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Subject: Arkansas Nuclear One-Unit 2
Docket No. 50-368
License No. NPF-6
Post-Steam Generator Replacement Startup Testing Summary Report

Gentlemen:

During Refueling Outage 2R14, Arkansas Nuclear One-Unit 2 (ANO-2) replaced the original steam generators with Westinghouse Model Delta 109 steam generators.

ANO-2 Technical Specification 6.9.1.1 requires submittal of a summary report of plant startup and power escalation testing following (a) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (b) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

The attached report summarizes the relevant testing conducted during startup and power escalation to demonstrate satisfactory plant performance with the new steam generators. Detailed test data is located on site.

This submittal contains no regulatory commitments.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

JDV/dwb
Attachment

IE26

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ARKANSAS NUCLEAR ONE UNIT 2
REFUELING OUTAGE 2R14 STARTUP REPORT

1. Containment Cooling System

The objective of this test was to verify proper operation of the containment cooling system after installation of new chilled water cooling coils and adjustment of fan blade pitch from 33 to 27 degrees.

Pre-modification and post-modification measurements of system airflows, wet & dry bulb temperatures, static pressures, chilled water flow, motor power, amps, voltage, power factor and fan vibration were taken in the normal and emergency configurations. The data was used to verify that each motor does not exceed the acceptance criteria of 78 horsepower for design basis accident conditions and that system performance is sufficient for normal and emergency conditions.

Evaluation of the test data verified that the containment cooling system was successfully modified to allow maximum airflow within acceptable motor horsepower limits.

2. Spillover Test

The purpose of this test was to determine at what levels the reactor coolant system (RCS) cold and hot legs spill over into the steam generator channel heads in order to verify adequate RCS level to support shutdown cooling system operation without danger of vortexing.

The acceptance criterion for this test was a minimum level of 27 inches. This value is conservative since the actual required level is 19 inches.

Spillover levels were determined to be 30 inches for the cold legs and 31 inches for the hot legs of both steam generators.

3. Structural Integrity Test

The steam generator replacement resulted in an increase in containment peak accident pressure (P_a) from 54 psig to 58 psig. This required an increase in containment design pressure from 54 to 59 psig. The Structural Integrity Test (SIT) was conducted with a peak pressure of 68 psig.

The SIT was performed in conjunction with the Integrated Leak Rate Test (ILRT) to reduce the number of pressure transients on the building. This resulted in an extended period at the peak pressure while the ILRT was being performed.

The structural response to pressure was monitored at closely spaced intervals while the containment building was pressurized. During building pressurization, pressure was maintained at 62.1 psig (nominal) to allow time for concrete surface examinations. It was then held essentially constant at 68 psig for about 19 hours to allow time for concrete surface examinations and the ILRT. Concrete surface examinations were again conducted at 62.1 psig during the depressurization phase of the test.

The results of the SIT were satisfactory. The recorded data confirmed that the structure responds elastically to SIT pressure and demonstrated that yielding of conventional reinforcing steel did not occur during the test. The SIT provided direct evidence that the containment responds to pressure loading in an acceptable manner and that it has the capacity to retain the design basis accident pressure with at least the prescribed margin of safety.

4. Integrated Leak Rate Test

Replacement of the steam generators required a construction access opening to be cut into the Unit 2 Containment Building to allow movement of the equipment in and out of the building. Upon completion of the steam generator replacement and repair of the construction access opening, an Integrated Leakage Rate Test (ILRT) was performed to the requirements of 10CFR50, Appendix J, Option B.

The ILRT was performed in conjunction with a Structural Integrity Test (SIT). The test conformed to the requirements of 10CFR50, Appendix J, Option B, except that it was conducted at the SIT pressure of 68 psig (1.15 times design pressure, 59 psig) rather than the required ILRT accident pressure of 58 psig . Justification for this deviation from the 10CFR50 test pressure requirement was approved by the NRC in a letter dated October 12, 2000 (2CNA100002). Performing the ILRT in conjunction with the SIT and at SIT pressure reduced the number of pressure transients on the containment building.

Examinations and inspections were performed in conjunction with the SIT and satisfied the requirements of 10CFR50 Appendix J and the ANO Containment Leak Rate Testing Program.

No unusual occurrences were observed or reported during the ILRT. The instruments functioned throughout the test with no indication of erratic or erroneous readings.

5. Feedwater Control System/Reactor Regulating System Power Ascension Testing

The purpose of this testing was to verify proper operation of the feedwater control system and the reactor regulating system with the replacement steam generators.

The following test methods were employed during unit power ascension, and at steady state power conditions:

- Open loop feedwater flow tests to verify the control system software setpoints and tuning parameters along with verification of component response using actual plant response characteristics.

  This test was initially performed at approximately 15% power and again at approximately 60% power. The results of these tests were as expected with no tuning adjustments required.

- Closed loop perturbation tests to verify control system response to a narrow range level setpoint step.

  The objective of these tests was to demonstrate the ability of the feedwater control system to respond to a mismatch between steam generator level and setpoint.

  The acceptance criteria were that the actual steam generator levels remain within specified limits of the programmed values, and that steam generator levels automatically returned to, and remained within, design limits of the level setpoint following a level setpoint change.

  This test was performed at reactor power levels of approximately 15%, 60%, and 96% power. The 96% power test was performed a second time following an adjustment to the main feedwater pump demand curve. For each steam generator, a -5% level setpoint change was initiated and response of the level control system was monitored. This was followed by a +5% level setpoint change and response of the level control system was monitored.

  All acceptance criteria were met for each test with no additional tuning adjustments required after the initial pump demand curve adjustment.

- Monitoring of normal plant power ascension operations to verify proper control system performance.

  The feedwater control system performed adequately during normal power ascension and steady state operations with no tuning adjustments required. All acceptance criteria were met.

6. Steam Generator Blowdown System

The objective of this test was to verify proper operation of the steam generator blowdown system following replacement of the steam generators and the blowdown heat exchangers.

Testing was conducted in various operating modes of the blowdown system including steam generator wet layup, steam generator draining, hot standby, and power operation. Plant instrumentation and temporarily installed test equipment was used to measure system temperatures, pressures, and flows. System piping was also visually monitored for thermal expansion and vibration.

The steam generator blowdown system performed acceptably during all operating modes. There was no unacceptable vibration or thermal expansion identified during system walkdowns.

7. Inside Containment Thermal Measurements/Walkdowns

The objective of this test was to collect thermal expansion measurements and perform interference checks of the reactor coolant system, main steam, main feedwater, blowdown, and other systems at various temperature plateaus during startup following steam generator replacement.

Thermal expansion measurements included system piping, component and hanger displacements, and component support and pipe whip restraint gap settings. Measurements and visual inspections to verify acceptable clearances were conducted with reactor coolant system temperature at 185, 400, and 545 degrees.

All identified deficiencies were either corrected or evaluated as acceptable during plant heatup.

8. Vibration measurements Inside Containment

The objective of this test was to collect vibration data for the reactor coolant, main steam, main feedwater, and emergency feedwater systems during startup and power ascension following steam generator replacement to verify that piping vibration is within acceptable limits.

Reactor coolant system vibration data was collected using permanently installed instrumentation while main steam, main feedwater, and emergency feedwater data was collected using temporarily installed triaxial accelerometers. Data was collected from plant heatup to 100 percent power.

Vibration data for each of the monitored systems was evaluated by Engineering personnel and verified to be within acceptable limits.

9. Feedwater Stratification Measurements

The purpose of this test was to collect temperature data for a main feedwater nozzle during low flow conditions to verify that the design of the replacement steam generators precludes temperature stratification at low flow conditions.

Seven thermocouples were installed around the circumference of a main feedwater nozzle to collect data during low flow conditions while the steam generators were being fed by either the auxiliary feedwater pump or the motor-driven emergency feedwater pump.

The data collected during startup verified that the design of the replacement steam generators precludes temperature stratification during low flow conditions.