U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-388/86-29

Docket No. 50-388

License No. NPF-22

Licensee: Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Facility Name: Susquehanna Steam Electric Station, Unit II

Inspection At: Berwick, Pennsylvania

Inspection Conducted: November 3-7, 1986

Inspectors: C Petrone — 11/26/86
C. Petrone, Lead Reactor Engineer — date

Approved by: Jon R. Johnson — 11/26/86
J. Johnson, Chief, Operational Programs Section — date

Inspection Summary: Routine Unannounced Inspection conducted on November 3-7, 1986

Areas Inspected: Post refueling startup testing.

Results: No violations were identified.

DETAILS

1.0 Persons Contacted

Licensee

*T. Crimmins, Plant Superintendent
*J. Blakeslee, Assistant Plant Superintendent
*R. J. Prato, Licensing
*R. Lombard, Acting Reactor Engineer

NRC

*J. Johnson, Chief, Operational Programs Section
*L. R. Plisco, Senior Resident Inspector
*J. Stair, Resident Inspector
*D. LeQuia, Radiation Specialist
*M. Kaminski, Radiation Specialist

*Denotes those present at the exit meeting on November 7, 1986.

2.0 Fuel Cycle Design Report

The inspector reviewed the Susquehanna Unit 2, Cycle 2, Fuel Cycle Design Report, dated April 1986 and Supplement 1 to that report, dated August 1986, provided by Exxon Nuclear Company (ENC). This report describes the Cycle 2 design and presents the results of the fuel management analysis. It contains the reactor core loading and cycle design which was used as the basis for the Cycle 2 licensing calculations including the projected control rod patterns and evaluations of thermal margin at various points throughout Cycle 2. These analyses were performed to verify that adequate cycle length capability, hot excess reactivity, and cold shutdown margin exist for the cycle. The Cycle 2 reload batch is composed of 324 fresh ENC fuel assemblies which contain 79 fueled rods and two water rods in a 9x9 array. Each assembly contains seven burnable poison rods. The exposed fuel in the core consists of 440 once exposed initial core load General Electric 8x8R fuel assemblies. The projected Cycle 2 full power energy capability is 1,435 gwd (10,500 mwd/mt).

This inspection reviewed the status of the startup and power ascension test program with startup test, and reactor engineering personnel. The reactor was at approximately 60% power at that time and was increased to 100% by the end of the inspection. The licensee's planned startup tests included: a core loading verification; control rod functional checks; subcritical shutdown margin demonstration; in-sequence critical and shutdown margin determination; core flow calibration; and HPCI pump performance testing.

3.0 Core Verification

The inspector viewed portions of the videotapes made by the licensee during their verification of the core fuel loading. The inspector verified that the fuel assemblies had been installed in the correct core location with the proper orientation. This review was made of a sample of ten percent of the core. The Serial number of each of these fuel assemblies was verified using the core map contained in the Fuel Cycle Design Report.

During their performance of the core verification, the licensee identified that fuel assembly S/N X21-259 was mis-oriented by 180 degrees. The inspector verified that the fuel assembly had been removed and replaced in the correct orientation. This was confirmed by review of a followup videotape.

4.0 Control Rod Scram Time Measurements

Control rod scram time measurement were performed using procedure SR-255-001, Scram Time Measurement of All Operable Control Rods, Revision 3, dated October 15, 1986. This procedure was used to demonstrate that the maximum scram insertion times for all control rods do not exceed Technical Specification (TS) requirements. This test was performed prior to exceeding 40% Thermal Power following core alterations. Rod drop times can be measured simultaneously using the GETARS computer, or individually using a Gould recorder. The licensee used the GETARS data for most of the 185 rods. Data for approximately twelve of these was not recorded by GETARS and obtained by individual rod testing using a Gould recorder.

The scram time data obtained from both the GETARS and the recorder traced were compiled using a licensee developed computer program NDAI-01 which compared the results to technical specification requirements. These results were:

TS 3.1.3.2 Maximum Individual Rod Scram Insertion Time

Rod	Maximum Time(sec)	TS Time Limit (sec)
42-31	3.16	7.00

TS 3.1.3.3 Average Scram Time of Operable Rods

Dropped From Rod Position	Average Time(sec)	TS Time Limit(sec)
45	0.28	0.43
39	0.58	0.86
25	1.30	1.93
05	2.37	3.49

TS 3.1.3.4 Slowest Four Rod Array/Average of Three Fastest Rods

Rods	Dropped from Rod Withdraw Position	Average Time(sec)	TS Time Limit(sec)
26-39 26-43 30-39 30-43	45	0.29	0.45
26-43 26-47 30-43 30-47	39	0.62	0.92
26-43 26-47 30-43 30-47	25	1.42	2.05
26-43 26-47 30-43 30-47	05	2.56	3.70

Rod scram times met TS requirements.

The inspector independently reviewed the recorder traces for ten control rods and verified that the scram times had been measured accurately and had been input correctly into the computer program.

5.0 Control Rod Subcritical and Rod Functional Check

Control rod subcritical and rod functional checks were performed in conjunction with fuel movement operations. The inspector reviewed the RE-TI-004 fuel movement data sheets and verified that the steps which require performance of cell subcritical and rod functional checks had been signed off as required.

6.0 Shutdown Margin Demonstration

The licensee is required to demonstrate a shutdown margin of (.38% + R) delta k/k where R is the correction for the difference between the Beginning of Cycle (BOC) reactivity and the minimum shutdown margin during the cycle. For this core load the R = 0.17 delta k/k. A shutdown margin demonstration was performed on October 22, 1986, using procedure SR-200-003, Revision 3, dated October 2, 1986. The strongest rod 50-15 was fully withdrawn. Pulling rod 46-19 to rod position 12 gave an uncorrected value of 1.252% delta k/k. Correcting this value for the difference between the actual moderator temperature (122°F) and the "cold" temperature (68°F) gave a corrected value of 1.0468 for the shutdown margin which exceed the minimum required shutdown margin of .39% delta k/k. A moderator temperature coefficient of -3.8×10^{5} delta k/k per degree F was used to correct the reactivity for temperature difference.

The licensee also performed an in-sequence critical shutdown margin demonstration using procedure SR-200-008, Revision 0, dated October 2, 1986. This procedure was used to determine the actual shutdown margin during the first startup after core alterations. This data was used to assure there was no reactivity anomaly by verifying that the actual Keff is within 1% delta k/k of the predicted Keff. Using corrections for moderator temperature and reactor period, the actual measured shutdown margin was 2.794% delta k/k. The reactivity difference between the actual and predicted reactivity was .065% delta k/k which is within the required 1% delta k/k. The reactor engineer stated criticality was achieved within two steps of predicted criticality.

7.0 Core Flow Calibration

The licensee performed a preliminary run through of RE-2TP-022, Core Flow Calibration on November 5, 1986. This was done to checkout the procedure and computer program used to make the calculations. The calculated value of core flow and recirculation loop driving flows are used to calibrate jet pump and recirculation loop flow instrumentation. The inspector witnessed the data taken by an I&C technician and a reactor engineer. The procedure prerequisites were met, operations permission obtained and procedure steps were followed.

The reactor engineer then input the data into an off-line computer to evaluate the data. The computer program did not complete the calculation due to a problem with the input data. Evaluation by the reactor engineer determined that it was caused by a modification which changed out the flow transmitters which provide the A and B Recirculation Loop Drive Flows. The original transmitters provided an output of 10-50 millivolts (mv); while the replacement transmitters provide an output of 4-20 mv. Subsequently, the computer program and the assicuated procedure were modified to accommodate the new transmitters.

On November 6, 1986 the data was retaken and the program re-run successfully. The inspector reviewed the data from this test and noted that new jet pump summer gain adjustment factors and APRM/RBM flow unit gain adjustment factors had been successfully calculated and supplied to I&C for their use.

At the exit meeting the inspector questioned whether the modification which changed out the flow transmitters would require any other procedure or software revisions. The licensee's management committed to review all procedures or software which might have been affected by the change out of these transmitters, and revised them where necessary. This satisfied the inspectors concern.

8.0 Backup Core Thermal Power Evaluation

The licensee uses procedure RE-OTP-002, Core Thermal Power Evaluation, Revision 1, as a backup method for calculating core thermal power in the event that the process computer is unavailable. It involves the use of a detailed heat balance on the nuclear boiler using steady state plant parameters. Under those conditions the nuclear boiler output is obtained as the difference between the total heat removed from the system and the heat added by the flow streams returning to the boiler.

On November 2, 1986 the inspector witnessed the data taking by one of the reactor engineers. This data was recorded in the procedure and then input into an off-line computer program which performed the calculations described in the procedure. The core thermal power calculated by the backup method was 3289 mwth which was in good agreement with the 3290 mwth calculated by the process computer.

9.0 HPCI Pump Performance Verification

The inspector witnessed portions of the HPCI Pump Performance Verification performed on November 7, 1986 by reactor engineering and operations personnel. This test was performed to verify the performance of a newly installed pump impeller. Tests included; constant speed/variable flow, constant flow/variable speed; and a hot quick start. The tests were run by pumping from the Condensate Storage Water Tank (CST) back to the CST. During the witnessing of this test the inspector verified:

- Precautions were followed;
- Prerequisites were signed off;
- Required verifications were signed-off; and
- Personnel were knowledgeable.

The inspector witnessed approximately the first half of the test. No violations were identified.

10.0 Core Thermal Hydraulic Stability Testing

The licensee performed core thermal hydraulic stability tests during the startup to confirm core thermal hydraulic stability of the Cycle 2 reload which includes Exxon 9x9 and GE 8x8 fuel assemblies. Procedure RE-2TP-078, Core Stability Data Acquisition, Revision 0, dated October 9, 1986 was written to establish the procedure for taking the necessary data. Two loop stability testing was performed on November 2, 1986 and was witnessed by NRC consultants from Oak Ridge National Laboratory (ORNL) who took independent measurements. The licensee's corporate engineering staff was still evaluating the results of this test at the time of this inspection. However, a preliminary results evaluation indicates the core decay ratio was approximately .4, which is considered acceptable.

Another test will be performed, at a later date, to determine the core decay ratio for single loop recirculation operation.

The results of these tests will be evaluated by ORNL.

11.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at the entrance interview on November 3, 1986. The findings of the inspection were discussed with licensee representatives at the exit meeting on November 7, 1986.

No written material was provided to the licensee by the inspector.