FINAL REFUELING TEST REPORT

Unit 2 Cycle 9

Dates Performed: 03-25-96 through 4-29-96

Deficiencies: There were 2 Test Deficiencies.

Remarks: This report will be submitted to the NRC per the requirements of TS 6.9.1.1, Startup Report, due to the installation of fuel that has a different design. This cycle will contain the first full reload, (200 bundles), of GE11 fuel assemblies. (Note that a partial reload of 32 GE11 assemblies were loaded in unit 3 cycle 7.) Test results indicate that BFN Unit 2 systems are capable of meeting their design functions and that power operation can be safely and efficiently continued.

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Plant Manager Date

FINAL REFUELING TEST REPORT
UNIT 2 CYCLE 9

POWER HISTORY

Open Vessel Testing, Phase I

- 0-TI-147, Fuel Loading And In-Core Shuffle was started on 03/25/96, with the first fuel move of 1656 planned in core steps being performed at 02:30 on March 26, 1996. Core shuffle was completed at 03:23 on April 14, 1996.

- The Open Vessel portions of 2-TI-299, Control Rod Drive Testing were performed from April 16 through 19, 1996 with no test deficiencies noted.

- The Open Vessel portion of 2-TI-135, Process Computer and Core Performance was performed on April 16, 1996.

Heatup to 55% Power Testing, Phase II

- Following completion of all prerequisites, control rod withdrawal for plant startup commenced at 1635 on April 21, 1996. Initial criticality was achieved at 2030 on April 21, 1996 and the verification of core shutdown margin was performed per 2-SI-4.3.A.1, Reactivity Margin Test.

- Reached 250 psig (0055, 4/22/95) and performed main steam relief valve (MSRV) testing per 2-SI-4.6.D.2 Main Steam Relief Valve Manual Cycle Test (beginning at 0615, 4/22/96) and initial testing of the new digital feedwater control system per PMT - 281 (including placing startup level control in automatic) and testing for front standard replacement of the Reactor Feed Pump Turbines (RFPTs) per PMT 275 (including initial operation for all three RFPTs).

- APRM calibration per 2-TI-136 APRM Calibration was completed at 0749 on 4/22/96. No APRM adjustment was required.

- Plant heatup to rated conditions resumed at 0058 on 4/23/96, the reactor mode switch was taken to RUN at 0508 on the same day.

- APRM calibration per 2-TI-136 was completed at 0750 on 4/23/96, resulting in an adjustment of the APRMs.

After satisfactorily retesting of SRV 2-1-30, for 2-SI-4.6.D.2, the main turbine generator was connected to the grid for the first time at 1236 on 4/23/96. The generator was then removed from the grid at 1247 because of bearing number 3 vibration at 11 mils due to

rubbing. The main generator breaker was reclosed at 1932 and the main turbine manually tripped again at 2022 due to rubbing induced vibration of 11 mils on bearings 3 and 4. The main generator breaker was reclosed at 0244 on April 24, 1996 and re-opened at 0628 to complete main turbine startup activities. Final generator synchronization was achieved at 0754 on 4/24/96.

- Testing per 0-TI-135, Process Computer and Core Performance began at 1945 on April 23, 1996 and was completed satisfactorily for this test plateau at 0846 on April 24, 1996.

- Control rod scram time testing 2-TI-299 Control Rod Drive System Testing After Refueling began at 2012 on 4/24/96, at 38% reactor power. Scram time testing was completed at 2135 on 4/25/96. LPRM hook up verification was completed in conjunction with scram time testing.

- At 0104 on 4/26/96, reactor power was increased to 50% core thermal power for performance of TIP calibrations and 2-TI-131 Feedwater System Testing

- 2-TI-149, Reactor Water Level Measurements, phase 2, was completed at 0252 on 4/26/96.

55% to 100% Power Testing, Phase III

- 2-TI-131 Feedwater System Testing continuing at 57% power on 4/26/96 at 1810 with all three feed pumps in service in for the first time.

- 2-TI-131 Feedwater System Testing level checks begun at 0210 on April 27, 1996 after raising power to 70% and completed at 0325, after confirming minimal feedwater system fluctuations with all three feedpumps in automatic and with feedwater level control in three element. At 0500, reactor power reached 80%.

- While resolving low flow problems with the 2A and 2B feedpumps, it was found that CRD 10-47 would not insert. It was determined that the cooling water check valve was stuck open and after manipulating the HCU valves to free the check valve, CRD 10-47 was inserted without any problems. 2-TI-131 Feedwater System Testing continued at 80% power.

- After reaching 90% power at 1908, 2-TI-131 testing at 90% power was performed and completed at 2040.

- At 0021 on April 28, 1996, began LPRM calibrations per 2-SI-4.1.B-3, 0-TI-135, Process Computer and Core Performance and 2-TI-137, Core Power Distribution. However due to high pressure feedwater heater isolations, testing was stopped at 0230.

- 1015 MWe was achieved at 1450 after a preconditioning ramp up from the target rod pattern and Phase III of 0-TI-135, Process Computer and Core Performance was completed at 1522 on April 28, 1996.

- At 1950, 99% power was reached and the 12 hour soak was begun for performance of 2-SI-4.1.B-3 and 2-TI-137, Core Power Distribution with 2-SI-4.1.B-3 being completed at 1040 on 4/29/96.

- At 1135, on April 29, 1996, 2-TI-82, Drywell Atmosphere Cooling System was completed.

- 2-TI-137, Core Power Distribution was completed at 1256 and 2-TI-174, Recirculation System Flow Calibration was performed, starting at 1349 on 4/29/96.

- After reducing power to 3260 MW_{th}, 2-TI-131 was resumed at 1649 and completed at 1805. This testing completed all of the feedwater system testing per 2-TI-131 and power was returned to 100 % power.

RESULTS

0-TI-147, Fuel Loading And In-Core Shuffle

Fuel shuffle began at 02:35 on March 26. There were several planned pauses during the shuffle to do in-vessel inspections, CRD and control blade replacement. Fuel shuffle continued until completion on April 14 at 03:23 hours. During the in-vessel work two of the fuel support castings removed for control blade replacement would not fully re-seat. New castings were prepared and seated without further difficulty.

The significant delays during fuel loading were as follows.

o	ESF actuation and water clarity	17:40 hrs:min
o	SRM problems	8:20 hrs:min
o	Restore RPIS	6:49 hrs:min

- o Refueling Bridge problems 5:35 hrs:min
- o Correct mis-oriented bundle 1:30 hrs:min

Six Field Changes were used as described below:

1) Returned fuel assembly to SFSP location after SRM spiking.
2) Correct mis-orientation.
3) Re-seat assembly in original core location when cable reel spring failed.
4,5&6) Added additional empty cells for CRD work and replacement of fuel support castings.

0-TI-299 / 2-TI-20, Control Rod Drive System Testing After Refueling

The Phase I, Open Vessel portions of 2-TI-299, Control Rod Drive Testing were performed from April 16 through 19, 1996 with no test deficiencies noted. All control rod drives have been tested for coupling integrity and verified to have full insert and withdraw stroke times between 40 to 60 seconds. The rod position indications were also tested. Twenty eight control rods were missing the "- -" at full in over-travel. Eleven different unique position indication problems were also found, ranging from dimness of a particular digit or a missing LED segment to indication missing at 02, 03, 06, and 07 (rod 10-35 - repaired) and lower left indication in 4 rod display missing "- 0" (34-27 - repaired) Various work orders were submitted to correct these problems and all significant problems have been repaired. The missing " - -" is believed to be due to a switch adjustment problem that is being resolved as the position indication probes are removed for other reasons. The nine missing in the ones digit for rod 02-27 is a cable problem that will be addressed in the future.

Phase II control rod scram time testing per 2-TI-299 Control Rod Drive System Testing After Refueling began at 2012 on 4/24/96, at 38% reactor power. Scram time testing was completed at 2135 on 4/25/96. LPRM hook up verification was completed in conjunction with scram time testing. CRD 34-11 initial scram time was excessive at 5.730 seconds, but was normal at 2.466 seconds when immediately retested and 3.378 when retested after allowing more than the recommended minimum two hour soak time between tests.

0-TI-135 Process Computer and Core Performance:

Phase I was performed during the OPEN VESSEL test plateau on April 16, 1996 at zero reactor pressure with the reactor vessel head in place. The cycle dependent data was successfully installed and verified for Unit 2 Cycle 9. The 3D Monicore system

was successfully initialized. There are no Technical Specification or FSAR acceptance criteria for this portion of the test.

Phase II was performed during the HEATUP to 55% test plateau on April 23-24, 1996 at approximately 18% core thermal power and 36% core flow.

The following were performed during this phase:
1. Verified no changes occurred to the installed BOC case since initial installation.
2. Verified that the control rod position log agreed with Panel 2-9-5 indications.
3. Verified that the LPRM readings log agreed with Panel 2-9-14 indications within 3 units.
4. Restarted 3D Monicore after the turbine generator was placed on-line.
5. Verified that the 3D Monicore core power and flow log and the ICS NSSS heat balance calculation of core thermal power agreed with a manual heat balance within $\pm$ 2%.
6. Verified that the exposure and power distribution logs were reasonable compared to the Cycle Management Report data.
7. Verified that the 3D Monicore calculation of thermal limits for minimum critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR), and maximum linear heat generation rate (LHGR) agreed within $\pm$ 2% of a qualified backup calculation and that MCPR occurred in the same location.

All acceptance criteria were met.

Phase III was performed during the 55 to 100% test plateau on April 28, 1996 at approximately 85% core thermal power and 70% core flow.

The following were performed during this phase:
1. Verified that the LPRM failure status log, LPRM deviations log, LPRM exposure corrections log, and LPRM exposure values log were consistent and reasonable. Test Deficiency number 1 was written because the LPRM remaining life log contained some negative or non-existent values for remaining exposure. These values are not unreasonable given the fact that 3D Monicore has not been configured with the appropriate data to calculate LPRM remaining life. LPRM depletion is tracked in accordance with 0-TI-11. Validation comments were generated to remove review of the LPRM remaining life log from 0-TI-135.
2. Verified that the 3D Monicore core power and flow log and the ICS NSSS heat balance calculation of core thermal power agreed with a manual heat balance within $\pm$ 2%.
3. Verified that the 3D Monicore calculation of thermal limits for minimum critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR), and maximum linear heat generation rate (LHGR) agreed within $\pm$ 2% of a qualified backup calculation and that MCPR occurred in the same location.

All acceptance criteria were met.

Based on these results, the 3D Monicore thermal limits calculations were considered operable and the process computer performance is considered acceptable, indicating that the plant's computer systems are ready to support Unit 2 full power operations.

2-SI-4.3.A.1, Reactivity Margin Test

This test is performed in conjunction with the initial in-sequence critical to demonstrate that the reactor can be made subcritical with a margin of at least 0.38% ΔK/K with the strongest control rod fully withdrawn and at the most reactive time in core life. It also verifies that the actual critical rod configuration is within 1.0% ΔK of the predicted critical rod configuration.

Rod withdrawal for reactor startup and the shutdown margin demonstration began at 16:34 on 4/21/96. The reactor was supercritical with a 168 second period at 20:39 on 4/21/96. Criticality data was collected when rod 38-27 (RWM group 8) was withdrawn to position 08, with a moderator temperature of 195°F. The following results were obtained:

1. The unit 2 cycle 9 shutdown margin was calculated to be 2.0% ΔK/K. This meets the requirements of technical specification 4.3.A.1, which requires a minimum shutdown margin of 0.38% ΔK/K.

2. The difference between the predicted and actual critical rod configuration was determined to be 0.0775% ΔK. This meets the requirements of technical specification section 3.3.D, which requires that the difference between the predicted and actual critical rod patterns be no greater than 1.0% ΔK.

All test acceptance criteria were successfully demonstrated and should be considered fully acceptable in meeting the criteria of technical specifications 4.3.A.1, 3.3.D, and 4.3.D, and FSAR section 13.10.2.2. There were no test deficiencies.

0-TI-136, APRM Calibration

Plant data was collected during reactor heatup and ascension in power. Core thermal power was determined by the bypass valve comparison and the APRMs were adjusted accordingly. No problems or difficulties were encountered. There were no Test Deficiencies written. The first performance of 0-TI-136 was on April 22, 1996 at 5.5% core thermal power. The second performance of 0-TI-136 was on April 23, 1996 at 18.4% core thermal power. Both performances were during the Heatup to 55% power test plateau.

No adjustments to the APRMs were required for the first performance of 0-TI-136; the APRMs were indicating reactor power of 6 to 8% power at a calculated core thermal power of 5.5%. All APRMs were adjusted during the second performance to more accurately indicate reactor power; the APRMs were indicating reactor power of 28 to 33% power at a calculated core thermal power of 18.4%.

The results from each test performance fully met the acceptance criteria listed below.

Level 1 Acceptance Criteria -
At least two APRM's in each RPS channel must be calibrated to read greater than or equal to the actual thermal power.

Level 2 Acceptance Criteria -
If the level 1 criteria is satisfied, then the APRM channels are considered to be reading accurately if they do not read more than 7 percent greater than the actual core thermal power.

2-TI-149, Reactor Water Level Measurements

The purpose of this test is to collect sufficient data during Phase II and III testing to ensure the instrumentation used to monitor reactor water level operated correctly from atmospheric to rated pressure and from 10% to 100% power. This was achieved by recording readings of the control room indications for RPV narrow range compensated, narrow range uncompensated, wide range, fuel zone, and floodup range instruments and by comparing these readings to those of like calibration and to all others based on predicted readings taking into account off-calibration conditions. Also, this procedure utilizes the data collected at the various power plateaus to perform a check of the substitute core flow calculated based on recirculation flow to the measured core flow.

Data for the Test Plateaus were collected at the following conditions:

Reactor Pressure psig	Reactor Power (%)	Core Flow (Mlbm/hr)
38	01.3	29.9
161	01.3	29.1
253	01.5	27.7
382	01.7	27.4
707	01.8	25.6
924	02.5	23.3

924	13.4	34.3
928	19.6	36.4
959	36.7	78.8
966	49.5	63.2
986	76.3	64.7
998	90.1	79.5
1006	98.6	90.4

The results form each test performance fully met the acceptance criteria.

The uncompensated narrow range level instruments agree within 3 inches of one another. During heatup the uncompensated instruments are broken into two groups. The instruments within each group agree within 3 inches of one another within that group.

The compensated narrow range instruments should agree within 3 inches of one another.

The wide range level instruments should agree within 10 inches of one another.

The determined actual levels for all designated instruments should agree within 10 inches after correcting for off-calibration and off-normal conditions.

The value of measured core flow should be within 5% of rated of the calculated value of core flow.

Based on the data collected, the level instrumentation is responding as expected and is adequate for full power operation. In addition, based on the data collected, the substitute core flow correlation used by the Integrated Computer System (ICS) is considered correct and adequate for continued operation.

0-TI-137, Core Power Distribution

This test calculates the total uncertainty associated with the TIP system, checks the core power distribution and gross TIP symmetry, and verifies the proper hookup of the TIP system. The data from these TIP sets is compared statistically using the computer program TI137 to determine the total average TIP uncertainty. In addition, gross TIP symmetry and core power distribution are checked by comparing symmetric traces from the TIP sets and

by examining the normalized full power adjusted TIP readings. The computer program TI137 also calculates the percent difference for each symmetric TIP pair to determine if any asymmetries exist. This test was performed during Phase III (55% to 100% power) of the Power Ascension Test Program. Testing began on April 28, 1996, and was completed on April 29, 1996. TIP sets were run with core thermal power at 98.6% power and at 98.9% power.

Sections 7.4 and 7.5 were initially performed with core thermal power at 98.6% on 4/28/96 (this run was intended to serve as the performance at greater than 55% power). The total average TIP uncertainty for the first TIP run was determined to be 1.537%. Gross symmetry checks were well within the acceptance criteria of 25% for all of the symmetric pairs (the largest percent difference was 3.81%).

Sections 7.6 and 7.7 were performed with the core thermal power at 98.9% on 4/29/96 (this run was intended to serve as the performance at greater than 75% power). The total average TIP uncertainty for this TIP set was determined to be 1.465%. Gross symmetry checks were well within the acceptance criteria of 25% (the largest percent difference was 3.69%).

The average value of the total average TIP uncertainty from the two successful TIP sets was calculated to be 1.501%, well within the acceptance criteria of 9.0%.

2-TI-131, Feedwater Level Control System

The purpose of this Feedwater Level Control System test was to verify the Reactor Feedwater Pumps (RFP) and the Reactor Feedwater Control (RFWC) system were tuned properly to support plant operation. This test was performed in Phases II and III, on April 24 - 29, 1996 with the reactor power ranging from 20% to 99%. The level 1 criteria was satisfied for each RFP response. The RFP responses did not meet all the level 2 criteria stated in the FSAR and Test Deficiency No. 1 was written.

Performance of Feedwater Level Control system tuning consisted of initiating a one pump flow disturbance and observing the response for that RFP. This was done at several power levels and in One Element, (1E) control and Three Element, (3E) control. Each RFP response was observed and the results were as listed below:

Level 1 Acceptance Criteria - The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to feedwater system changes.

All applicable variables were verified to have a decay ratio of less than 1.0.

Level 2 Acceptance Criteria -

1. The decay ratio of an oscillatory control loop mode must be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system changes where the unit is operating above the lower limit setting of the Master Flow Controller.

   All applicable variables were verified to have a decay ratio of less than 0.25.

2. The transient response of each feedwater pump to a 10 percent flow demand input change, as measured by the turbine speed and flow recorder outputs shall be as follows:

   a. Time to 10 percent of demand should be 1.1 seconds and must be less than or equal to 2.2 seconds.

      All three RFP responses were verified to reach 10 percent of demand in less than or equal to 2.2 seconds.

   b. Time from 10 percent to 90 percent of demand should be 1.9 seconds and must be less than or equal to 2.5 seconds.

      RFP A & C responses failed to reach 90 percent from 10 percent in less than or equal to 2.5 seconds. RFP B response was verified to reach 90 percent from 10 percent in 2.5 seconds.

   c. Settling time to within $\pm$ 5 percent of the final value should be 14 seconds.

      All three RFP responses were verified to be within 5 percent of the final value in less than or equal to 14 seconds.

d. Peak overshoot should be equal to or less than 15 percent of demand.

All three RFP responses were verified to have a peak overshoot of less than or equal to 15 percent of demand.

The overall system response and tuning were performed at:
20% power (1E, 1 pump)
35% power (1E & 3E, 2 pumps)
49% power (1E & 3E, 2 pumps)
57% power (1E & 3E, 2 pumps)
71% power (1E & 3E, 2 & 3 pumps)
76% power (1E & 3E, 3 pumps)
90% power (1E & 3E, 3 pumps)
99% power (1E & 3E, 3 pumps)

The overall system response appears to respond better than the old system. In all cases, the water level response decay ratio was less than 0.25. The testing ensures that the Feedwater Level Control system will respond as expected to plant transients. All acceptance criteria were met with the exception of certain RFP response times. The test deficiency has been dispositioned as acceptable. The slow response was consistent with previous test results and was previously evaluated by GE and TVA as acceptable.

<u>2-TI-82, Drywell Temperatures</u>

The purpose of this test is to verify the ability of the drywell (DW) atmosphere cooling system to maintain design temperature conditions in the drywell during reactor power operation. The test was performed on April 29, 1996 with reactor thermal power at 3028 MWth (92%). Bulk Volumetric Average DW temperature was calculated to be 117.4 °F, which is well below the acceptance criteria 150 °F. All acceptance criteria were met and there were no test deficiencies.

2-TI-174, Recirculation Flow Control Calibration

This test is performed with the reactor at or near rated power and flow conditions to demonstrate that the core flow instrumentation is accurately reflecting total jet pump flow. This test makes use of empirically-determined flow coefficients for the four double-tapped jet pumps to allow determination of effective flow coefficients for all twenty jet pumps based on their individual single-tap measured pressure differentials. The effective flow coefficients are then used to calculate all jet pump flows, which are summed to calculate total core flow. This test also calculates Gain Adjustment Factors (GAFs) for the APRM/RBM loop proportional amplifiers to verify that they are accurately correlated to core flow, and calculates jet pump riser and nozzle plugging parameters to allow detection of possible flow obstructions within the jet pump assemblies. The test was performed on April 29, 1996, at 99.8% power and 97.7 Mlb/hr indicated core flow.

The results/acceptance criteria verification for the test performance on 4/29/96 were as follows:

1. On panel 2-9-5, 2-XR-68-50, Total Core Flow (red pen) indicated 97.7 Mlb/hr, and core flow on the NSSS Heat Balance indicated 97.9 Mlb/hr. Both agreed with the calculated core flow of 98.2 Mlb/hr within the required tolerance of ± 3.0 Mlb/hr.

2. The APRM/RBM loop proportional amplifiers 2-FQ-68-5 and 2-FQ-68-81 had calculated GAFs of 1.005 and 1.008, respectively. These GAF values fell inside the acceptable range specified by the procedure (between 0.975 and 1.025).

3. The single tap loop proportional amplifiers 2-FM-68-45 and 2-FM-68-47 had calculated GAFs of 0.982 and 1.014, respectively. These GAFs were inside of the acceptable range specified by the procedure.(between 0.975 and 1.025).

4. Jet pump loop flow variation is the fractional difference between the calculated loop flow derived from summing all ten jet pump flows and that flow that would be calculated by extrapolating from the two calibrated jet pump flows only. It is essentially a measure of how representative the calibrated jet pumps are of all jet pumps. Jet pump loop flow variation for loops A and B were 1.41% and 1.14%, respectively. This is less than the maximum allowable variation of 3.0% specified by the procedure.

5. The value of the jet pump riser plugging parameter calculated for the ten risers ranged from 0.09% to 2.6%. These are all less than the maximum expected value specified in the procedure as 10%.

6. The value of the jet pump nozzle plugging parameter calculated for the ten jet pump pairs ranged from 0.0% to 13.7%. Pairs 1 & 2, 11 & 12, and 19 & 20 were greater than the maximum expected value of 10% specified in the procedure. This may be indicative of some amount of jet pump nozzle plugging. The Recirc System Engineer was notified to evaluate these jet pump pairs as required by TI-174

2-TI-130, Main Steam Pressure Control was not performed during this startup as no major maintenance or design changes were performed that could have significantly effected the performance of the pressure control system.

2-TI-132 Reactor Recirculation System was not performed during this startup as no major maintenance or design changes were performed that could have significantly effected the performance of the reactor recirculation system.

CONCLUSION

These test results demonstrate that BFN Unit 2 systems are capable of meeting their design functions and that power operation can be safely and efficiently continued.