.99	Limiting Value
Assembly Flowrate	XN-NF-79-59(A)	2.80	Convoluted

Commonwealth Edison
72 West Adams Street, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

May 22, 1989

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Nuclear Power Station Unit 3
Summary Startup Report - Cycle 12
NRC Docket No. 50-249

Dr. Murley:

Enclosed for your information and use is the Dresden Station Unit 3 Cycle 12 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.

Very truly yours,

J. A. Silady

J. A. Silady
Nuclear Licensing Administrator

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Attachment

cc: A.B. Davis - Regional Administrator, Region III
S.G. DuPont - Senior Resident Inspector - Dresden
B.L. Siegel - Project Manager, NRR

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DRESDEN UNIT 2

CYCLE 12

STARTUP TEST REPORT

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DRESDEN UNIT 2

CYCLE 12

STARTUP TESTING SUMMARY

Dresden Unit 2 resumed commercial operation for Cycle 12 on February 21, 1989, following a refueling and maintenance outage. During the outage, the fourth reload of Advanced Nuclear Fuels Corporation (formerly Exxon Nuclear Company) fuel was installed. The reload consisted of 200 9x9 fuel assemblies. This was the second reload of 9x9 fuel for Unit 2. An operating license amendment was submitted for NRC review on August 25, 1988 to facilitate CECo reviews of this and subsequent reloads per 10 CFR 50.59. The amendment was approved by letter from B.L. Siegel to H.E. Bliss dated January 6, 1989. CECo notified the NRC Staff of completion of the CECo review of the Cycle 12 reload licensing analyses and applicability of 10 CFR 50.59 by letter from J.A. Silady to T.E. Murley dated February 8, 1989.

The startup test program was similar to that performed for previous reloads at Dresden 2 and 3. The program consisted of various physics tests (shutdown margin, critical eigenvalue comparison, etc.), and instrument calibrations (LPRM, TIP's, flow instrumentation) as addressed by the Technical Specifications, Final Safety Analysis Report, and previous commitments to the Nuclear Regulatory Commission. No unusual conditions were noted and the test results were as expected.

Summaries of the startup tests identified in the Draft Regulatory Guide SC 521-4 on refueling and startup tests for LWR reloads are attached per DPR-19 Technical Specification 6.6.A.1. Additional test results are available at the site.

DRESDEN UNIT 2

CYCLE 12

STARTUP TEST NO. 1

CORE VERIFICATION AND AUDIT

PURPOSE

The purpose of this test is to visually verify that the core is loaded as intended.

CRITERIA

The as-loaded core must conform to the reference core design used in the various licensing analyses. At least one independent party must either participate in performing the core verification or review a videotaped version prior to unit startup. Any discrepancies discovered in the loading will be promptly corrected and the affected areas reverified to ensure proper core loading prior to unit startup.

Conformance to the reference core design will be documented by a permanent core serial number map signed by the audit participants.

RESULTS AND DISCUSSION

The Cycle 12 core verification consisted of a core height check performed by the fuel handlers and two videotaped passes over the core by the nuclear group. The height check verifies the proper seating of an assembly in the fuel support piece while the video-tapes verify proper assembly orientation and location. On Janury 19, 1989, the core was verified as being properly loaded and consistent with the Advanced Nuclear Fuels Cycle 12 core reload design. Therefore, the as-loaded core configuration is consistent with that assumed in the evaluation of the Dresden Unit 2 Cycle 12 Reload Licensing Analyses.

DRESDEN UNIT 2

CYCLE 12

STARTUP TEST NO. 2

CONTROL ROD OPERABILITY AND SUBCRITICALITY CHECK

PURPOSE

The purpose of this test is to ensure that no gross local reactivity irregularities exist, that each control blade is latched to its control rod drive, and that all control blades are functioning properly.

CRITERIA

The following must be met:

1. Following the core reload, each control blade will be withdrawn and reinserted. This will guarantee that the mobility of the control blade is not impaired.

2. During control blade movement, the process computer or an alternate method is utilized to time the travel of the blade between notch positions in order to verify proper withdrawal and insertion times.

3. After the core is fully loaded, each control blade will be withdrawn and inserted individually to assure that criticality will not occur. As it is withdrawn, nuclear instrumentation will be monitored to verify subcriticality. Once withdrawn, each control blade is tested for overtravel by continually applying a withdrawal signal. A blade fails this check if rod position indication is not evident or if an overtravel alarm is received.

RESULTS AND DISCUSSION

All control rod drive functional tests were completed by January 18, 1989. After performing these tests, all control blades demonstrated acceptable mobility, proper withdrawal and insertion times, and subcriticality. In addition, all blades but one passed their overtravel checks.

On January 17, control rod N-7 was found to be uncoupled when it was fully withdrawn for an overtravel check. The control rod drive was replaced on February 17. Subsequent overtravel testing showed rod N-7 to be properly coupled to its drive.

DRESDEN UNIT 2

CYCLE 12

STARTUP TEST NO. 3

TIP SYSTEM SYMMETRY - UNCERTAINTY

PURPOSE

The purpose of this test is to perform a gross symmetry check and a detailed statistical uncertainty analysis on the Transversing In-Core Probe (TIP) System.

CRITERIA

1) TIP Symmetry - Gross Check

   The maximum deviation between symmetrically located TIP pairs of LPRM strings should be less than 25%.

2) TIP Symmetry - Statistical Check

   The calculated χ^2 of the integrated TIP responses should be less than 34.81.

NOTE: One data set may be used to meet the above criteria. If either criteria is not met, the instrumentation and data processing system should be checked for any problems that could lead to asymmetries. If the problem persists, the fuel vendor should be consulted to assure that the larger than expected TIP asymmetries do not significantly affect core monitoring calculations.

RESULTS AND DISCUSSION

One complete set of data required for evaluating TIP uncertainty was obtained during the D2 BOC12 Startup Testing Program on March 3, 1989. Data were obtained at a steady state power level, 97% of rated. The control rod pattern maintained mirror symmetry across the axis that defines the line of symmetry for the TIP system. The results for each method of analysis are summarized below.

1) TIP Symmetry - Gross Check

   In order to determine the overall symmetry of the TIP system, the machine-normalized, 6-inch TIP readings were obtained and averaged over nodes 1 through 24 for each symmetric TIP pair (the symmetric locations are given in Table 3.1). The absolute percent deviation for each symmetric TIP pair was calculated and is summarized in Table 3.2. The average absolute deviation for all symmetric TIP pairs was 9.78%, with a maximum absolute deviation of 20.55% which is below the 25% criteria.

2) TIP Symmetry - Statistical Check.

The TIP symmetry statistical analysis was performed using the standard χ^2-test as recommended by Advanced Nuclear Fuels. The machine-normalized, 6-inch TIP readings obtained from a TIP set performed on March 3, 1989 were used for the analysis. These TIP readings were summed over nodes 3 through 22 for each TIP tube location. The percent relative difference (Dm) for each symmetric TIP pair was then calculated using equation 3.1 (the results are summarized in Table 3.3). The TIP data variance (S^2TIP_{ij}) was calculated to be 60.55 using equation 3.2 and χ^2 was calculated to be 30.27 using equation 3.3. Note that the value for χ^2 is within the limit of 34.81 established by Advanced Nuclear Fuels.

Dresden is aware that, although we are still within limits, this value of χ^2 is significantly higher than that obtained during start-up testing for the previous cycle (χ^2 = 9.02 for Dresden 2 Cycle 11). Dresden is continuing to investigate this phenomenon.

TABLE 3.1. Symmetric TIP Locations

TIP PAIR	LPRM	TIP PAIR	LPRM
1	08-17 16-09	10	24-33 32-25
2	08-25 24-09	11	24-41 40-25
3	08-33 32-09	12	24-49 48-25
4	08-41 40-09	13	24-57 56-25
5	08-49 48-09	14	32-41 40-33
6	16-25 24-17	15	32-49 48-33
7	16-33 32-17	16	32-57 56-33
8	16-41 40-17	17	40-49 48-41
9	16-49 48-17	18	40-57 56-41

TABLE 3.2. TIP Symmetry - Gross Check

Symmetric TIP Pair	Absolute Percent Deviation
1	12.90
2	9.77
3	7.75
4	5.39
5	12.97
6	8.38
7	9.94
8	8.10
9	7.79
10	4.37
11	0.66
12	17.46
13	8.58
14	5.79
15	7.43
16	16.31
17	11.91
18	20.55

Average Absolute Percent Deviation: 9.78

Maximum Absolute Percent Deviation: 20.55

TABLE 3.3. TIP Symmetry - Statistical Check

Symmetric TIP Pair	Relative Difference Dm
1	13.277
2	10.297
3	7.700
4	5.611
5	13.482
6	8.123
7	9.636
8	8.162
9	8.464
10	3.739
11	0.870
12	17.321
13	8.610
14	5.642
15	7.033
16	16.565
17	11.987
18	20.963

Equation 3.1 $$Dm = \frac{100\,(Tm_1 - Tm_2)}{\left(\frac{Tm_1 + Tm_2}{2}\right)}$$

Note: $Tm_1 = \sum_{k=3}^{22} T_1(k)$ for TIP_1 and $TM_2 = \sum_{k=3}^{22} T_2(k)$ for TIP_2

Where TIP_1 and TIP_2 are symmetric TIP pairs, and $T_1(k)$ and $T_2(k)$ are the machine normalized, 6-inch TIP readings for the respective TIP pair locations.

Equation 3.2 (Variance)

$$S^2_{TIP_{ij}} = \frac{\sum_{m=1}^{18} Dm^2}{36} = 60.55$$

Equation 3.3

$$\chi^2 = \frac{18(S^2_{TIP_{ij}})}{36} = 30.27$$

DRESDEN UNIT 2

CYCLE 12

START-UP TEST NO. 4

INITIAL CRITICALITY COMPARISON

PURPOSE

The intent of this procedure is to perform a critical Eigenvalue comparison. This is done by comparing the predicted control rod pattern to the actual control rod pattern at criticality taking into account period and temperature coefficient corrections.

CRITERIA

The actual cold critical rod pattern shall be within 1.0% $\Delta K/K$ of the predicted control rod pattern. If the difference is greater than $\pm$ 1.0% $\Delta K/K$, Advanced Nuclear Fuels and Commonwealth Edison Company Core Management Engineers will be promptly notified to investigate the discrepancy. The Nuclear Regulatory Commission will be notified within 24 hours.

RESULTS AND DISCUSSION

Unit 2 went critical on February 19, 1989 at 18:20 hours utilizing an A-2 sequence. The moderator temperature was 155°F and the period was 79.8 seconds. Advanced Nuclear Fuels predictions and rod worths were calculated using the XTGBWR Code, which assumed a moderator temperature of 170°F.

After corrections were made for temperature and period, the actual critical was within 0.1913% $\Delta K/K$ of the predicted critical. This is well within 1.0% $\Delta K/K$ of the predicted critical. Table 4-1 summarizes the results.

TABLE 4-1

INITIAL CRITICALITY COMPARISON CALCULATIONS

ITEM	k/k
k_{eff} with all rods in adjusted to 170°F	= 0.9518
ρ inserted by group 1 rods	= 0.0337 *
ρ inserted by group 2 rods at criticality	= 0.01636 *
Predicted k_{eff} at critical rod pattern (170°F)	= 1.00186*
Moderator temperature coefficient	= -5.0×10^{-5} (Δ k/k)/°F *
Temperature correction between 155°F and 170°F	= +0.00075
Predicted k_{eff} with temperature correction at critical rod pattern	= 1.00261
k_{eff} at time of criticality with ∞ period	= 1.000
Period correction for 79.8 second period	= +0.000697**
Actual k_{eff} with 79.8 second period	= 1.000697
(Predicted k_{eff} - actual k_{eff})	= 0.001913 Δ k/k
Percent Difference	= 0.1913% Δ k/k

SOURCES

* Letter, D. F. Kelter to E. D. Eenigenburg, dated January 12, 1989 Supplemented by letter dated February 7, 1989.

** ρ vs. τ tables