From: Wayne Schmidt
To: Brian Holian, David Lew, Edmund Sullivan, Emmett Murphy, Hubert J. Miller, Jack Strosnider, Stephanie Coffin, Suzanne Black(...)
Date: Fri, Jul 7, 2000 4:27 PM
Subject: IP2 SG Summary

Hello all, please find attached my revisions to the previous summary, based on the discussions yesterday in headquarters. I hope this clears up some of the tech staff's concerns.

This is very draft and has not received Region I management review, but due to the time constraints involved I am sending it out.

Bare in mind that we are not complete with the sludge pile issue, but the write-up reflects our current understanding,

If there are any major hard spots I will be checking my e-mail over the weekend.

CC: Daniel Holody, David Nelson, Pete Eselgroth, Peter Habighorst, Richard Urban, William Raymond

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Indian Point 2 Steam Generator Special Inspection Summary

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

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.

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 also failed to identify outside diameter stress corrosion cracking (ODSCC) in five tubes in the sludge pile area, just above the tube sheet. With respect to the U-bend indications, 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.

Performance Issues;

1. Con Edison operated Indian Point Unit 2 during Cycle 14 with steam generator tubes in service that should have been removed from service during the 1997 refueling outage. Con Edison conducted a hindsight review of the 1997 eddy current data following identification in 2000 that SG 24 R2/C5 failed due to ID PWSCC and indication of ODSCC flaws in the sludge pile areas. TS 4.13 requires that tubes with defects greater than 40% through wall be removed from service by plugging. Based on this review Con Edison identified:

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1. four SG tubes that had 1997 estimated U-Bend PWSCC defects greater than 40%, one of which was the tube that failed. Con Edison documented this in Condition Report (CR) 2000-1939. Con Edison did not submit an LER on this apparent TS violation.
2. six SG tubes that had 1997 estimated sludge pile ODSCC defects greater than 40%. Con Edison documented this in their June 2, 2000 CMOA submital. Con Edison did not generate a CR nor submit an LER on this apparent TS violation.

2. 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.

3. 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.

4. 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

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row 1.

5. Con Edison's root cause determination, dated April 14, 2000, did not adequately address the failure to identify the tube flaws in the U-bend region during the 1997 outage. While the root cause analysis attributed the failure to eddy current signal noise, it did not identify inadequacies in Con Edison's technical oversight and management of the 1997 steam generator inspections. The root cause analysis failed to address Con Edison's lack of corrective action in response to a new SG degradation mechanism. The root cause analysis did not identify the improper set-up of the eddy current probe, and inadequate inspection and evaluation of the upper support plate denting and/or flow slot hourglassing.

Risk and Significance Assessment:

<u>NRC Assessment:</u>

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

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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 ($1x10^{-6}/1.3x10^{-2}$)

Significance Determination Process:

The magnitudes of the yearly incremental CDF for an SGTR in the NRC (1 $x10^{-4}$/RY and the Con Edison estimate (.77 x 10^{-4}/RY) are relatively the same.
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 this event is considered to be clearly above the 10^{-5}/RY criterion for a "red" significance determination.

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Potential Notices of Violations

A. Technical Specification 4.13 .B requires that steam generator tubes with defect depths of greater than 40% are not considered acceptable for continued service and shall be plugged.

Contrary to the above, during 1997 steam generator tubes that had depths of degradation greater than 40 % were considered acceptable for continued service and were not plugged, based on hindsight look at the 1997 eddy current data Specifically:

1. As document on Con Edison CR 2000 - 1939 - four tubes had U-bend indications with estimated depths greater than 40 % through wall as follows: SG 21 tube R2/C87 - 53%; and SG 24 R2/c5 - 87%, R2/C69 - 53% and R2/C72 - 75%

2. As documented in the Con Edison June 2, 2000, Condition Monitoring and Operational Assessment report (CMOA), Attachment SG-00-05-010, Table 4.5-1 Titled Indian Point 2 Sludge Pile ODSCC Growth rates from Bobbin Coil Analysis - five tubes had sludge pile ODSCC indications with estimated maximum depths greater that 40%through wall as follows: SG 22 tubes R34C51, R35C51, R35C52, R34/C54and R33/C49.

B. 10 CFR 50, Appendix B Criterion XVI - Corrective Actions, requires, in part, that Con Edison, promptly identify and take corrective actions for conditions adverse to quality.

Contrary to the above, Con Edison failed to promptly identify and plug six steam generator tubes with identifiable U-bend inside diameter primary water stress corrosion crack (PWSCC) during the 1997 refueling outage. Consequently, these tubes were left in-service after the 1997 refueling outage, eventually leading to the February 15, 2000, steam generator 24 tube row 2 column 5 failure.

C. 10 CFR 50, Appendix B Criterion IX - Control of Special Processes, requires, in part, that measures shall be established to assure those special processes, including nondestructive testing, are controlled and accomplished using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Paragraph 4.3 of Specification No. NPE-72217, "Eddy Current Examination of Nuclear Steam Generator Tubes, Indian Point 2," Revision 10 dated December 17, 1996, states, in part, "The examination technique shall be performed using qualified methods that are capable of detecting axial, skew, and circumferential cracking. The techniques used shall be qualified to the EPRI Steam Generator Examination Guidelines, Appendix H."

The EPRI Steam Generator Examination Guidelines, Appendix H qualified technique for low radius u-bends (96511Pwscc_ubend.doc) specified a phase rotation setting of 10°

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for a calibration standard 40 percent inside diameter through-wall circumferential and axial notches.

Contrary to the above, the Indian Point 2 specific qualification sheet (Sheet IP2-97-E, Revision 0,) specified a phase rotation so that probe motion was horizontal and the calibration standard did not include 40 percent through-wall circumferential and axial inside diameter notches. As such, the plus point probe technique used at Indian Point 2 in 1997 was not calibrated or set-up in accordance with the EPRI Appendix H qualified u-bend examination technique.

D. Technical Specification 4.13.C.3 requires, in part, the monitoring for significant hour-glassing (closure) of the upper support plat flow slots to ensure the long term integrity of small radius U-bends beyond row 1.

Contrary to the above, Con Edison did not adequately monitor for significant hour-glassing of the upper support plate flow slots. Specifically, Con Edison did not have a method to measure the flow slot hour-glassing nor a criteria to determine when it was significant, with respect to long term integrity of small radius U-bends beyond row 1.

E. CFR 50, Appendix B Criterion XVI - Corrective Actions, requires that Con Edison, determine the cause and take actions to prevent recurrence for a significant conditions adverse to quality.

Contrary to the above, Con Edison did not adequately determine the cause for the failure of SG 24 tube R2C5, as such corrective actions may not have been taken for a significant condition adverse to quality. Specifically, the root cause analysis did not identify inadequacies in Con Edison's technical oversight and management of the 1997 steam generator inspections. It failed to address the lack of corrective action in response to a new SG degradation mechanism and did not identify the improper set-up of the eddy current probe, and inadequate inspection and evaluation of the upper support plate denting and/or flow slot hour-glassing.

F. 10 CFR 50.73 requires that Con Edison submit a licensee event report within thirty day after the discovery of conditions prohibited by plant technical specifications.

Contrary to the above, as of July 7, 2000, Con Edison did not submit an LER within thirty days after discovery of conditions prohibited by plant Technical Specification 4.13. Technical specification 4.13 requires that steam generator tubes with defects greater than 40 % through wall be removed from service prior to returning the unit to operation. Specifically,

1. On March 3, 2000, ConEdison documented, in CR 2000-1936, that four steam generator tubes with U bend indications were greater than 40 % through wall.

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2. On June 4, 2000, Con Edison documented, in their steam generator Condition Monitoring and Operational Assessment (CMOA), Attachment SG-00-05-010, Table 4.5-1 Titled Indian Point 2 Sludge Pile ODSCC Growth rates from Bobbin Coil Analysis, that six tubes had sludge pile ODSCC indications with estimated maximum depths greater that 40% through wall.