50-247

Union of Concerned Scientists

June 29, 2000

Mr. Leonard A. Wiens, Petition Manager
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: INDIAN POINT 2 STEAM GENERATOR TUBE INTEGRITY AND EMERGENCY PLANNING

Dear Mr. Wiens:

On June 2, 2000, Consolidated Edison submitted a lengthy report to the Nuclear Regulatory Commission on the steam generators at Indian Point 2. Based on this report, Con Ed is seeking the NRC's permission to restart the plant without replacing the steam generators. The company contends that it can safely operate the plant with the existing steam generators for 310 Effective Full Power Days.[1] Based upon our review of Con Ed's report, the petitioners conclude that Con Ed's claim is not justified by the available data. In fact, the report reinforces our conviction that safety—not to mention the NRC's regulations—dictates that the steam generators at Indian Point 2 be replaced before the plant resumes operation. Additionally, the petitioners contend that Indian Point 2 will not in compliance with federal regulations unless Con Ed performs a full-participation[2] exercise of the emergency response plan prior to restart. Therefore, we conclude that the NRC should not permit Con Ed to restart the plant until after a full-participation exercise has been successfully completed.

The approach taken by Con Ed and its consultants to evaluate steam generator tube integrity is sound in theory, but falls apart in practice. The approach was to determine how long the plant could operate with the existing steam generators by using the tube inspection data collected in 1997 and 2000 to calculate crack growth rates. These crack growth rates were then used along with the crack data from the 2000 inspections to calculate how long it would take for the worst cracked tube to reach the limiting crack size.[3] The following hypothetical example illustrates the approach taken by Con Ed:

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[1] Executive Summary, page 1, first paragraph

[2] According to Appendix E to 10 CFR Part 50:

> "Full participation" when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite local and State authorities and licensee personnel physically and actively take part in testing their integrated capability to adequately assess and respond to an accident at a commercial nuclear power plant. "Full participation" includes testing major observable portions of the onsite and offsite emergency plans and mobilization of state, local and licensee personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.

[3] Executive Summary, page 6, second paragraph

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Example: Tube XYZ showed 21.7% cracking during the 1997 inspections
Tube XYZ showed 33.0% cracking during the 2000 inspections
Indian Point 2 operated for 500 effective full power days (EFPDs) between inspections
Limiting crack size is 40% through-wall

$$\frac{\text{(2000 data - 1997 data)}}{\text{operating period}} = \text{crack growth rate}$$

$$\frac{\text{(33.0\% - 21.7\%)}}{\text{500 EFPDs}} = \text{0.0226\% crack growth per effective full power day}$$

$$\frac{\text{(Limit - 2000 data)}}{\text{crack growth rate}} = \text{remaining margin}$$

$$\frac{\text{(40.0\% - 33.0\%)}}{\text{0.0226\% / EFPD}} = \text{310 effective full power days}$$

Con Ed essentially performed this set of calculations for all the remaining in-service steam generator tubes and took the lowest remaining margin to define how long the plant could operate until the next inspection.

The reason that this approach is fundamentally flawed is that each number, with the sole exception of the limiting crack size value (40% in the above example) is not known with much precision. Consequently, the valves are so uncertain that the mathematical process used to derive the remaining margins amounts to little more than guessing.

Data from the 1997 inspections were used as one end point in the crack growth rate calculation. However, the 1997 data are questionable. Westinghouse reported:

> *A retrospective review of the 1997 inspection data showed that the indication* [of the tube that ruptured in February 2000] *was masked by signal distortion (noise) due to deposits on the tube including copper and geometry effects due to tube ovality*[4]
>
> *It is seem from the table* [Table 3.1] *that the bad data percentage was 53% in the low row U-bend tubes exmained for the midrange probe and 0.8% for the high frequency probe. ... As a general rule, the midrange probe data quality did not improve significantly from the lower rows (row 2) to the higher rows (row 4).*[5]
>
> *Only 400 kHz +Point* [i.e., midrange probe] *data are available from the 1997 inspection for development of axial PWSCC growth rates for U-bend indications.*[6]

Con Ed stated during its May 3, 2000, presentation to the NRC staff that noise levels during the 1997

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[4] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, page 2-2, paragraph 2.

[5] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, page 3-4, Section 3.3.

[6] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, page 3-6, paragraph 3.

tube inspections prevented actual crack indications from being detected. To remedy this problem, a different probe was used to inspect tubes in 2000 that reduced the signal-to-noise ratio, unmasking the crack indications. But the new probe cannot re-create the tube conditions of 1997. Therefore, the 1997 tube inspection data are imprecise and of questionable value in crack growth rate calculations, particularly for the tubes in the low rows. Con Ed reportedly plugged all of the tubes in Row 2,[7] but the 1997 inspection data quality problem affected the tubes in Rows 3 and 4.[8]

Data from the 2000 inspections were used as the other end point in the crack growth rate calculation. However, the 2000 data are also questionable. Westinghouse provided a chart (attached) showing the probability of the new probe detecting a crack.[9] The new probe has a 50% chance (same as getting heads when flipping a quarter) of finding a 40% through-wall crack caused by axial denting.

In other words, the new probe could inspect a tube and find no crack indications and re-inspect that same tube and find a 40% through-wall crack indication. Clearly, if the probe's ability to find a serious crack is that uncertain, the results from the probe must also be questioned. For example, if the probe inspected Tube XYZ in 2000 and indicated a 33% through-wall crack, could the actual crack size be 28%, 38%, 45% or more? Therefore, the uncertain 2000 tube inspection data are of questionable value in crack growth rate calculations.

The remaining margin calculations are inherently non-conservative due to Con Ed's use of effective full power days instead of either critical days or hot power days. Effective full power days are essentially calendar days multiplied by the plant's capacity factor. In other words, if the plant operates for 365 days at a capacity factor of 85%, it will operate for 310.25 effective full power days.

But the plant can achieve an 85% capacity factor various ways. It could operate at 100% power for 310.25 days and be shut down the remainder of the year. Or it could operate at an average power level of 85% for the entire year. The difference is important with respect to steam generator tube cracking. Westinghouse reported:

> *Operating temperature is an important factor in the initiation of PWSCC* [primary water stress corrosion cracking]. ... *The same temperature dependent trend is expected to apply for initiation of cracking in row 3 U-bends.* [10]

By using effective full power days in its calculation of remaining margins, Con Ed is clearly non-conservative. Con Ed claimed that it is safe to restart Indian Point 2 and operate for 310 effective full power days (EFPDs) before the next inspection. As we indicated earlier, 310 EFPDs could be reached by running the plant at 100% power continuously for 310 days or by operating the plant for 365 days at 85% power. The crack growth rate is more dependent on operating temperature and pressure than upon reactor power level. The operating temperature and pressure at 85% reactor power is virtually equal to the conditions at 100% power. Thus, the actual crack growth rate will be approximately the same. But the

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[7] Executive Summary, page 2, pargraph 2.

[8] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, page 3-9,Table 3-1.

[9] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, Figure 5-2, page 5-19.

[10] Westinghouse Electric Company, "Indian Point-2 U-Bend PWSCC Cycle 14 Condition Monitoring and Cycle 15 Operational Assessments," SG-00-05-008, May 30, 2000, page 7-3, section 7.2.2, "Effect of Operating Temperature."

plant will operate for 55 more days in the second case than in the first case—or 55 days beyond the point at which the remaining margin is exhausted. The right thing to do would be to define the remaining margin in terms of critical days (i.e., number of days that the reactor is critical) or hot power days (i.e., number of days that the reactor is at rated temperature and pressure).

In summary, the end points used by Con Ed to calculate crack growth rates are so unreliable that the calculation results are rendered meaningless. Even if the results were valid, Con Ed's use of effective full power days to define remaining margins is non-conservative with respect to safety. Consequently, we reiterate the conclusion expressed in our 2.206 petition to the NRC—the steam generators at Indian Point 2 should be replaced before the plant restarts.

Full-Participation Emergency Response Exercise

Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," requires:

> Each licensee at each site shall conduct an exercise of its onsite emergency plan every 2 years. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section. In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities.

The last full-participation exercise at Indian Point 2 was conducted on June 24, 1998. According to statements by FEMA representatives at the NRC public meeting in Cortlandt, New York, on June 25, 2000, the next full-participation exercise at Indian Point 2 is scheduled for 2002. That's a frequency of four years instead of the two years required by federal regulations.

Representatives of both FEMA and the NRC attempted to rationalize the 4-year frequency during the June 25th meeting on the basis of a full-participation exercise scheduled for November 2000 at Indian Point 3. But Indian Point 3 is owned and operated by a different licensee. The licensee personnel involved in the exercise and interfacing with federal, state, and local officials will be different from those personnel responsible for the Indian Point 2 emergency response plan. Thus, federal regulations dictate that a full-participation exercise at Indian Point 2 this year, not in the year 2002. The petitioners seek nothing less than compliance with this federal regulation.

On a related matter, Chairman Meserve responded to concerns raised by Congressman Maurice D. Hinchey with a letter dated June 13, 2000. In that response, the Chairman stated:

> In order to ensure that the emergency plans are adequate, evaluated exercises are conducted every 2 years to test the integrated capability of the onsite and offsite emergency response organizations to assess and respond to radiological emergency at a nuclear power plant.

We are providing copies of this letter to Congressman Hinchey and the NRC Inspector General. Last year, the NRC Inspector General found that the NRC Chairman had provided misleading information in a response to Congressman Markey. It would appear that this problem has been repeated—unless, of course, it is the NRC staff's intent to enforce Appendix E to 10 CFR Part 50 at Indian Point 2 this year.

The undersigned submits this letter on behalf of all of the petitioners: Nuclear Information and Resource Service, PACE Law School Energy Project, and Public Citizen's Critical Mass Energy Project.

Sincerely,

David A. Lochbaum
Nuclear Safety Engineer

Attachment: Figure 5-2 from Westinghouse report dated May 30, 2000

cc: The Honorable Ben Gilman
The Honorable Maurice D. Hinchey
The Honorable Sue Kelly
The Honorable Nita Lowey
The Honorable Edward Markey
Chairman Richard A. Meserve
Hubert Bell, Inspector General
Jack Strosnider
Hubert J. Miller
Paul Gunter
Jim Riccio
Ed Smeloff