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SUBJECT: "Susquehanna Steam Electric Station, Unit 1, Cycle 4 Startup  
Test Summary."

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SUSQUEHANNA SES UNIT 1 CYCLE 4

STARTUP TEST SUMMARY

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## ABSTRACT

### Susquehanna Unit 1 Cycle 4 Startup Test Summary

Susquehanna Unit 1 resumed commercial operation for Cycle 4 on November 24, 1987 following a 10 week refueling and maintenance outage. The Unit 1 Cycle 4 (hereafter referred to as SIC4) reload included:

*	36	GE	8x8	initial core
	192	ANF	8x8	twice burned
	296	ANF	8x8	once burned
	240	ANF	9x9	unirradiated fuel assemblies

The following startup tests, identified in the SIC4 Reload Licensing Submittal, are discussed in this report:

- |     |  |
|-----|--|
| 1.0 | Core Loading Verification and Audit                    |
| 2.0 | Control Rod Functional (Insert and Withdrawal Checks)  |
| 3.0 | Subcritical Shutdown Margin Demonstration              |
| 4.0 | In-Sequence Critical and Shutdown Margin Determination |
| 5.0 | TIP Asymmetry  |

In addition to the above mentioned tests, the startup program included a POWERPLEX input deck validation, scram time testing, core flow and LPRM calibrations, thermal limits monitoring and baseline recirculation data acquisition. A summary of these activities is also included in this report.

(ANF - Advanced Nuclear Fuels)



Susquehanna Unit 1  
Cycle 4  
Startup Test No. 1  
Core Verification and Audit

Purpose

The purpose of this test is to visually verify that the core is loaded per the analyzed designs.

Criteria

Upon completion of core alterations during the refueling outage, the core must be verified to conform with the reference core design used in the various licensing analyses. The verifications to be performed include fuel bundle location, fuel bundle orientation, and proper seating of the fuel bundles within the core. The verifications will be performed by the Reactor Engineering Group utilizing an underwater television camera. The verification will be videotaped so that an independent verification may be performed. Any discrepancies discovered in the loading will be promptly corrected and the affected bundles shall be reverified prior to unit startup.

Results

The UIC4 core was analyzed to have a 1.38%  $\Delta K/K$  shutdown Margin with the strongest rod fully withdrawn at BOC 4. (Startup and Operations Letter Report, Susquehanna Unit 1 Cycle 4). This figure was significantly less than previous cycles, therefore the following precautions were taken to prevent a misloaded fuel bundle. During the offload all bundles were placed in the fuel pool in the order in which they were to be reloaded, a pool verification was performed (10/7/87) of all fuel before the reload commenced and a partial core verification was performed (10/22/87) after all irradiated bundles were placed in the core, before any new fuel was loaded.

The Cycle 4 final core verification consisted of two videotaped passes over the core. During the first pass, the fuel bundle serial numbers were recorded on the videotape to verify proper location. The second pass was performed to verify proper fuel assembly seating (assembly height check) and correct orientation.

The core tapes were independently verified to be correct by the Reactor Engineering Supervisor and a representative of Quality Control on 10/27/87. Therefore, the as-loaded core configuration is consistent with the core design Advanced Nuclear Fuels used in the evaluation of the SIC4 Reload Licensing Analyses. The SIC4 core map is included as Figure 1.





Susquehanna Unit 1  
Cycle 4  
Startup Test No. 2  
Control Rod Functional (Insert and Withdrawal Checks)

Purpose

The purpose of this startup test is to assure proper control rod function and demonstrate that criticality will not occur due to the withdrawal of a single rod.

Criteria

Control Rod Functionals include mobility, overtravel and subcritical checks. These may be performed as each control cell is loaded in its final configuration.

Each control rod will be cycled individually to ensure mobility. As each rod is fully withdrawn, it will be checked for overtravel by continually applying a withdrawal signal. Subcriticality will also be verified with the rod withdrawn.

Results

Due to Shutdown Margin considerations, no control rod functionals were performed on fully loaded control cells until the partial core verification was completed on 10/22/87. No control rods overtraveled and subcriticality was maintained as each rod was individually fully withdrawn and reinserted.



Susquehanna Unit 1  
Cycle 4  
Startup Test No. 3  
Subcritical Shutdown Margin Demonstration

Purpose

The purpose of this startup test is to assure at least the minimum required shutdown margin exists with the strongest worth control rod fully withdrawn.

Criteria

The minimum required shutdown margin at BOC for Susquehanna Unit 1 Cycle 4 is  $0.75\% \Delta K/K$ . This test will verify at least this amount by performance of a subcritical shutdown margin demonstration. The highest (strongest) worth control rod is fully withdrawn, then a diagonally adjacent rod is slowly notched out verifying subcriticality at each step until the analytically determined reactivity worth of the control rods at their respective notch position equals or slightly exceeds the required amount of SDM.

Results

The reactor remained subcritical with the highest worth control rod fully withdrawn and an additional diagonally adjacent rod pulled to a notch position with a calculated worth of  $1.143\% \Delta K/K$ . The required shutdown margin to be demonstrated was calculated to be  $1.042\% \Delta K/K$ . This is  $.75\% \Delta K/K$  plus a correction factor for the recirculation loop temperature (141 degrees F) at the time of the test. Using data supplied by ANF it was determined that the following rods pulled to the indicated position would demonstrate a shutdown margin of  $1.143\% \Delta K/K$ .

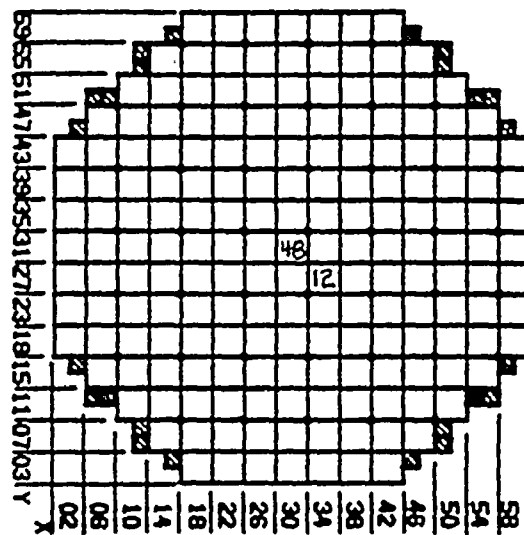
<u>ROD</u>	<u>POSITION</u>	<u>TOTAL WORTH % <math>\Delta K/K</math></u>
30-31*	48	-
34-27	12	1.143

\*analytically determined strongest rod.

Due to the preliminary BOC SDM calculation of  $1.38\% \Delta K/K$ , precautions were taken during the performance of this test to reduce individual rod notch worths. Overtravel checks were performed on both rods and the notch-down procedure was used to minimize the effect of the high incremental notch worths of the adjacent rod.

As rods were pulled, subcriticality was verified after each notch. Subcriticality was also verified with the rods at the above indicated positions, thus satisfying the purpose of this startup test. Figure 2 is a core map showing the test rod positions.

FIGURE 2. CORE MAP SHOWING TEST ROD POSITIONS FOR  
SUBCRITICAL SHUTDOWN MARGIN DEMONSTRATION



Susquehanna Unit 1  
Cycle 4  
Startup Test No. 4  
In-Sequence Critical and SDM Determination

Purpose

The purpose of this startup test is to calculate the actual shutdown margin of the cycle 4 core and to demonstrate that no reactivity anomaly existed.

Criteria

1) Shutdown Margin

Technical Specification 3.1.1 requires an adequate shutdown margin to ensure the reactor can be made subcritical from all operating conditions. This value,  $.38\% \Delta K/K$  has been determined to be the minimum required SDM to bring a reactor subcritical under the worst case conditions - a cold, xenon-free core at the most reactive point in the cycle with the highest worth control rod unavailable for reactivity control. At beginning of cycle, the required SDM value must be increased by a factor, R, if it is determined that core shutdown margin is less at another point in the cycle than the initial shutdown margin (for Cycle 4,  $R = 0$ ). A prediction uncertainty of  $.37\% \Delta K/K$  is also added at BOC to assure the validity of the analytical calculations. The required beginning-of-cycle SDM for Susquehanna Unit 1 Cycle 4 is  $0.75\% \Delta K/K$ ; the actual SDM will be calculated from data obtained during the initial startup criticality.

2) Reactivity Anomaly

Core reactivity is monitored to prevent excessive reactivity additions due to unforeseen reactivity changes or reactivity anomalies. At BOC, a  $1\% \Delta K/K$  difference between predicted and actual critical control rod positions might indicate improper core loading or a computer code that is unreliable. Data gathered during the in-sequence critical, specifically the Keff at the notch position of the control rod at which criticality occurs is compared to predicted critical control rod position Keff and a % reactivity difference is calculated.

Results

The calculated SDM was  $1.46\% \Delta K/K$  and the difference between actual Keff and predicted Keff at criticality was  $0.08\% \Delta K/K$ .

Control rods were withdrawn in the B sequence until the reactor was on a stable, positive period. The notch position at which criticality concurred was rod 38-47, notch 18, step 35. A special log was initiated to record SRM count rates and recirculation loop temperatures. The average period was 194 seconds and the average loop temperature  $153.5$  degrees F which yield period and temperature corrections of  $.332 \times 10^{-3} \Delta K/K$  and  $3.42 \times 10^{-3} \Delta K/K$  respectively.

1) Shutdown Margin

The equation used to calculate SDM

$$SDM = \frac{K_{crit} - K_{sro}}{K_{crit} * K_{sro}} - \Delta p \text{ (period)} - \Delta p \text{ (temp)}$$

Kcrit is Keff at the actual critical control rod position (1.00457) and Ksro is Keff predicted with the strongest rod out (0.98642)

The minimum required SDM for Unit 1 Cycle 4 at beginning of cycle was 0.75%  $\Delta K/K$ ; the calculated shutdown margin based on this test was 1.46%  $\Delta K/K$ , thus satisfying the acceptance criteria.

2) Reactivity Anomaly

The reactor went critical at step 35 with Kcrit of 1.00457. The equation used to calculate reactivity difference was

$$\text{Reactivity difference} = \frac{K_{crit} - 1}{K_{crit}} - \Delta p \text{ (period)} - \Delta p \text{ (temp)}$$

The calculated reactivity difference was 0.08%  $\Delta K/K$ . This satisfies  $\pm 1\% \Delta K/K$  acceptance criteria.

A comparison of the predicted versus actual critical control rod patterns is included as Figure 3.

[illegible][illegible]

ACTUAL CRITICAL PATTERN

Susquehanna Unit 1  
Cycle 4  
Startup Test No. 5  
Tip Asymmetry

Purpose

The purpose of this test is to check core symmetry by performing a statistical uncertainty analysis on the Traversing In-Core Probe (TIP) System. Also, by the performance of this test, the proper operation of the TIP system will be assured.

Criteria

The  $\chi^2$  test of significance will be performed with the significance level fixed at 1%. The test will be performed utilizing an octant symmetric rod pattern at a power level greater than 75% of rated power.<sup>2</sup> The startup test criteria for symmetric TIP differences is that the  $\chi^2$  value calculated shall be less than the critical  $\chi^2$  value. Since Susquehanna has 19 symmetric TIP pairs, the calculated  $\chi^2$  value must be less than a critical  $\chi^2$  value of 36.19 (as determined by ANF). If the calculated  $\chi^2$  value exceeds the critical value, the instrumentation and data processing system should be reviewed for any problems which may contribute to abnormal TIP asymmetries. A second determination of  $\chi^2$  should be then made. If the new measured value of  $\chi^2$  exceeds the critical value, the fuel vendor shall be consulted and appropriate action taken to assure that a larger than anticipated TIP asymmetry does not adversely affect the safe operation of the reactor.

Results

A complete set of TIP data was obtained at the completion of Susquehanna Unit 1 BOC4 Startup Testing Program at rated thermal power. The nodal TIP values (Nodes 3 through 22) were summed up for each symmetric TIP pair using equation 5.1 with the results summarized in Table 1. Using Equations 5.2 and 5.3, the variance and  $\chi^2$  were calculated to be 4.96 and 2.62 respectively. The  $\chi^2$  value of 2.62 is well within the 36.19 limit established by ANF.

Table 1  
Absolute Relative Difference

<u>Symmetric TIP Pair</u>	<u>Absolute Relative Difference dm</u>
1	1.04
2	3.08
3	4.35
4	3.22
5	6.57
6	1.92
7	0.02
8	0.02
9	6.15
10	1.89
11	3.61
12	0.98
13	0.94
14	3.27
15	0.85
16	2.07
17	3.45
18	0.18
19	4.22

Equation 5.1

$$dm = \frac{100 (T_{m1} - T_{m2})}{\left( \frac{T_{m1} + T_{m2}}{2} \right)}$$

Note:  $T_{m1} = \sum_{K=3}^{22} T(k)$  for TIP1 and  $T_{m2} = \sum_{K=3}^{22} T(k)$  for TIP2

where TIP1 and TIP2 are symmetric TIP pairs

Equation 5.2 (Variance)

$$S^2_{TIPij} = \frac{\sum_{M=1}^{19} dm^2}{36} = 4.96$$

Equation 5.3

$$x^2 = \frac{19 S^2_{TIPij}}{36} = 2.62$$



Susquehanna Unit 1  
Cycle 4  
Startup Program Summary

Rod Scram Time Testing

Purpose

To demonstrate the maximum scram insertion times of all rods following core alterations.

Criteria

Susquehanna Technical Specification 4.1.3.2 states that scram insertion times of all control rods shall be demonstrated through measurement with reactor coolant pressure greater than 950 psig prior to exceeding 40% thermal power after core alterations. For Unit 1 cycle 4, one-half of all control rods scram times were to be determined by performing a black-and-white scram from the B sequence and using GETARS scram data. The remaining rods were to be individually scram time tested.

Results

Due to a GETARS software problem, control rod scram times from the black-and-white scram were not printed out directly, but had to be calculated from the GETARS stored files. The remaining rods were individually scram timed November 25-26, 1987.

All control rod scram insertion times were determined to be within Technical Specification limits. The results are included as Table 2.

	ROD	ROD POSITION	TIME AS FOUND	T.S. LIMIT
MAXIMUM INDIVIDUAL ROD SCRAM INSERTION TIME T.S. 3.1.3.2	26-27	05	3.36	7.00
AVERAGE SCRAM INSERTION TIME OF OPERABLE RODS T.S.3.1.3.3		45 39 25 05	0.34 0.65 1.40 2.50	0.43 0.86 1.93 3.49
AVERAGE SCRAM INSERTION TIME OF SLOWEST 2x2 ARRAY T.S. 3.1.3.4		45 39 25 05	0.36 0.71 1.55 2.71	0.45 0.92 2.05 3.70

TABLE 2: Results of Scram Time Testing of All Control Rods SIC4.

Susquehanna Unit 1  
Cycle 4  
Startup Program Summary

POWERPLEX INPUT DECK VALIDATION

Purpose

To ensure the POWERPLEX input deck is updated correctly before the start of every new fuel cycle.

Criteria

POWERPLEX is the ANF software system designed to perform in-core monitoring of BWR cores. Core monitoring is performed by the module, XTGBWR, a three-dimensional reactor simulator code which calculated bundle nodal powers. The POWERPLEX input deck consists of all constants needed for the execution of this code and subsequent calculation of the margin to thermal limits. These constants must be updated prior to the start of every new fuel cycle in order to ensure satisfactory core monitorings of the new core configuration. The deck is updated by an ANF core management engineer and validated jointly by members of the Reactor Engineering Group at Susquehanna and the Nuclear Fuels Group in Allentown.

Results

The POWERPLEX input deck was verified to be correct and successfully loaded into the POWERPLEX system prior to SIC4 startup.

Susquehanna Unit 1  
Cycle 4  
Startup Program Summary

The following is a short summary of additional Reactor Engineering activities performed during the Startup Testing Program.

Thermal Limit Monitoring

Thermal Limits were checked throughout the startup period through review of the POWERPLEX core monitoring program, MONITOR, output. At no time did thermal limits exceed Technical Specification limits.

TIP System - OD-1 Performance

A full set of TIPS was run at 30% power to update the core power distribution before the first core performance calculation was initiated. Subsequent TIP sets were performed at 70 and 100% power in conjunction with two LPRM calibrations. The LPRM currents were updated and the LPRM GAFS found to be within the acceptable range.

Power Distribution Comparison with Offline Monitoring

Favorable results were obtained when actual core power distribution data was compared to SIMULATE core modelling code data. The SIMULATE code is used by the Nuclear Fuels core management engineer to predict power distributions throughout the cycle. This comparison, at 0.242 GWD/MTU is included as Figure 4.

Core Flow Calibration

A core flow calibration was performed at 97% core flow. Jet pump and recirculation loop flow instrumentation was adjusted to ensure correct core flow indication and correct calculation of the flow biased rod block and scram setpoints by the APRM flow units.

Recirculation Loop Baseline Data Acquisition

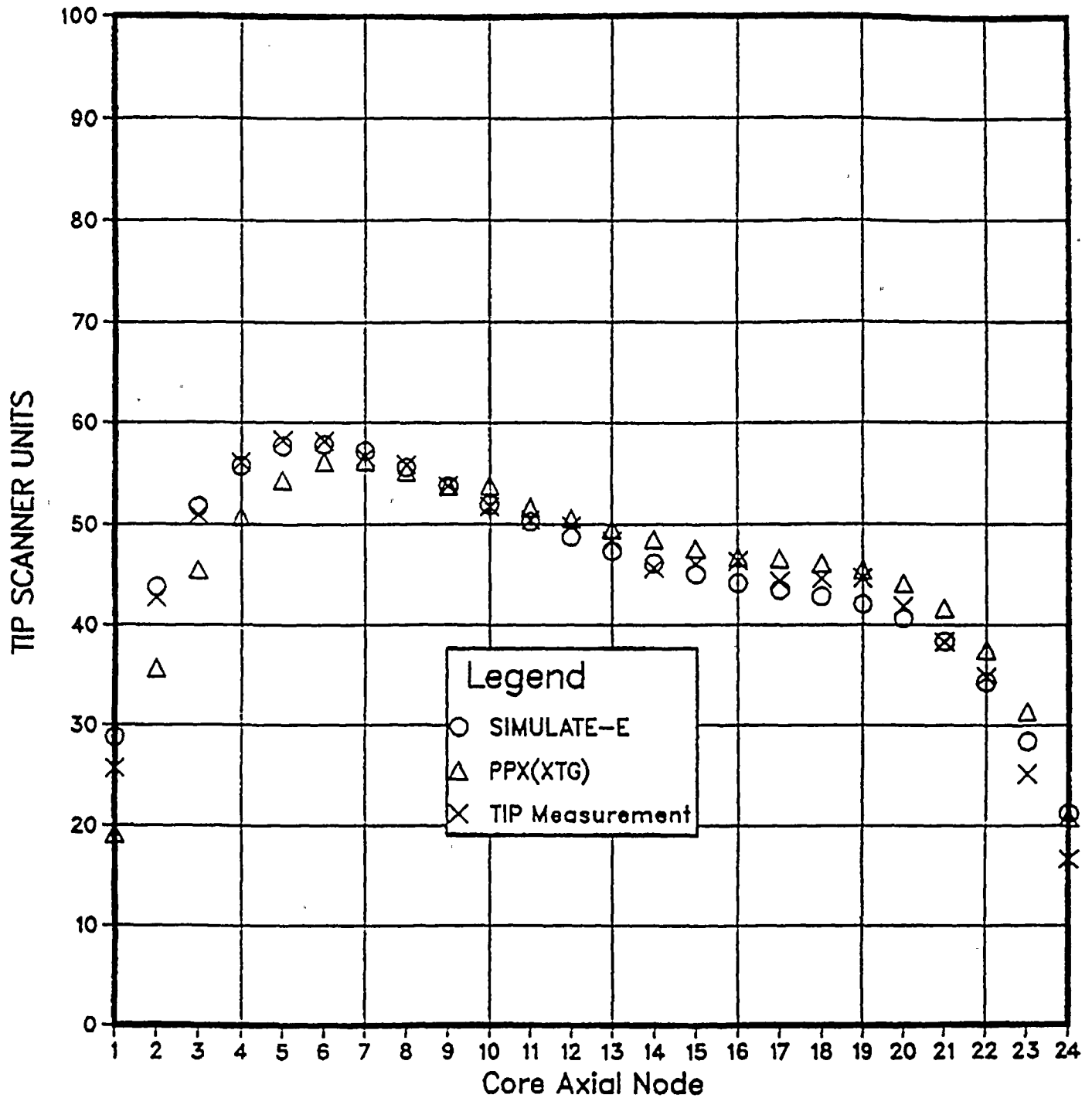
Recirculation loop data was collected throughout the startup program to provide baseline data for plant performance monitoring in two loop and single loop operation. This data is used throughout the cycle during the performance of the technical section Jet Pump Operability Surveillance.

Neutron Noise Baseline Data Recording

APRM and LPRM baseline neutron flux noise data was collected as outlined in Technical Specification 4.4.1.1.4. This data was analyzed and incorporated into the Neutron Flux Noise Level Recording surveillance.

FIGURE 4

U1C4  
CORE AVERAGE TIP COMPARISON AT 0.242 GWD/MTU





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FEB 22 1988

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SUSQUEHANNA STEAM ELECTRIC STATION  
UNIT 1 STARTUP REPORT  
PLA-2984 FILE R41-2A

Docket No. 50-387

Dear Mr. Russell:

Attached is a copy of the Susquehanna SES Unit 1 Startup Report for the Unit 1 Cycle 4 startup. This report is submitted in accordance with Technical Specifications Section 6.9.1.1 through 6.9.1.3. The report addresses those startup tests described in our application for reload dated June 19, 1987 (PLA-2875).

Very truly yours,

H. W. Keiser  
Vice President-Nuclear Operations

Attachment

cc: NRC Document Control Desk (w/original)  
NRC Region I  
Mr. M. C. Thadani - NRC Project Manager  
Mr. F. I. Young - NRC Resident Inspector

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