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From: Wayne Schmidt
To: "tees@[redacted], David Lew, Gregory Cranston
Date: Wed, Jul 19, 2000 3:45 PM
Subject: SDP Briefing package

I Burner

EX 6

please see the attached and comment ASAP - so that we may get it to Rick Urban tomorrow. Greg and I will handle any changes tomorrow.

P/14

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~~PREDECISIONAL INSPECTION INFORMATION~~
~~INTERNAL NRC STAFF USE ONLY~~

Indian Point 2 Steam Generator Special Inspection Summary

Prepared by: Wayne Schmidt - Team Leader - Region I - 610-337-5315

Attachments:

1 - NRC Preliminary CDF/LERF assessment
2. - Con Edison - Calculation of CDF and assessment of no LERF.

I. Inspection Scope:

The NRC conducted a special team inspection to review the causes of the failure of a steam generator tube on February 15, 2000. The NRC team members included personnel from the Office of Nuclear Reactor Regulation and Region I, and NRC-contracted specialists in steam generator eddy current testing. The purposes of the special inspection were to determine the adequacy of Con Edison's performance during the 1997 steam generator inspections and to assess Con Edison's root cause evaluation, date April 14, 2000. The team also reviewed portions of the June 2, 2000, Con Edison steam generator condition monitoring and operational assessment report (CMOA) for possible regulatory issues.

II. Conclusion/Root Cause:

Con Edison returned Indian Point, Unit 2, to service in 1997 in a condition that deteriorated with time to the point that a steam generator tube failure occurred within approximately 23 months of operation

A failure to identify significant performance issues during the 1997 steam generator inspection resulted from Con Edison's weak technical management and oversight of the steam generator inspection program. Of most significance, Con Edison failed to identify: inside diameter (ID) primary water stress corrosion cracking (PWSCC) in six small radius U-bend SG tubes, including tube R2C5 in SG 24, which failed in February 2000. Con Edison failed to identify several factors that caused significant limitations and uncertainties in data collection and analyses, this increased the likelihood that steam generator tubes with detectable flaws would have been left in service. Specifically, Con Edison did not evaluate and take necessary actions to compensate for equipment and technique challenges to flaw detection or to consolidate steam generator condition information to assess the significance of the new ID PWSCC degradation mechanism. Overall, Con Edison did not ensure an adequate, integrated technical understanding of the steam generator conditions.

III. Performance Issues;

1. Based on a independent NRC review of the eight U-bend PWSCC indications detected during the 2000 inspection, the NRC determined that six should have been identified in 1997. This included SG 24, R2C5, the tube that leaked on February 15, 2000. During the 1997 steam generator inspection Con Edison did not adequately respond to issues that decreased the probability of detection of small radius U-bend tube indications and increased the likelihood of apex flaws in the small radius U-bend steam generator tubes.

    1. Con Edison did not adequately evaluate poor quality data (low signal to noise ratios) that was encountered during the eddy current inspections in 1997. Con Edison failed to evaluate the effect on the probability of detection of small radius U-bend tube indications.

    2. Con Edison did not adequately responded to a PWSCC indication in the U-bend area of tube R2C67 in SG 24, which was identified during the 1997 outage. This indication, which was located in the apex of this small diameter tube, was a new and significant degradation mechanism at Indian Point 2. Apex cracking is more likely to burst than other u-bend cracks. After identifying an apex U-bend PWSCC flaw in SG 24 tube R2C67, Con Edison took no actions to determine the root cause and took on actions to ensure that this new mechanism understood.

    3. Con Edison did not sufficiently assess eddy current probe restrictions in the upper support plate encountered during the 1997 steam generator inspections, with respect to the potential for flow slot hourglassing. Con Edison did not evaluate the potential for increased apex stresses and PWSCC.

2. Con Edison did not properly set-up the U-bend plus-point eddy current probe in 1997, which negatively affected the probability of detection of U-bend indications. The probe was not set-up with the required calibration standard or with the phase rotation required by the EPRI qualified technique sheet.

3. Con Edison did not have an accurate method of measuring nor some criteria for determining when significant hourglassing of the upper tube support plates had taken place. As such Con Edison could not conduct and submit an evaluation of how the hourglassing affected the long term integrity of the small radius U-bends tubes beyond row 1.

4. The team also concluded that Con Edison's root cause analysis for the event, dated June 14, 2000, did not adequately address their failure to identify deficiencies and limitations related to the 1997 inspection of the low radius U-bend regions. While the root cause analysis attributed the tube failure to a flaw that was obscured by eddy

current signal noise, it did not identify nor address inadequacies in the management of the 1997 steam generator inspection.

IV Risk and Significance Assessment:

NRC Assessment:

During the February 15, 2000, event the leakage from the apex crack in SG 24 tube R2C5 did not reach the full steam generator tube rupture (SGTR) flowrate, due to remaining crack ligaments in the flaw area. However, if additional stress had been placed on the flaw by any larger than normal differential pressure the SGTR leakrate could have been reached. Therefore the risk analysis was done assuming an SGTR. The risk associated with the condition of the tubes during Cycle 14 comes from several potential accident sequences:

1. Spontaneous rupture of a tube, not successfully mitigated by plant operators, causing core damage and bypass of the containment by large radioactive releases.

2. Rupture of one or more tubes induced by a steam system depressurization event, not successfully mitigated by plant operators, causing core damage and bypass of the containment by large radioactive releases.

3. Rupture of one or more tubes induced by a reactor system over-pressurization event, causing core damage and bypass of the containment by large radioactive releases.

4 A core damage event that occurs with the reactor system at normal operating pressure, inducing tube rupture by increasing tube temperature and/or tube differential pressure, causing bypass of the containment by large radioactive releases.

Of these, the first two increase both the core damage frequency (CDF) and the frequency of large radioactive releases bypassing the containment and reaching the environment (hereafter assumed to be a "large early release"). The latter two sequences are already included in the plant's core damage frequency estimate, but would not normally be included in its large early release frequency (LERF). The induced tube ruptures cause them to make contributions to LERF.

The NRC staff estimated the sum of these tube degradation related risk contributions to get a yearly incremental CDF/LERF for an SGTR of approximately 1 $X10^{-4}$/reactor year (RY). Using the single SGTR over a 23 month period established a low bound event frequency of approximately 0.5 SGTR/RY . Because the condition deteriorated with time, it can be argued that the initiating event frequency had not increased over the first year but only during the last year of operation. This would establish a high bound of 1 SGTR/RY. Multiplying these two estimates of the initiating event frequency by the SGTR CDF/LERF probability results in estimates for the incremental CDF of between 5×10^{-5}/RY and 1×10^{-4}/RY.

Con Edison Assessment:

The preliminary Con Edison assessment states that the probability of CDF resulting from a SGTR is 1x $10^{-6,}$/RY the initially assumed frequency of a SGTR as 1.3 X10^{-2}/RY, so the yearly incremental CDF conditional core dame probability is .77 x 10^{-4}/RY (1x10^{-6}/1.3x10^{-2})

Con Edison completed a more detailed calculation of CDF for the actual condiotns present at the time of the tube failure and for the actual leakrate observed. This caluclation assumes that the flow rate from the leak remains at below the design basis rate, whcih reduces the time to core damage and postpones the release time tot eh point that Con Edison believes it would not be considered an early release.

Significance Determination Process:

The magnitudes of the yearly incremental CDF for an SGTR in the initial NRC (1 x10^{-4}/RY and the preliminary Con Edison estimate (.77 x 10^{-4}/RY) are relatively the same.

The new Con Edison calculation indicates a specific conditional CDF of 2.2 E-6, with no LERF

The current guidance for assigning risk significance is contained in a draft NUREG/CR titled "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP) - Inspection Findings That May Affect LERF." The Office of Research is sponsoring the project at Brookhaven National Laboratory that is developing this guidance. The guidance is summarized in Table 1 of that document as shown here.

Table 1 Risk Significance Based on LERF and CDF		
incremental CDF Range/ry	**SDP Based on CDF**	**SDP Based on LERF**
$\geq 10^{-4}$	**Red**	**Red**
$< 10^{-4} - 10^{-5}$	**Yellow**	**Red**
$<10^{-5} - 10^{-6}$	**White**	**Yellow**
$<10^{-6} - 10^{-7}$	**Green**	**White**
$<10^{-7}$	**Green**	**Green**

Therefore, the CDF/LERF increment associated for a SGTR event is considered to be clearly above the 10^{-5}/RY criterion for a "red" significance determination. However the Con Edison assumption for lower than design SGTR leakage drops the CDF to 2.2 e-6 with no LERF - so this would be a white CDF finding with no LERF.

V. Potential Notices of Violations

10 CFR 50, Appendix B Criterion IX - Control of Special Processes and Criterion XVI - Corrective Actions, require, in part, that Con Edison, conduct an steam generator eddy current inspection program that promptly identifies and takes corrective actions for significant conditions adverse to quality.

Contrary to the above, Con Edison in 1997 failed to conduct a steam generator eddy current inspection program that promptly identified and took corrective actions for significant conditions adverse to quality. Consequently, a steam generator tube with detectable degradation was left in service following the 1997 refueling outage, eventually leading to the February 15, 2000, steam generator 24 tube row 2 column 5 failure. Specifically, Con Edison did not:

1. adjust the program to compensate for high noise signals in the low radius U-bend areas; these high noise signals negatively affected flaw detection capability;

2. take adequate corrective actions following identification of a new tube degradation mechanism, i.e., inside diameter (ID) primary water stress corrosion cracking (PWSCC) at the apex of a low radius U-bend tube;

3. establish a mechanism to monitor for and have an acceptance criteria for significant upper support plate flow slot hourglassing. Further, the potential existence and impact of upper support plate hourglassing on PWSCC flaws in the apex region of a low row U-bend tube was not assessed following the identification in 1997 of eddy current probe restrictions; and

4. ensure the use of properly qualified eddy current techniques. The U-bend plus-point eddy current probe, was not set-up with the proper calibration standard or with the phase rotation specified by the EPRI qualified technique sheet, which affected the probability of detection of U-bend indications.

SDP/ENFORCEMENT PANEL WORKSHEET
7/19/2000, Rev. C
Indian Point 2 - Steam Generator Tube Failure - 2/15/2000

EA:
Date of Panel: July 25, 2000

Licensee: Consolidated Edison Company of New York, Inc.
Facility/Location: Buchanan, New York

Docket No(s).: 50-247
License No(s).:
Inspection/OI Report No(s): 2000-010
Date of Exit Meeting/OI Report Date: July 18, 2000

Panel Chairman (SES Sponsor): Brian Holian
Responsible Branch Chief/Lead Inspector: David Lew/Wayne Schmidt
Enforcement Representative: Rick Urban
Other regional attendees: LATER
Headquarters attendees: LATER

References (attached):

1. Indian Point 2 Steam Generator Special Inspection Summary, dated July 19, 2000 - with attachments 1) NRC SDP; 2) Con Ed PSA Calculation

1. Brief Summary of Issues/Potential Violations:

Con Edison management did not establish an effective 1997 steam generator inspection program that provided for adequate overall technical direction, as required by 10CFR50, Appendix B, Criterion IX Control of Special Process and XVI Corrective Actions. As a result, Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality relating to eddy current data collection/analysis and specific steam generator conditions. This lack of program quality contributed to the February 15, 2000, tube failure, in that detectable flaws in low radius U-bend tubes, including the tube that failed, were not identified,

Severity Level III, White to Red finding is proposed. See Reference 1, Section V -Draft Notice of Violations.

2. Purpose of Panel:

Discuss the Special Inspection Team's summary of findings and conclusions document in Reference 1, Section 1.

Discuss and decide on the proper SPD assumptions for this event, see Reference 1, Section IV - Risk and Significance Assessment.

Based on an initial review of EGM 96-003 Steam Generator Tube Inspections, Updated June 1, 2000 Case #6 appears to apply - Potential Severity Level III. The risk significance of the event is discussed below currently it ranges from a White (prelim. Con Ed) to Red (prelim. NRC).

3. Regional Recommended Strategy:

The proposed enforcement action is a Severity Level III Violation. The Initial NRC Significance Determination Process Red characterization is preliminary and based on a initial assumption of a SGTR with core damage and a large early release (LER) from a stuck open safety valve.

The Con Edison provided calculation based on assumptions for the day of the event to determine a CDF with a time profile greater than 50 hours, their assumption is that this does not result in an LER, but a release after the emergency plan has had time to act to prot4ect the public.

A regulatory conference is recommended to discuss the perfroamcne issues and the CDF/LERF assumptions..

4. Analysis of Significance/Root Cause:

a. Actual Consequence(s):

The event had moderate risk significance. It involved a steam generator tube failure that resulted in an initial primary-to-secondary leak of reactor coolant of approximately 146 gallons per minute, and required an "Alert" declaration (the second level of emergency action in the NRC required emergency response plan). The event resulted in a minor radiological release to the environment that was well within regulatory limits. No radioactivity was measured off-site above normal background levels and, consequently, the event did not impact the public health and safety. The licensee's staff acted to protect the health and safety of the public. Specifically, the operators promptly and appropriately took those actions in the emergency operating procedures to trip the reactor, isolate the affected steam generator, and depressurize the reactor coolant system. Additionally, the necessary event mitigation systems worked properly. Notwithstanding the above, the NRC identified problems in several areas including operator performance, procedure quality, equipment performance, technical support, and emergency response. These problems challenged the operators, complicated the event response, and delayed the plant cooldown.

b. Potential Consequence(s):

See Reference 1, section IV - Risk and Significance Assessment for a discussion of preliminary NRC and Con Edison review.

Reference 1, Attachments 1 and 2 provide the initial NRC and Con Edison reviews, respectively.

c. Potential for Impacting Regulatory Process: LATER

d. Willful Aspects: None

e. Root Causes:

See Reference 1 - Section II - Conclusion/Root Cause

5. Apparent Severity Level(s)/Color and Basis:

See Section 2. above.

6. Application of Enforcement Policy

a. Enforcement/Performance History:

Indian Point 2 is an Agency Focus plant.

b. Is Credit Warranted for Identification? Explain:

Credit is not warranted for identification. The problem was revealed through the steam generator tube failure event of February 15, 2000. Con Edison did not:

1. adjust the program to compensate for high noise signals in the low radius U-bend areas; these high noise signals negatively affected flaw detection capability;

2. take adequate corrective actions following identification of a new tube degradation mechanism, i.e., inside diameter (ID) primary water stress corrosion cracking (PWSCC) at the apex of a low radius U-bend tube;

3. establish a mechanism to monitor for and have an acceptance criteria for significant upper support plate flow slot hourglassing. Further, the potential existence and impact of upper support plate hourglassing on PWSCC flaws in the apex region of a low row U-bend tube was not assessed following the identification in 1997 of eddy current probe restrictions.

4. ensure the use of properly qualified eddy current techniques. The U-bend plus-point eddy current probe, was not set-up with the proper calibration standard or with the phase rotation specified by the EPRI qualified technique sheet, which affected the probability of detection of U-bend indications.

c. Is Credit Warranted for Corrective Actions? Explain:

Credit is not warranted for Corrective Actions. Though the actions to correct the problem of stopping the primary to secondary leakage and associated release of radioactivity to the environment, the corrective actions are not comprehensive and are still being reviewed by NRR. Additional corrective actions, such as plugging all row 3 steam generator tubes is under discussion. Issues related to Con Edison's Condition Monitoring and Operational Assessment of the event are not yet resolved.

The root cause provided by the licensee was inadequate as described Reference 1, Section III - Performance Issues

d. Should Discretion Be Exercised to Mitigate or Escalate Sanction?

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There are three issues on the 'List of Issues That May Warrant Discretion' for consideration. See Attachment 2 for discussion.

7. **Is action being considered against individuals?**

No.

8. Non-Routine Issues/Additional Information/Relevant Precedent/Lessons Learned:

Generic communication may be needed for this issue regarding NRC expectations and observations related to the use of the EPRI Guidelines on steam generator eddy current testing, poor signal to noise ratios (high noise levels), the significance of top tube support plant hourglassing and U-bend/top support plate restrictions, and contractor oversight. NRR would provide any programmatic guidance deemed necessary.

Attachment 1

SDP/Enforcement Panel Disposition Record

Licensee : ________________
EA No. ____________________
Panel Date: ___________
Issue: ____________________________________

Attendees

Chair - ________ Branch Chief -_______ Enf Reps ____________
OI Rep. - __________ RI Counsel -_______ Others -________________________
HQ Reps ___________________

Required Actions (Preliminary Proposed Actions - See OE Strategy Form for official record of panel decision.)

1)

Responsible Person: ________________ ECD: _______

2)

Responsible Person: ________________ ECD: _______

3)

Responsible Person: ________________ ECD: _______

4)

Responsible Person: ________________ ECD: _______

Examples of Specific Actions To Be Documented

- Call Licensee and Schedule Conf or give heads up on choice letter
- Prepare summary of OI findings as attachments to choice letter or conf letter
- Issue letters scheduling conference or providing choice
- Gather additional information and repanel
- Prepare the draft enforcement action
- Finalize the enforcement action
- Forward Package to OE

Attachment 2

ISSUES TO CONSIDER FOR DISCRETION

☐ **Case involves particularly poor licensee performance. Yes**

☐ **Excessive duration of the problem resulted in a substantial increase in risk. Yes**

☐ **Discretion should be exercised by escalating or mitigating to ensure that any proposed civil penalty reflects the NRC's concern regarding the violation at issue and that it conveys the appropriate message to the licensee. Yes escalate**

The team found that Con Edison returned Indian Point, Unit 2, to service in 1997 in a condition that deteriorated with time to the point that a steam generator tube failure occurred.

The team concluded that during the 1997 steam generator inspection, Con Edison did not recognize and take corrective actions for significant conditions adverse to quality relating to eddy current data collection and analysis and specific steam generator conditions. These missed opportunities caused significant limitations and uncertainties, resulting in tubes with detectable flaws being left in service. Collectively, these opportunities, along with a new active degradation mechanism, increased the likelihood of tube integrity problems during the subsequent operating cycle. Since the plant is already an Agency Focus Plant escalation may not be needed.