DRESDEN UNIT 3

CYCLE 9

STARTUP TESTING SUMMARY

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Dresden Unit 3 resumed commercial operation for Cycle 9 on July 24, 1984 following a refueling and extended main turbine repair outage. During the outage, the second reload of Exxon Nuclear Company (ENC) fuel was installed. The reload consisted of 184 prepressurized 8 x 8 assemblies with natural uranium blankets on each end. Also, eight ASEA-ATOM control blades were installed in support of an EPRI sponsored program aimed at demonstrating longer blade exposure lifetimes.

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, moderator temperature coefficient, etc.), instrument calibrations (LPRM, TIP's, flow instrumentation), and determination of baseline recirculation flow data as addressed by the Technical Specifications, Final Safety Analysis Report, and previous commitments to the Nuclear Regulatory Commission. A test to confirm the reactivity characteristics of ASEA control blades was also conducted during startup. A summary of the test is enclosed. No unusual conditions were noted and all 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-25 Technical Specification 6.6.A.1. Additional test results are available at the site.

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Commonwealth Edison
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Address Reply to: Post Office Box 767
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October 17, 1984

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

Subject: Dresden Station Unit 3
Summary Startup Test
Report - Cycle 9
NRC Docket No. 50-249

Dear Mr. Denton:

Enclosed for your information and use is the Dresden Station Unit 3, Cycle 9 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 - Dresden

Attachment

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DRESDEN UNIT 3
CYCLE 9
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 be properly loaded 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 9 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 videotapes verify proper assembly orientation and location. During the height check, control cells J-3 and L-8 were found to have assemblies seated higher than the remaining assemblies in the core. These assemblies were correctly reseated by removing and reinserting them. On February 29, 1984, the core was verified as being properly loaded and consistent with Exxon Nuclear Cycle 9 core design. Therefore, the as-loaded core configuration is consistent with what Exxon Nuclear used in their evaluation of Dresden Unit 3 Cycle 9 Reload Licensing Analyses.

DRESDEN UNIT 3
CYCLE 9
STARTUP TEST NO. 2
CONTROL ROD OPERABILITY AND SUBCRITICALITY CHECK

PURPOSE

The intent 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:

a) Each control blade will be withdrawn after the four fuel assemblies in the given control cell are loaded. This will guarantee that the mobility of the control blade is not impaired.

b) During control blade movement, the process computer is utilized to time the travel of the blade between notch positions and verify proper withdrawal and insertion times.

c) 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 (SRM's) 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

Every control blade was withdrawn, checked for overtravel, and inserted to position 00 after fuel was loaded in that given cell. Therefore, each control blade's mobility was assured.

All control blades were timed during insertion and withdrawal and were found to be acceptable.

During and after core loading, each control blade was withdrawn to position 48 to verify subcriticality. The SRM's were observed during the withdrawal and subcriticality was confirmed for every control blade. All control blades also passed their overtravel checks.

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

One complete set of data required for evaluating TIP uncertainty was obtained during the D3 BOC9 Startup Testing Program. Data was obtained at steady state power levels greater than 75% of rated power. 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, full power adjusted TIP readings were obtained and averaged for each symmetric TIP pair (the symmetric locations are given in Table 3.1). The absolute percent deviation between each symmetric TIP pair was calculated and summarized in Table 3.2. The average absolute deviation for all symmetric TIP pairs was 5.71%, with a maximum absolute deviation of 18.3% which is well below the 25% criteria.

2) TIP Symmetry - Statistical Check.

The TIP symmetry analysis was performed using the standard χ^2 test as recommended by Exxon Nuclear Company. TIP values obtained from a whole core LPRM calibration performed during the startup test program were summed (elevations 5 through 44) for each TIP location. The absolute relative difference (Dm) for each symmetric TIP pair was then calculated using equation 3.1 - the results are summarized in Table 3.3. From equations 3.2 and 3.3 the variance and χ^2 were calculated to be 31.5 and 15.8 respectively. Note that the value for χ^2 is well within the limit established by Exxon of 34.81.

LE 3.1. Symmetric TIP Location

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

TABLE 3.2. TIP Symmetry - Gross Check

Symmetric TIP Pair	Absolute Percent Deviation
1	8.79
2	.740
3	1.93
4	10.8
5	10.9
6	8.41
7	.140
8	1.24
9	13.4
10	1.55
11	1.39
12	3.13
13	18.3
14	2.61
15	.386
16	6.83
17	8.51
18	3.73

Average Absolute Percent Deviation: 5.71

Maximum Absolute Percent Deviation: 18.3

TABLE 3.3. TIP Symmetry - Statistical Check

Symmetric TIP Pair	Relative Difference Dm
1	9.67
2	.839
3	2.21
4	11.6
5	11.3
6	8.33
7	.662
8	2.10
9	12.7
10	1.79
11	.577
12	4.59
13	18.6
14	3.75
15	1.53
16	7.28
17	8.53
18	4.38

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

Note: $Tm_1 = \sum_{k=5}^{44} T(k)$ for TIP_1 and $TM_2 = \sum_{k=5}^{44} T(k)$

for TIP_2 where TIP_1 and TIP_2 are symmetric TIP pairs.

Equation 3.2 (Variance)

$$S^2_{TIP_{ij}} = \frac{\sum_{m=1}^{18} Dm^2}{36} = 31.5$$

Equation 3.3

$$\chi^2 = \frac{18 S^2_{TIP_{ij}}}{36} = 15.8$$

DRESDEN UNIT 3
CYCLE 9
STARTUP 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% ΔK/K of the predicted control rod pattern. If the difference is greater than $\pm$ 1.0% Δk/k, Exxon Nuclear Company and Commonwealth Edison Company Core Management Engineers will be promptly notified to investigate the anomaly.

RESULTS AND DISCUSSION

Unit 3 went critical on March 22, 1984 at 8:56 a.m. utilizing an A-2 sequence. The moderator temperature was 156°F and the period was 120 seconds. Exxon Nuclear 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 1.0% Δk/k of the predicted critical. Table 4-1 summarizes the results.

TABLE 4-1

INITIAL CRITICALITY COMPARISON CALCULATIONS

Predicted (XTGBWR Code) k_{eff}:

k_{eff} with all rods in adjusted to 170°F	= 0.9395 *
Reactivity inserted by all group 1 rods	= 0.037 *
Reactivity inserted by all group 2 rods	= 0.0199 *
Reactivity inserted by additional rods from group 3 at criticality	= 0.00148 *
Predicted k_{eff} at critical rod pattern (170°F)	= 0.99788

Temperature correction to predicted k_{eff}:

Moderator temperature coefficient = -4.1×10^{-5} ($\Delta k/k$)/°F *

Temperature correction between 156°F and 170°F	= 0.000574
Predicted k_{eff} at critical rod pattern (156°F)	= 0.998454

Period correction to actual k_{eff}:

k_{eff} at time of criticality with ∞ period	= 1.000
Period correction for 120 second period	= 0.0005146 **
Actual k_{eff} with 120 second period	= 1.0005146

Difference:

$|\text{XTGBWR } k_{eff} - \text{actual } k_{eff}| \times 100\%$ = 0.206% $\Delta k/k$

SOURCES

* Exxon letter, L.C. O'Malley to L.J. Scott, dated Dec. 21, 1983.

** ρ vs. τ tables

DRESDEN UNIT 3

START-UP TEST NO. 5

ASEA-ATOM CONTROL BLADE TESTS

A. PURPOSE

The purpose of this test was to verify the operability of the ASEA-ATOM (A-A) control blades and the acceptability of these control blades with respect to blade worth. Blade worth verification was based on comparisons of measured to predicted responses of Core Thermal Power (CTP), Traversing Incore Probes (TIP's) values and Local Power Range Monitors (LPRM's) readings resulting from the A-A control blade movement.

B. CRITERIA

1) Each of the 8 ASEA-ATOM control blades will be checked for sub-criticality after the four fuel assemblies in the given control cell are loaded, friction tested to ensure proper rod motion (ref. Dresden Technical Staff Surveillance 300-6, Control Rod Drive Friction Testing), and hot scram tested to verify that all insertion times meet the requirements of Technical Specification 3.3.C.

2) The effect of an A-A control rod movement on TIP's and LPRM's will be examined by comparing normalized, predicted TIP and LPRM data at the four closest LPRM axial locations to the actual TIP and LPRM readings after the control rod is withdrawn. The A-A control rod will be moved from the fully inserted position (00) to the full out position (48) and, then, the actual CTP was calculated. The actual values after the rod pull for CTP, TIP's and LPRM's will be compared and should agree to within 10%, i.e.

$$\Delta CTP = \frac{\left|\Delta CTP^{actual} - \Delta CTP^{predicted}\right|}{\Delta CTP^{actual}} < \varepsilon_1 = 0.10 \qquad (1)$$

$$\Delta TIP = \frac{\left|TIP_f^{\ actual} - TIP_f^{\ norm}\right|}{TIP_f^{\ actual}} < \varepsilon_2 = 0.10 \qquad (2)$$

$$\Delta LPRM = \frac{\left|LPRM_f^{\ actual} - LPRM_f^{\ norm}\right|}{LPRM_f^{\ actual}} < \varepsilon_3 = 0.10 \qquad (3)$$

where

ΔCTP^{act} = actual Core Thermal Power change after the rod movement to the final position.

ΔCTP^{pred} = predictive Core Thermal Power change after the rod movement to the final position.

$TIP_f^{\ act}$ = actual TIP trace value after the rod pull.

$TIP_f^{\ norm}$ = normalized TIP predicted values at the final rod position.

$LPRM_f^{\ act}$ = actual LPRM reading after the rod pull.

$LPRM_f^{\ norm}$ = normalized LPRM reading at the final rod position.

The 10% criterion is based on the previous comparison of the on-line core monitoring code versus the offline results. Failure to meet this criterion would not necessarily indicate control blade problems but would warrant further investigation.

C. RESULTS AND DISCUSSIONS

1) All 8 ASEA-ATOM control blades were found to meet the sub-criticality check, the control rod drive friction times and the hot scram insertion times criteria.

2) In order to ensure that the neutronic characteristics of the A-A blades are consistent with the POWERPLEX predictions, a blade worth test was performed on July 27, 1984. The test was performed during the start-up testing program with the reactor at approximately 45% rated core thermal power and steady state. Xenon conditions were near equilibrium. One of Hafnium-tipped and one of non Hafnium-tipped A-A blades were tested and both have yielded satisfactory results.

The Hafnium-tipped blade at reactor location 30-15(H-4) and the non Hafnium-tipped blade at reactor location 38-39(K-10) were used for this test. The special test procedure (SP 84-3-20) called for the withdrawal of each test rod (30-15 or 38-39) from full in to full out. In each of the test cases, a POWERPLEX prediction was performed to determine expected TIP and LPRM values and to estimate the resulting increase in core thermal power (CTP) when the rod was fully withdrawn. This was accomplished by iterating on an estimate of the final CTP until the k_{eff} for the predict run with the rod at 48 (full out position) matched the k_{eff} for the run with the rod at 00 (fully inserted position). Actual TIP data were obtained from the reactor locations that closest monitored the test rods.

Once the POWERPLEX predictive results and TIP data were obtained, the A-A rod was then moved from 00 to 48 and the actual CTP was calculated. TIP data was then obtained with the rod at 48. Using equation (1), the actual and predicted values of ΔCTP for each case was found to agree within 3.0%. This is shown in Table 1. In comparing the normalized six-inch TIP predicted values at the LPRM locations with the actual TIP trace values, as shown in Table 2, it was found that for rod 30-15, the largest percent difference between calculated and actual value, which was calculated using equation (2), was 5.20% at the A-level of TIP location 32-17. And, for rod 38-39, the largest percent difference occurred at the D-level of location 40-41 with a value of 5.42%. The effect of the rod movement was also examined by comparing the normalized, predicted LPRM readings with the actual readings and the difference was calculated using equation (3).

The LPRM string at reactor location 32-17 has a maximum difference of 4.52% at the A-level and LPRM string 40-41 yields the largest difference of 6.29% at the B-level. The detailed results of this comparison are shown in Table 3.

In summary, all three parameters used for comparison purposes are well within the 10% criterion and the results of each of the above comparisons clearly indicates that the ASEA-ATOM control blade reactivity worth behaved as anticipated and is accurately modeled by the current fuel vendor's core monitoring code. The outcome of this test yielded the anticipated results.

Table 1

Rod I.D.	ΔCTP^{pred}	ΔCTP^{act}	ΔCTP
30-15 (H-4)	46.9	46.12	1.69%
38-39 (K-10)	44.9	46.12	2.64%

Table 2

Rod I.D.	TIP I.D.	Level	TIP_f^{norm}	TIP_f^{act}	ΔTIP
30-15 (H-4)	32-17	D	17.51	18.2	3.79%
		C	21.21	21.8	2.71%
		B	26.99	26.4	2.23%
		A	51.34	48.8	5.20%
38-39 (K-10)	40-41	D	18.16	19.2	5.42%
		C	20.82	21.5	3.16%
		B	26.04	26.7	2.47%
		A	41.82	42.2	0.90%

Table 3

Rod I.D.	LPRM I.D.	Level	$LPRM_f^{norm}$	$LPRM_f^{act}$	$\Delta LPRM$
30-15 (H-4)	32-17	D	16.06	16.2	0.86%
		C	21.33	21.5	0.79%
		B	25.06	24.0	4.42%
		A	50.69	48.5	4.52%
38-39 (K-10)	40-41	D	17.16	17.1	0.35%
		C	21.59	20.7	4.30%
		B	26.04	24.5	6.29%
		A	46.40	44.5	4.27%