U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS. 50-286/93-28

LICENSE NO. DPR-64

FACILITY NAME: Indian Point 3 Nuclear Power Plant

INSPECTION AT: Buchanan, NY

INSPECTION DATES: November 15 through November 19, 1993

INSPECTORS: P. Patnaik for Charles K. Battige, Reactor Engineer — 12-27-93 Date
Materials Section, EB, DRS

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APPROVED BY: Michael C. Modes, Chief — 12/28/93 Date
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Areas Inspected: Review of erosion/corrosion monitoring program and the steam generator leakage monitoring program.

Results: Licensee's erosion/corrosion monitoring program covering the selection of components for analysis and the evaluation of data was found to be satisfactory. The root cause evaluation of severe wall thinning of the 18-inch diameter branch off the 28-inch diameter main run in the fifth extraction steam line was plausible.

The steam generator leakage monitoring program has some weakness. The governing procedure does not specify the action required in the event of step increases in primary to secondary leak rate, and there was no administrative limit to initiate power reduction or shutdown. A review of the steam generator tube rupture emergency operating procedure raised concerns about the operability of the Unit 1 processing system and the structural integrity of the transfer system and the processing tank of Unit 1.

DETAILS

1.0 LONG-TERM EROSION/CORROSION (E/C) MONITORING (49001)

1.1 Scope

The monitoring of erosion/corrosion in susceptible high energy piping is important to maintain structural integrity of piping and components. During this inspection, the licensee's program covering the selection of components for examination, the analysis of systems, the ultrasonic thickness data, and their evaluations were reviewed. Also, the licensee's evaluation of root cause of severe wall thinning of the 18-inch branch off the 28-inch main run in the fifth extraction steam line and the corrective action was reviewed.

1.2 Findings

New York Power Authority, the licensee for Indian Point 3, selected approximately two hundred components for examination of wall thinning based on predictive analysis, industry experience, plant specific experience, and the past ultrasonic data. The components were selected from the following systems:

- Extraction Steam
- Condensate
- Preseparator Drain
- Feedwater
- Heater Drains
- Reheater Drains
- Moisture Separator Drains
- Feedwater Recirculation

The licensee performed the predictive analysis using the EPRI CHECMATE computer model in order to prioritize inspection locations. During this inspection, the inspector reviewed the analysis for the extraction steam and the condensate systems. The design basis and the design data input were verified to be appropriate and were translated correctly to conduct this analysis. The analysis identified specific locations where erosion/corrosion damage may occur. Although, in some cases, the analysis overpredicted the wear on some components, and it prioritized areas of excessive wall thinning for examination.

The licensee's "Procedure for Structural Evaluation of Erosion/Corrosion Thinning in Carbon Steel Piping" was reviewed. The procedure outlined criteria to calculate minimum wall thickness, evaluate local thinning, and provided approximate estimate of remaining service life of a thinned component so that the next inspection of the thinning area can be scheduled in a timely fashion.

A sample of E/C data sheets for the condensate and the extraction steam systems were reviewed to ensure compliance with the above procedure for structural evaluation. The evaluation of wall thinning due to erosion/corrosion was found to be conservative.

The root cause analysis of the severely eroded tee, found by the licensee in 35 extraction steam lines, was reviewed. During an erosion/corrosion examination, the licensee's personnel discovered an unreinforced branch connection in the extraction steam line. Subsequent inspections revealed that all branch connections in the extraction steam system were unreinforced. In preparation for replacement of the unreinforced branch connection in the fifth extraction steam line upstream of the 35C feedwater heater, an ultrasonic examination of the component to determine a suitable cut location, revealed severe wall thinning of the branch line in the vicinity of the main run. A visual examination of the interior of the tee revealed improper fitup of the joint between the branch line and the main run creating a lip over one-third of the circumference. The protrusion was approximately 3/8" into the pipe which created flow disturbances. These flow disturbances caused accelerated thinning at the region downstream of the lip inside the branch line.

The licensee examined a total of forty components (27 tees and 13 elbows) for wall thinning in the extraction steam, feedwater, and condensate and heater drain systems. As a result of the examination, three components (2 tees and 1 elbow) were identified to have wall thinning requiring replacement and two components required weld repair. In addition, the licensee has initiated replacement of sixteen components (tees) that were found to be unreinforced at the branch line. The remaining components were acceptable.

The licensee made efforts to model the improper pipe-to-pipe branch fitup in the CHECMATE program and came to the conclusion that the program was unable to model the situation.

The inspector concurred with the licensee's analysis for the root cause of severe wall thinning of the branch line in the fifth extraction steam line upstream of the 35 C feedwater heater and the corrective actions taken by the licensee following the situation.

2.0 STEAM GENERATOR MANWAY BOLTING

The Westinghouse letter: INT-89-707, dated March 17, 1989, indicated that low strength bolting had been inadvertently installed in steam generator manways at some Westinghouse plants. The suspect bolts, which did not meet the tensile or chemical requirements for American Society of Mechanical Engineers SA-193, Grade B7, bolting materials, were installed on steam generators shipped from Westinghouse during the 1970-71 timeframe. The Westinghouse letter advised that bolts with a Brinell hardness greater than 250 at approximately 250°F were of the required strength.

The licensee tested eight bolts from each of the three manways of the steam generator #31. The Surveillance Report 3-45, dated November 15, 1993, indicated that the Brinell hardness readings were between 292 and 329. Since the actual test temperature (66°F) was lower than recommended, an engineering evaluation of the test result was planned.

The surveillance report also indicated that bolts from other manways would be tested as they are removed.

3.0 STEAM GENERATOR LEAKAGE MONITORING PROGRAM

The inspectors reviewed the New York Power Authority's procedures for primary-to-secondary leak rate calculations, leak rate monitoring, steam generator tube rupture (SGTR) events, and post-SGTR cooldown. Procedure RE-CA-110, "Determination of Rate of Leakage of Primary Coolant from the Reactor Coolant System to the Secondary Side of the Steam Generator," Rev. 4, contains a preferred and an alternate method of calculating leak rates, setpoint calculation for the condenser air ejector radiation monitor, release rate, and quarterly release rate limit. The inspector identified an error in the prescribed calculation for the maximum alarm setpoint for radiation monitor R-15 for the condensate air ejector. During the course of the inspection, the licensee issued a temporary procedure change to correct the setpoint formula and clarify that the setpoint will be set at or below the maximum technical specification leakage from any steam generator (0.3 gallons per minute).

The inspectors found that Procedure ONOP-SG-1, "Steam Generator Tube Leak," Rev. 9, did not specify an administrative limit for leakage to initiate power reduction or shutdown in the event of a primary-to-secondary leak. The procedure also did not specify a sampling frequency in the event of increasing leakage nor a specific isotope to be used for calculations of the leak rate. The inspectors stated that some of the recent tube rupture events experienced by the nuclear industry have occurred prior to reaching the technical specification limit of 500 gallons per day (gpd). Hence, it is prudent to specify management actions and frequency of sampling of condensate air ejector activity in the event of a step increase in leak rate below the administrative limit. (The program used by the Chemistry Department for performing calculations specifies that the chemist should contact Chemistry management if the leak rate is in excess of 0.02 gpm or the rate increases more than 25% per day). The licensee stated that these procedural deficiencies will be resolved prior to restart of the plant.

3.1 Steam Generator Tube Rupture Emergency Operating Procedures (URI 286/93028-01)

NYPA Procedure E-3, "Steam Generator Tube Rupture (SGTR)," Rev. 7, provides instructions to operators to terminate leakage from the primary (reactor coolant) system into the secondary system after a SGTR. The final step of the procedure, Step 42, directs operators to "Go to appropriate post-SGTR cooldown method as directed by the shift SRO or shift supervisor." The three methods listed employ, the backfill (ES-3.1), the blowdown

(ES-3.2), or the steam dump (ES-3.3) systems. Step 10 of the blowdown procedure directs operators to go to alternate post-SGTR cooldown methods (backfill or steam dumps) if blowdown is not available; thus, no preference is given to the operators for cooldown of the plant following a SGTR. Procedure ES-3.2, "Post-SGTR Cooldown Using Blowdown," Rev. 6, indicates that if the blowdown activity exceeds 3 x 10E-5 μCi/cc, all blowdown should be directed to the Unit 1 Secondary Boiler Blowdown Purification System (via 3" Line #915). Step 10 of ES-3.2 diverts the blowdown to Unit 1 for processing in accordance with SOP-SG-1.

A separate utility, Consolidated Edison (Con Ed), owns and maintains Indian Point Unit 1, which was licensed in 1962 and shutdown in 1974. A Memorandum of Understanding, dated February 15, 1980, between Con Ed and NYPA, stipulates that the Unit 1 system be maintained in working order and ready to receive discharges from the Unit 2 or the Unit 3 blowdown systems. NYPA indicated, however, that no surveillances of the Unit 1 system have been performed as yet, and that it had not been used in recent corporate memory. During the course of the inspection, NYPA and Con Ed personnel performed a walkdown of the visible parts of the system and found them in order, although much of the system connection between Units 1 and 3 is contained in an underground guard pipe.

The inspectors questioned the structural integrity of the transfer line and the processing tank, the operability of system valves, and the ability of the Unit 1 system to effectively process blowdown from all four steam generators of Unit 3 in the event of a SGTR, and the ability of the existing blowdown system to effectively cool down the plant following a SGTR. Lack of structural integrity resulting in system failure could potentially result in an unmonitored release of contaminated water following a SGTR. This is an unresolved item of this inspection (URI 286/93028-01).

4.0 EXIT MEETING

The findings of the inspection were presented to and discussed with members of the licensee's management at the exit meeting on November 19, 1993. The licensee concurred with the findings of the inspection and did not voice any objections. A list of attendees of the exit meeting on November 19, 1993, is attached to this report as Attachment 1.

ATTACHMENT 1

Exit Meeting

New York Power Authority

R. J. Dolansky	E/C Engineer
J. Garrity	RM-IP3
W. A. Josiger	Vice President-Nuclear Operations
M. Kerns	Sr. Chemical Engineer, IP3
P. Kokolakis	Director, PWR
S. Munoz	Director, PWR Projects Engineer
D. O'Brien	Technical Licensing Coordinator
R. Penny	Supervisory Maintenance Engineer

U. S. Nuclear Regulatory Commission

K. Battige	Reactor Engineer Intern
P. Patnaik	Reactor Engineer
B. Westreich	Resident Inspector