From: Wayne Schmidt
To: "Doddcv[redacted]tees@[redacted] David Lew, Gregory Cranston, Stephanie Coffin
Ex. 6
Date: Wed, Aug 9, 2000 7:16 PM
Subject: Hot off my keyboard

Here is the first draft - see what you think - I am being asked for a very quick turn around - by tomorrow PM - so if it isat all possible please try.

I still need to do a few more cosmetic things like cleaning up the Acronyms and generating the list and the TOC. I'll do that tomorrow.

Everyone please give it a good general read.

Caius - please look closely at the Eddy Current Examination Technique section, Section 1RS2, and section 1RS4 Items 1 and 2. I still need the graphs with the revised lines. After we talked I included your graph foR R2C67 - so could you please include that in the list to redo the lines on - thanks. Also what was your depth estimate for R2C5 ?- could not find that.

Ian - please focus on Section 1RS3, and Section 1RS4 Items 1, 2, and 4&5.

Greg - please focus on the BACKGROUND Section and Section 1RS2.

Stephanie please give it a good general read - focusing on any areas where we may be setting policy inadverteantly.

I KNOW I HAVE NOT SAID IT ENOUGH - BUT - THANKS FOR ALL THE HARD WORK - IT SHOULD NOT BE LONG NOW.

53

P/14

EA No. 00-179-1

Mr. A. Alan Blind
Vice President - Nuclear Power
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: NRC SPECIAL INSPECTION 50-247/2000010- STEAM GENERATOR TUBE FAILURE

Dear Mr. Blind:

This letter transmits the results of a special inspection conducted by an NRC team at your Indian Point 2 reactor facility from March 7, through July 20, 2000, 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 team reviewed the adequacy of Con Edison's performance during the 1997 Steam generator inspections, and assessed Con Edison's root cause evaluation, dated April 14, 2000. On July 20, 2000, the results were discussed with you and other members of your staff. The preliminary team findings were sent to you by letter dated July 27,2000.

The team concluded that the overall technical direction and execution of the 1997 Steam generator inspection were deficient in several respects. Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected eddy current data collection/analysis. This increased the likelihood that detectable flaws in low row U-bend tubes were not identified.

During the 1997 Steam generator inspections, a new and significant degradation mechanism, PWSCC in the apex of a low row u-bend tube, and restriction at the upper support plate locations were identified and indicated increased susceptibility to this degradation mechanism. While the PWSCC indication, which was identified in 1997, was in an area of relatively low noise, the noise in similar areas was much higher and limited detection capability. However, Con Edison did not adjust or modify the inspection program to ensure an understanding of the extent of condition and increase probability of detection of other indications in tubes in the low row areas. As a result, four indications which should have been identified in 1997 were not identified and left in service until the failure of one of these tubes occurred on February 15, 2000.

The report identifies the failure to evaluate and take action to correct and account for signal noise and, to adjusted or modify inspection methods and analysis to account for the anomalies and other new conditions encountered as an issue of high safety significance with a significant reduction in safety margin, which is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. This issue was assessed using the Reactor Safety Significance Determination Process as an apparent significant finding that was preliminarily determined to be Red. This issue was of high safety significance because of the increased risk of a steam generartor tube rupture.

As discussed with Mr. John McCann of your staff we have scheduled a Regulatory Conference for September 7, 2000, in the Region I office to discuss your evaluation and any differences with the NRC evaluation prior to our final significance determination on the10 CFR 50, Appendix B Criterion XVI issue discussed above.

The NRC also identified an issue involving improper calibration and setup of the eddy current technique used to examine the U-bend areas of low row tubes. This issue was evaluated under the Reactor Safety Significance Determination Process as of very low safety significance (Green). The issue involved a violation of NRC requirements, but because of the very low safety significance, normally the violation would not be cited. However, you disagreed with the violation at the exit meeting. We will be prepared to discuss this issue during the September 7, 2000, Regulatory Conference, prior to our final enforcement determination.

The Regulatory Conference is an opportunity to provided us with additional information including your position on the significance of both issues discussed in the attached report, the bases for your position, and whether you agree with the apparent violations. The Regulatory Conference on these matters will be open for public observation. Accordingly, no enforcement is presently being issued for these inspection findings. Following the conference we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Should you have any questions regarding this report, please contact Mr. David C. Lew at 610-337-5120.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000247
License No. DPR-26

Enclosure: Inspection Report 05000247/2000-010

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Mr. A. Alan Blind 4

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REGION I

Docket No.	05000247
License No.	DPR-26
Report No.	05000247/2000-010
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Broadway and Bleakley Avenue Buchanan, New York 10511
Dates:	March 7, through July 20, 2000
Team Manager:	David C. Lew, Chief, Performance Evaluation Branch, DRS
Team Leader:	Wayne L. Schmidt, Senior Reactor Inspector, DRS
Inspectors:	I. Barnes, NRR, Part-Time Contractor S. Coffin, NRR G. Cranston, Reactor Inspector, DRS L. Dudes, Senior Resident Inspector, Oyster Creek C. Dodd, NRR, Part-Time Contractor M. Modes, Senior Reactor Inspector, DRS E. Murphy, NRR J. Trapp, Senior Reactor Analyst, DRS
Approved by:	David C. Lew, Chief Performance Evaluation Branch Division of Reactor Safety

Mr. A. Alan Blind

SUMMARY OF FINDINGS

Indian Point 2 Nuclear Power Plant
NRC Inspection Report 05000247/2000-010

Using the guidance in NRC Management Directive 8.3 and Inspection Manual Chapter 2515 the NRC conducted a Special Team Inspection from March 7, through July 20, 2000, to review the causes for the Steam generator tube failure event that occurred on February 15, 2000. The team also assessed the safety significance of the findings using the Reactor Safety Significance Determination Process in Inspection Manual Chapter 0609 (see Attachment 1). The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED).

Cornerstone: Barrier Integrity and Initiating Events

- **TBD -Potential Red** During the 1997 refueling outage, a significant condition adverse to quality existed at Indian Point 2, namely, primary water stress corrosion cracking (PWSCC) flaws in the low row u-bends of four tubes in the steam generators; however, as of February 15, 2000, when one of those tubes failed while the plant was at 100% power, measures were not established to ensure that the condition adverse to quality had been identified and corrected, despite opportunities that existed to do so. Those prior opportunities involved other significant conditions adverse to quality for which the causes had not been determined. Specifically, during eddy current testing of Steam generators during the 1997 outage,

  1. a PWSCC crack was identified at the apex of one of the low row u-bend tubes. Since this was the first time in the facility's history that a crack had been identified at the apex of any tube, it signified the potential for other similar cracks in the low row tubes.

  2. indications of tube denting were discovered for the first time in the uppermost support plate of Steam generator tubes when restrictions were encountered as eddy current probes were inserted into those tubes. These restrictions in 19 low row tubes signified the susceptibility to deform the flow slots (hour-glassing) at the uppermost support plate, which, in turn, indicated additional PWSCC stresses on the low row u-bend tubes.

  3. significant eddy current signal interference (noise) was encountered in the data obtained during the actual eddy current testing of several other low row u-bend tubes, which could impede the detection of similar indications that may have existed in other tubes.

  Although the above issues were reasonably identifiable Con Edison (1) did not evaluate nor take action to correct and account for these impediments (to detection of any other flaws) that the noise created at the time; and, (2) did not adjusted or modify inspection methods and analysis during the inspections process to account for the anomalies and other new conditions encountered. As a result, four indications were not promptly

Mr. A. Alan Blind

identified in the 1997 outage and were left in service until the failure of one of these tubes occurred on February 15, 2000. These involved matters that had high safety significance with a significant reduction in safety margin since the potential for a Steam generator tube rupture event was significantly increased. The team identified this as an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. Con Edison disagreed with the characterization of this issue during the exit meeting. (Section IRS2.4)

- **TBD Potential Green** - During the 1997 Refueling Outage the U-bend Plus Point eddy current probe was not properly setup to the correct calibration standard. This had a marginally negative effect on 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. This issue involved matters that had very low risk significance because there was no effect on the reactor coolant system integrity. The team identified this as a potential violation of 10 CFR 50, Appendix B, Criterion IX, Special Processes, because Con Edison disagreed with the characterization of this issue during the exit meeting (Section IRS3.5)

Mr. A. Alan Blind

TABLE OF CONTENTS

Mr. A. Alan Blind

Report Details

BACKGROUND

Summary of Plant Event

Following the steam generator (SG) tube failure on February 15, 2000, Consolidated Edison Company of New York, Inc. (Con Edison) took the Indian Point Unit 2 (Indian Point 2) to a cold shutdown condition. Con Edison conducted an evaluation and found that the tube that failed was row 2 column 5 (R2C5) in SG 24. This small radius (low row) tube had cracked at the apex of the U-bend, due to primary water stress corrosion (PWSCC) cracking. Con Edison conducted an eddy current (EC) examination of the SG tubes and conducted visual inspections of the secondary side of the SGs. During these EC inspections Con Edison found greater than 1% of the tubes contained defects in SGs 21 and 24 placing the unit in a condition that required NRC approval of a restart, in accordance with technical specification (TS). At the conclusion of the inspection the unit remained in cold shutdown pending NRC restart approval.

NRC specifically reviewed Con Edison's response to the February 15, 2000, event in the Augmented Inspection Team (AIT) Report 05000247/2000-003 and AIT Followup Report 05000247/2000-007.

Steam Generator Description

Indian Point 2 is a four loop pressurized water reactor, meaning that there are four SGs, one per loop, that transfer heat from the reactor coolant to the secondary water. This heat causes the secondary water boil and the associated steam is used to run the turbine with turns the electrical generator. Figure 1 shows a Westinghouse Model 44 SG, like those installed at Indian Point 2.

Each SG was built with 3,260 tubes, these tubes have the reactor coolant running through them and the secondary water/steam on the outside. The tubes are made of mill annealed Inconel alloy 600 and are arranged in an inverted U fashion with increasing distances and heights from the inter-most row (row 1) outward. The tubing has an outside diameter of 0.875 inches and a wall thickness of 0.050 inches average. Each tube is numbered by it row number, counting from the center out, and its column number, counting from one side of the SG. The low row tubes (rows 1 - 4) each have 92 tubes installed. The tubes are supported vertically by the thick tube sheet at the bottom of the SG and horizontally as they pass through drilled-holes in the six evenly spaced carbon steel tube support plates (TSP). In each TSP there are whole cut to allow water/steam flow around the tubes, also there are six evenly spaced flow slots cut that running across the diameter, between the two legs of the adjacent row 1 tubes. The flow slot openings are about 15 inches long (spanning about twelve tubes) and originally were about 3 inches wide. The U-bend area is located above the upper TSP.

The row 1 tubes were plugged prior to initial operation.

Technical Specification

SG tubes have an important safety role because they constitute a barrier between the radioactive primary side and non-radioactive secondary side of the plant. During operation SG tubing can degrade due to corrosion mechanisms and mechanical wear on the outside diameter

(OD) or the inside diameter (ID) of the tubing. The plant's technical specification require that a representative sample of the SG tubes be examined using EC during a shutdown to ensure identification of degraded tubes and the removal of tubes with defects from service. If degradation is found, the sample of tubes is expanded to ensure that the sample remains representative of the overall SG conditions. Tubes with degradation greater than 40% through the wall (TW) are considered defective and must be removed from service. Tubes are removed from service by inserting a plug at both end of the tube. These plugs are designed to limit the amount of reactor coolant that will enter the degraded tube.

During operation the amount of primary coolant that leaks into the secondary coolant is referred to as primary to secondary leakage. The rate of this leakage is controlled by the plant technical specifications and is limited to ____ gallons per minute (gpm). Primary to secondary leakage can exist from several sources, leaking tubes that are inservice and through plugs in tubes that have been removed from service. The primary to secondary leakage is monitored through mass balance (knowing how much water is added to and taken out of the primary system) and by radiological analysis (knowing the primary coolant activity and comparing it to the secondary water activity).

Technical specifications also contain a requirement to report significant deformation of the upper tube support plate (hour-glassing) (see Applicable Steam Generator Degradation Mechanisms below), since it can have a significant effect on the integrity of the tube beyond row 1.

Eddy Current Examination Technique

EC is a method of inspecting SG tubes by passing a probe that generates an electromagnetic field and senses the disturbance of the field due to defects in the tubing. The technique works due to the principle of electromagnetic impedance of a coil in an AC circuit. In such a circuit the impedance of the coil causes the circuit voltage and current to be out- of-phase. Changes in the coil impedance are observed by changes in the voltage across the coil an in how much the voltage and current are out-of-phase (refereed to as phase angle).

By definition an eddy current is an electrical current caused to flow in a conductor due to the variation of an electromagnetic field. In EC a varying electromagnetic field is generated when an alternating current is passed through the probe which consist of a wire coil. This eddy current is opposite to the probe current. The eddy current is directly effected by a defect that is perpendicular to its flow. When the probe is inside a tube EC looks for changes in the coil impedance due to a defect that is obstructing the eddy current flow with in a tube. The defect can be detected observing changes in the coil voltage and phase angle..

Single coil probes as shown in Figure 2 will induce the eddy current in only one direction, which is a compressed mirror image of the current in the coils. If the defect is not in the direction which interrupts the eddy-current flow (parallel to the defect direction rather than perpendicular to the current flow), then that particular coil will not detect the defect. Specially designed eddy-current probes can classify defects as axial cracks, circumferential cracks or both.

The frequency of the alternating current sent to the probe affects how deep the eddy current penetrates into the tube - the higher the frequency the lower the penetration. Probes have been designed that operate at several frequencies at one time, one probe may collect different frequency data during an examination.

The Plus Point probe consists of two coils wound at 90 degrees to each other, as shown in Figure 3. The coils are mounted on a shoe that rotates as it passes through the tube to allow a complete examination. The turns of the two coils are interleaved so that both are effectively the same distance from the surface of the conductor. The coils are connected in a bridge circuit, as shown in Figure 4, and voltage the difference between the two signals is amplified. The two coils allow the scanning for both axial and circumferential defects. The mid range Plus Point probe used during the 1997 examination is multifrequency probe, operating at 10, 300, and 400 KHz.

Noise in eddy current testing is defined as any non-relevant signal that tends to interfere with the normal reception or processing of a desired flaw signal. Signal to noise ratio is a way of evaluating the magnitudes of a relevant signal (defect) to the non-relevant signal (noise). The higher the signal to noise ration the easier it is to detect a defect.

EC techniques are qualified for SG inspection use by the Electric Power Research Institute (EPRI). This qualification includes the verification that the technique can identify know defects with a probability of detection (POD) of greater than 80 % with a 90% confidence. The techniques are calibrated as with any instrument to known calibration standards during there use. These calibration standards include additional known defects and the analysts sets up the according to the known defect size.

EC information may be displayed in numerous forms, several of which are shown in the figures in the back of the report. The c-scan plot is a pictorial view, as if the tube was split and laid out flat, of the changes in probe impedance, it shows by a voltage reading that has been adjusted for phase angle (refereed to as the vertical component). The strip chart is a look at the high and low values shown on the c-scan. The lissajous is a pictorial view of the voltage and phase angle effects at a specific point in the tube. Flaws, outside tube deposits, TSPs, and the tube roll transitions all have an effect on the EC signal and have a characteristic lissajous signal.

Through extensive training and qualification the EC analyst becomes familiar with the different effects and is able to detect a flaw. Through different techniques and data analysis the analyst can make an estimate of the size (depth and length) of a defect.

Applicable Steam Generator Degradation Mechanisms

Stress corrosion cracking (SCC) is cracking caused by the simultaneous presence of a tensile stress, a specific corrosive medium, and a susceptible material. SCC can initiate from either the tube's ID or OD. When initiated on the ID it is referred to as PWSCC and on the outside ODSCC.

Based on the crack characteristics, a PWSCC (and a SCC defect in general) defect yields only 20 to 70 percent (and perhaps less) of the EC signal amplitude of a similarly sized calibration standard defect.

PWSCC in particular is associated with areas of high stresses and thus are most commonly found in the tubesheet expansion transitions, in the U-bend transition and apex regions of the small radius, low row tubes, and in the tube support intersections especially if the tubes are dented.

Denting of the tubes is the direct result of secondary side corrosion of the carbon steel tube support plates. When the SG is shutdown and cool there is a circumferential gap between the tube outer wall and the hole in the TSP through which it passes. The gap is there by design to allow for tube thermal expansion as the reactor coolant system temperature is increased prior to a reactor startup. However, while the SG is shutdown corrosion products can form, based on water chemistry, and harden in that gap. As the reactor coolant system and the tubes heat up tube expansion at the TSP is prevented due to the hardened corrosion products. The forces generated cause several things to happen.

- Since the tube cannot expand at the TSP, it tube becomes permanently dented, circumferentially. The cooldown, corrosion, heat up, and denting cycle reoccur with each shutdown and restart, as influenced by SG water chemistry.
- Eventually the denting process can continue until the tube ID is so closed that an eddy current probe will not pass through. This is a restricted tube.
- The forces causing the denting may induce tensile stresses in the tube ID or OD near the dent leading to localized SCC.
- The forces causing the denting also act against the tube support plate, in the area of the flow slots where the structural resistance is low enough, deformation and/or cracking of the TSP can occur. This happens on both sides of the flow slot forcing the sides of flow slot inward at the middle of the flow slot causing the previously rectangular shaped flow opening to develop the shape of an hour-glass and is referred to as hour-glassing. In the low row U-bend areas PWSCC is more significant if hour-glassing forces the tube legs closer together, concentrating tensile stress at the apex of the U-bend.

Steam Generator History

The team compiled a history of the Indian Point 2 SGs, finding that they have experienced a broad range of tube degradation modes, requiring plugging of tubes. The causes are common to the industry and include: tube sheet (TS) roll transition PWSCC, ODSCC in the area between the roll transition below the top of the TS (crevice), ODSCC in the sludge pile area, ODSCC and PWSCC and probe restrictions in dented areas, and U-bend PWSCC and ODSCC.

Due to the composition of some secondary system components at Indian Point 2, deposits on secondary wall of the tubes contain large amounts of hematite (Fe_2O_3), interspersed with metallic copper. These deposits generally do not o promote severe tube corrosion. However, they can have an effect of increasing the noise in an EC signal.

In May 1995 Con Edison completed refueling outage 12 (RFO 12), during that SG inspection no PWSCC defects were identified in the U-bend region, however there were PWSCC cracks identified at the roll transition in the tube sheet.

In May 1997 the unit was shutdown for the end of cycle 13 (EOC 13) refueling outage. Con Edison submitted and discussed the plans for the SG inspection with NRR staff, prior to the outage - the plan included a 100 % Plus Point scan of the low row U-bends. The examination completed in June 1997 finding the first low row U-bend PWSCC indication in SG 24 at the apex of R2C67. This tube was plugged prior to restart, no insitu pressure test was completed. Also during this examination Con Edison identified the first instances probe restrictions caused by denting at the upper TSP in low row U-bend tubes. These tubes were plugged because an examination could not be completed.

Con Edison returned Indian Point 2 to operation in early July 1997. The unit was shutdown in October 1997 due to problems with the operation of DB-50 circuit breakers. Following extensive corrective action the unit was returned to operation in August 1998. The unit remained in operation until August 1999 when a loss of offsite power caused an automatic trip. The unit restarted in October 1999.

Primary to secondary leakage during the operating periods remained low less than 2 gallons per day (gpd) through December 1999. By early February 2000 total leakage was approximately 2.1 gpd with 1.2 gpd attributed to SG 24. On February 15 initial primary to secondary leakage was 3.1 gpm and increased following the failure of tube R2C5 in SG 24 to approximately 150 gpm, greater than the capacity of two charging pumps, but not greater than the specific design basis SG tube rupture (SGTR) leak rate of ??? gallons per minute.

1. REACTOR SAFETY - SPECIAL (RS) - Cornerstones: Initiating Events, Barrier Integrity

1RS1 Initial Review of Eddy Current Data Following The Tube Failure (Cornerstone - Barrier Integrity)

g. Inspection Scope

The team initially conducted on site reviews of Plus Point EC data being taken on the U-bend locations in 2000.

h. Observations and Issues:

Initially Con Edison used the same data analysis guidelines as used in 1997 - there had been no revisions.

2000 data indicated high noise in the U-bend areas and low signal to noise ratios. There were no specific criteria to ensure the identification of that defects buried in the noise. As a result of NRC questioning of the high nosie, Con Edison and its contractor developed an additional training handout which provided more detail in how to interpret noise in the data stream.
The team questioned Analysis Technique Specification Sheet (ANTS) IP2-97-E, Rev 0, dated 5/8/97 for the of the 1997 data, finding that it had been incorrect (see Section 1RS4, below). Con Edison and there contractor subsequently used the correct phase angle setup in evaluation of the 1997 data.

Initially for the 200 outage the U-bend Plus Point phase setup (ANTS IP2-00-Rev 1, dated 2/27/00) was not properly setup, the setup had not changed from the erroneous setup in 1997. In early March the ANTS was revised (ANTS IP2-00-E, Rev2 , dated 3/4/00) to incorporate the appropriate setup in accordance with the ERPI qualification of the probe Examination Technique Specification Sheet (ETTS) # 96511 Pwscc_ubend.doc dated May 1996. All the year 2000 U-bend examinations that had previously been completed were repeated using the corrected setup.

i. Findings

There were no findings identified. The U-bend probe calibration and setup issue is discussed relative to 1997 performance in section 1RS4 below.

1RS2 Review of 1997 Inspection Relative to Low row U-bends (Cornerstone - Barrier Integrity)

.1 Eddy Current Data Review

a. Inspection Scope

The team reviewed the 1997 EC data collected on eight tubes that were identified as possibly having detectable flaws in 1997, including tube R2C5 in SG 24. During this review the team used the actual data collected in 1997 and assessed the detectability of these flaws and there potential size in 1997.

b. Observations and Issues

1997 data contained significant noise, possibly due to deposits on the U-bends tubes. Con Edison did not identify the possible effect that the noise could have on the POW of flaws. Techniques to minimize the effects of the noise on data quality were not used and a careful review of EC data affected by noise was not taken.

The depth profiles provided are the team's estimates of defect depth verse axial distance along the tube. The axial distance is relative to an approximately 13.3 inch distance (above the upper support plate) through the U-bend of a row 2 tube. The tube profiles show noise and a poor signal to noise ratio, which introduced a large uncertainty in the measurement of the crack depth.

1. R2C5 in SG 24 - this was the tube that failed during operation. Figure 5 is a c-scan plot of the vertical component of the EC voltage signal. The defect signal, indicated by the arrow, sits on a noise ridge that runs the length of the tube. This noise ridge is about 1-volt in amplitude and measures as a deep id defect, on the order of 70 to 100% deep. This ridge makes both the detection and sizing of this defect more difficult. In Figures 6 and 7 are the lissajous plots for the flaw area and the noise ridge, respectively. The signal-to-noise is slightly better for the 400 kHz frequency than the 300 kHz. No year 2000 data is available since the tube

failed.

The indication has been profiled for both frequencies, as is shown in Figure 4. The 400kHz profile is probably slightly more accurate than the 300 kHz profile. Neither is very accurate, and until the defect voltage increases above 1.2-volts, there is considerable error. The depth estimate based on 1997 data is ? % TW.

2 . R2C69 in SG 24 - Figure 9, shows the c-scan plot for the 1997 data, there is considerable noise present. For comparison the c-scan plot for the 2000 data is included as Figure 10. The noise features between the 1997 and 2000 data are similar enough to verify that this is the same defect at the same location. Figure 11 shows the profile, the defect voltage is only about 1 volt, and there is a considerable amount of noise on the tube, relative to the defect signal. The depth estimate based on 1997 data is 52.6 % TW.

3. R2C72 in SG 24 - Figure 12, shows the c-scan plot of the 1997 data, there is considerable noise present. The crack is sitting in a ridge of noise, and barely extends above a ridge of deposits. For comparison the c-scan plot for the 2000 data is included as Figure 13 Figure 14 shows the profile The crack barely extends above a 1-volt amplitude for a short length, and this is the only part of the crack that we can profile reliably. The depth estimate based on 1997 data is 79.2 % TW.

4. R2C87 in SG 21 - this tube was identified as having several cracks. Figure 15 shows the c-scan plot of the 1997 data, the most prominent crack is sitting in a relatively clean area of the tube. For comparison the c-scan plot for the 2000 data is included as Figure 16. Figure 17 shows the profile of the most prominent crack. The depth estimate based on 1997 data is 63.7% TW.

Based on this review, the team determined that Con Edison should have identified the defects in these four tubes and plugged them due to U-bend degradation, prior to restart from the 1997 refueling outage.

On March 20, 2000, Con Edison initiated CRS 200001939 which documented these four tubes as having defects greater than 40% TW prior to restart from the 1997 outage based on their review of the 1997 data. The depths recorded by Con Edison were SG 24; R2/C5 - 87%TW, R2/C69 - 53%TW, R2/C72 - 75%TW and In SG 21 R2/C87 - 53%TW. This review compares well with the independent team review. The team noted that the closure of this CRS did not provide a clear statement as to why this issue was not reportable as a TS violation. It appeared to use generic information such as NRC Draft NUREG -1477 and NEI 97-06 as justification for not complying with the Indian Point 2 technical specifications.

.2 Review of U-bend PWSSS Indication

a. Inspection Scope

The team reviewed the 1997 EC data and the actions taken upon discovery of a PWSCC flaw at the apex of SG 24 R2C67. As discussed above, Con Edison used the Plus Point technique to conduct the U-bend examination. A crack depth profile was also generated.

The team reviewed the Indian Point Steam Generator Life predictions with respect to U-bend PWSCC. These predictions were completed by Dominion Engineering in 1995 following the 1995 outage and in 1997 following completion of the 1997 outage.

b. Observations and Issues

Figure 18 shows the c-scan plot of the 1997 data from R2C67, the crack sits beside a ridge, in a valley, and is in an easily detectable portion of the tube. The large amplitude of the voltage signal, in relation to the standard calibration notch would indicate that this is a "mature" crack. No year 2000 data is available since the tube was plugged in 1997. Figure 19 shows the profile. The depth estimate based on 1997 data is 87.5% TW.

Upon discover of the apex indication neither Con Edison nor its contractor entered the issue into their corrective action programs. This was significant since this was first time this type of degradation was observed in the U-bend area at Indian Point 2. There was no specific review as to the significance of this flaw nor the possible extent of the condition.

The EC contractor reported the discovery in the end of outage report and by Con Edison reported it to the NRC in the 1997 SG Examination Refueling Outage report.

SG Life Predication

The 1995 report used industry data to predict the number of SG tubes that would have to be plugged due to PWSCC during the life of the unit. Based on a best case estimate and a reactor coolant T_{hot} of 589, the actual temperature through the period, the report predicted no PWSCC cracks in the U-bend area throughout the entire licensed life of the of Indian Point 2. The pessimistic estimate predicted 1 PWSCC U-bend crack at the end of the last cycle of operation (EOC 21)

This report recommended a rotating pancake coil(RPC) scan of the low row U-bends and further stated
"Industry experience shows that U-bend defects can often result in forced outages due to relatively rapid increases in coolant leakage through the defect. RPC inspection of the remaining in-service row 2 and 3 U-bends at IP2 over the next few outages is recommended, as a means for identifying U-bend PWSCC defects before they cause leaks. However, experience has shown that small PWSCC defects below the RPC detection threshold can grow through-wall or near through-wall during a single cycle. Consequently, it is difficult to completely protect against forced outages due to U-bend PWSCC for plants experiencing this type of degradation mechanism. Stress relief heat treatment can reduce the likelihood of through-wall defects occurring during a single cycle, but may not be effective if performed after a long period of service as shown by

the Diablo Canyon experience."

Following the 1997 outage and the identification of the one PWSCC indication Dominion Engineering updated their prediction. At that point, the best estimate case predicted one additional PWSCC indication at EOC 17, with an additional defect in EOC 19 and EOC 20.

.3 Denting and Hour-glassing

a. Inspection Scope

The team reviewed the TS 4.13, the 1997 SG Examination Refueling Outage report, dated July 29, 1997, NRC requests for additional information following the tube failure and Con Edison subsequent responses, and the Indian Point 2 Steam Generator Data Book, dated December 1, 1997, to assess SG conditions in 1997 relative to tube denting and hour-glassing.

The team reviewed the Indian Point Steam Generator Life predictions with respect to tube requiring plugging following denting restrictions. This predictions were completed by Dominion Engineering in 1995 following the 1995 outage.

b. Observations and Issues

Early in the inspection, the team questioned the possible hour-glassing that could have occurred to cause tube R2C5 to develop PWSCC and lead to the tube failure. The team found that Con Edison did not have inspection ports in SG 24 to allow such an inspection. Further the team found that Con Edison had not been doing any direct measurement of hour-glassing in the two SGs that had inspection ports in the upper TSP region. Con Edison conducted visual examination in the upper TSP areas using boroscopic techniques, but had no method of measuring nor a criteria for when hour-glassing was significant. As such Con Edison never reported any significant hour-glassing.

Con Edison installed an inspection port on SG 24 to allow the measuring of the hour-glassing near tube R2C5. Con Edison developed a technique to measure the defection of the row 1 tubes, finding that 0.46 inches of movement had occurred. Con Edison also conducted an engineering study to determine the amount of movement that would cause a critical stress in the apex of the U-bends for row 2, row 3 and row 4 tubes. The amount of movement to cause the critical stress increases with the increasing rows since the tube legs above the upper TSP are longer, further a part, and have larger radius U-bends. The critical movement for row 2 tubes was 0.1inches. This calculation proved that the stress in R2C5 was above the threshold for PWSCC..
The 1997 SG tube inspection identified 37 tubes that needed to be plugged due to denting at TSPs. Of significance 19 were recorded as U-bend restrictions as documented in the 1997 SG Examination Refueling Outage report. Through discussions with Con Edison the team found that denting in low row tubes (row 2 -15, row 3 - 3, and row 4 -1) at the upper TSP caused the U-bend restrictions, not allowing

examination of the upper TSP area. These tubes were plugged

These 19 tubes was a significant increase above the numbers of restrictions identified in the last several outages (1 - RFO-15, 0 - RFO- 14, and 1- RFO-13) . Further the total of 37 was above the 1995 SG life prediction best estimate of 25 tubes being plugged due to denting during the 1997 outage.

Con Edison did not identify the increase in tube denting at the upper support plat as a issue, with respect to the potential for flow slot hour-glassing adversely impacting tubes beyond row 1. (See Applicable Steam Generator Degradation Mechanisms above under BACKGROUND)

.5 Findings

During the 1997 refueling outage Con Edison reasonably should have identified, reviewed, and taken actions to assure that Indian Point 2 was not returned to service with SG tube that contained detectable PWSCC indication in the low radius U-bend area. The significant noise present in the EC data for the low radius U-bends, which hampered the capability to detect flaws in this region, coupled with identification of the first PWSCC defect in a low radius U-bend, and the first 19 tubes plugged due to upper TSP restriction, provided sufficient evidence of the potential for flow slot hourglassing and the resulting increased stresses and PWSCC at the apex of the U- bends.

The team concluded that the overall technical direction and execution of the 1997 SG inspection program were deficient in several respects. Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected EC data collection/analysis. This increased the likelihood that detectable flaws in low row U-bend tubes were not identified.

More specifically, Con Edison did not:

1. take appropriate corrective actions following identification of a new and significant tube degradation mechanism, i.e., inside diameter (ID) primary water stress corrosion cracking (PWSCC) at the apex of a low row U-bend tube. Operating experience indicates that apex cracking is more likely to result in tube failure than other U-bend cracks. The 1997 SG inspection program did not fully assess the implications of this new degradation mechanism and adjust, as appropriate, the inspection methods and analyses.

2. recognize the significance of, and fully evaluate, the flaw masking effects of the high noise encountered in the EC signal. In the case of the SG tube that failed, the magnitude of the noise was a problem that negatively impacted the probability of detection. The data analysis techniques were not adjusted to compensate for the noise to improve the identification of a flaw signal and ensure the appropriate probability of detection, particularly when conditions which increased

susceptibility to tube degradation existed.

3. appropriately establish procedures and implement practices to address the potential for hour-glassing in the upper support plate flow slots. Hour-glassing in this location is indicative of increased stresses on the SG tubes, which increase the likelihood of tube cracks. Further, the potential existence and impact of upper support plate hour-glassing were not assessed following the identification in 1997 of EC probe restrictions at the upper support plate and the identification of a PWSCC indication at the apex of a SG tube.

Using the Reactor Safety Significance Determination Process (SDP), as documented in Section 1RS5 below, the team's preliminary evaluation was that this is a matter of high safety significance with a significant reduction in safety margin which is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. This issue was of high safety significance because of the increased risk of an SGTR. We note that you disagreed with this issue during the team's exit meeting. In accordance with NRC Enforcement Policy and Reactor Safety SDP this matter is considered an apparent violation. **(AV 50000247/2000-010-01; EA 000-179-1)**

1RS3 Review of The 1997 Eddy Current Inspection Program (Cornerstone - Barrier Integrity)

.1 Eddy Current Technique Qualification

a. Inspection Scope

The team reviewed the overall qualification of the Plus Point EC probe for use during the 1997 inspections. Specifically the team reviewed:

- Specification No. NPE-72217; "Eddy Current Examination of Nuclear Steam Generator Tubes, Indian Point 2," Revision 10. Which contained the technical requirements for the 1997 SG tube examinations (Cycle 13 refueling outage)
- EPRI Steam Generator Examination Guidelines Rev 4, Appendix H
- "Eddy Current Low Row U-bend Examination, MIZ-18A and TC6700, Non-Mag. Bias and Mag. Bias Equivalency Qualification." This undated document which did not contain an alpha-numeric identifier was prepared by Westinghouse. The purpose of this equivalency qualification was to demonstrate that the magnetic bias Plus Point probe (which was used for examination of the IP2 low radius U-bends) had comparable detection capability to the non-magnetic bias Plus Point probe.
- ETSS # 96511Pwscc_ubend.doc, dated May 1996, the EPRI Performance Demonstration Data Base document that qualified the Plus Point probe for detection of circumferential and axial PWSCC in low radius U-bends.
- Analysis Technique Specification (ANTS) Sheet # IP2-97-E, Revision 0 - documentation of the analysis method of SG low radius U-bends at IP2 including requirements for setting of phase rotation and use of calibration standards.
- Westinghouse Drawing 1B79882, Revision 0, which pertained to the ACGT-006-97 EDM - the calibration standard that was used for the 1997 Plus Point probe examinations of

low radius U-bends

b. Observations and Issues

Specification No. NPE-72217, Paragraph 4.3 stated, 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,"

Paragraph H.1 in Appendix H, "Performance Demonstration For Eddy Current Examination," of the EPRI Guidelines states, in part, "...Each organization that performs EC examinations shall use techniques and equipment qualified in accordance with this Appendix...." Paragraph H.2.1.1 in Appendix H identifies that calibration method is an essential variable to insure proper data acquisition. Paragraph H.2.1.2 in Appendix H further requires the Analysis Technique Specification Sheet (ANTS) to define the method of calibration used for signal characterization.

Paragraph 7.1 in the EPRI Guidelines states, "Nondestructive examination of SG tubes shall be conducted using techniques capable of detecting and/or sizing the types of degradation known or reasonably expected to exist in accordance with industry experience. An inspection technique is qualified if sensors (coils, transducers, etc.) used have been proven capable by performance demonstration to meet the requirements of Appendices H and/or J.

ETSS # 96511Pwscc_ubend.doc was the EPRI Performance Demonstration Data Base that qualified the mid-range Plus Point probe, for detection of circumferential and axial PWSCC in low radius U-bends. This technique utilized a calibration standard containing 100% TW axial, and 40% TW axial and circumferential inside diameter EDM notches. A phase rotation setting of 10° was specified in the section of the ETSS entitled, "Data Analysis," for the 40% TW circumferential and axial notches. The "Analysis Guidelines" portion indicated, however, the use of a 10-15°phase rotation setting for the 40% TW EDM notches.

The team identified two instances in the 1997 implementation of the mid-range Plus Point U-bend technique where the requirements of ETSS # 96511Pwscc_ubend.doc were not met.

- The calibration standard ACGT-006-97 manufactured in accordance with Westinghouse Drawing 1B79882 did not include the 40% TW inside diameter axial and circumferential EDM notches required
- This ANTS sheet instructed the analyst to adjust phase rotation so that probe motion was horizontal, which was both not in accordance and considered technically deficient by the inspectors. The inspectors considered this guidance to be technically deficient, due to the insensitivity of the Plus Point probe to probe motion resulting in too small a signal to allow the adjustment to be accurately accomplished. The ANTS sheet additionally provided no instructions to the analyst with respect to the phase rotation criteria to be used for axial or

circumferential notches.

These issues resulted in performance of 1997 production analyses with calibration group setting requirements for EDM notches apparently left to the discretion of individual analysts. Overall this resulted in a marginal negative impact on ability to detect small PWSCC flaws

Review of the Westinghouse equivalent qualification document showed that a phase rotation setting of 40° for a 100%TW hole was utilized in the qualification process. This setting was estimated by NRC staff to result in the rotation setting for a 20% TW EDM notch being ~15% and the rotation setting for a 40% TW EDM notch being of the order of 23%. These values suggested that the technique, in the absence of complicating factors such as noise, would demonstrate the ability to detect small PWSCC flaws. ANTS Sheet # IP2-97-E, Revision 0, was not prepared, however, to comply with the phase rotation requirements of the equivalent qualification.

With respect to the ETSS # 96511Pwscc_ubend.doc requires setup the NRC staff estimated that use of a 10° setting for the 40% ID EDM notch would result in a rotation setting for a 20% TW EDM notch of ~2°, which could potentially negatively impact the ability to detect small PWSCC flaws.

.3 Data Analysis Guideline Review

a. Inspection Scope

The team reviewed the data analyst guidelines for use of the Plus Point probe contained in Westinghouse Procedure DAT-IP2-001, "Data Analysis Technique Procedure," Revision 0, and compared them with the guidance in Electric Power Research Institute (EPRI) "PWR Steam Generator Examination Guidelines," Revision 4. Eddy Current Probe Authorization List, Revision 1, dated May 14, 1997, which provided the specific probes and their authorized uses for the outage.

b. Observations and Issues

Separate guidance was not included with respect to the use of the medium frequency Plus Point probe for examination of low radius U-bends There was no specific guideline provided on the usage of the Plus Point probe examinations in the U-bends . The only guidance was provided in the context of the use of combination rotating probes containing a standard pancake coil (115 mils diameter), a Plus Point coil, and a high frequency shielded pancake coil (80 mils diameter). These probes were indicated by the Eddy Current Probe Authorization List, Revision 1, dated May 14, 1997, to have Appendix H (of the EPRI Guidelines) qualifications and to be authorized for use in characterization of indications in dented intersections and restricted tubes.

Other subject areas noted where strengthening of the procedure appeared warranted were:

- Inclusion of specific guidance relative to screening low frequency bobbin coil data for the presence of loose parts. The only current reference to loose parts noted during the review was in paragraph 9.2.1 which instructed the analyst to consider loose parts found in the SG when evaluating bobbin coil data.

- Development of more explicit guidance relative to data quality expectations, including measures to detect probe skipping and hanging.

.4 Analysis Training Review

a. Inspection Scope

The team endeavored to review the training provided to the data analysts in accordance with the EPRI Guidelines, Revision 4, Section 6.2 (Site-Specific Performance Demonstration states, in part, "...The actual preparation and administration of the analyst demonstration program should be approved by the utility with assistance from the ISI vendor, another vendor not involved in the SG examination, or other qualified individuals. It is important that strict rules be established during the initial preparation and future maintenance and updating of the performance demonstration so that the overall integrity of the program is maintained...."

A number of requests were made prior to and during the June 19-23, 2000, onsite inspection for the furnishing of lesson plans and practical test data that were utilized for the training and testing of the1997 refueling outage EC analysts.

On July 14, 2000, Westinghouse personnel faxed additional information to supplement test scores that had been previously provided. The received information consisted of: (a) a copy of a handwritten log for May 4-10, 1997, describing onsite activities; (b) a one page training introduction outline, (c) setup instructions for the combined Cecco-5 and bobbin probe, and (d) information regarding the contents of the practice data sets. No information was received regarding the contents of the written and practical tests. The practice data sets for the Plus Point probe (Reels 12 and 20) were noted to contain inside diameter (ID) flaws at free span locations. Due to the lack of identification at IP2 of primary water stress corrosion cracking (PWSCC) in low radius u-bends prior to 1997, data from other SGs was used for the Plus Point practice data sets.

b. Observations and Issues

The inspectors considered the incomplete status of the EC analyst training and testing information to be an indicator that the site-specific performance demonstration requirements of the EPRI Guidelines, Revision 4, had not been appropriately implemented for the 1997 refueling outage. Specifically, the submitted information, and the elapsed time in obtaining it, were not indicative of the establishment of strict rules relative to preparation, maintenance, and updating of the site-specific performance demonstration. Due to the delay in obtaining records, the degree of involvement of the licensee in the process for training and testing of EC analysts was not established.

.5 Findings

The 1997 analysis of SG low radius U-bends at IP2 was performed in accordance with the requirements of ANTS IP2-97-E, Revision 0, was not consistent with the requirements of either ETSS # 96511Pwscc_ubend.doc or the Westinghouse equivalency qualification .

While this issue had a small effect on the probability of detection of low row U-bend indications, there was no effect on the reactor coolant system integrity, in accordance with the Reactor Safety SDP Phase 1 a very low safety significance is attributed to this matter (Green) In 1997 Con Edison did not ensure the use of properly qualified EC techniques for U-bend inspection since the Plus Point EC probe was not set-up properly for use. Specifically, the proper calibration standard and phase rotation specified by the EPRI technique qualification standard were not used. In accordance with the NRC Enforcement Policy and the Reactor Safety SDP, the failure to adhere to 10 CFR 50, Criterian IX, Special Processes for EC inspection is being treated as an apparent violation since Con Edison disagreed with the violation at the exit meeting. This violation would normally be considered as a Non-Cited violation, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368).
(AV 05000247/2000-010-02, EA 00-179-1)

1RS4 Review of Con Edison Inspection Information Provided July 20, 2000, Prior to The Exit Meeting (Cornerstone - Barrier Integrity)

a. Inspection Scope

The team reviewed the information that Con Edison provided prior to the exit meeting on July 20, 2000, on five issues that Con Edison believed needed clarification based on discussions that were held during the inspection. The information which was marked DRAFT by Con Edison when presented, is included as Attachment 2.

b. Observations and Issues

Item Number 1 - noise present in the 1997 data could have masked a 70% to 100% TW defect.

Generally the information on noise dealt with R2C67 which was the defect that was found and was not relevant .

There was a discussion relative to R2C5 - here Con Edison compared the noise levels to the of the indication to the EDM notch defects and stated that the flaw depths would have been about 50% TW. Thus the disagreement on wether it was 70 - 100 % through wall was academic since TS requires a detection of 20 % TW.

Further Con Edison quoted Information Notice (IN) 97-26 seeming to indicate that the NRC knew of the detection problems and accepted them. This quote was out of context since the thrust of the IN was that there were problems with the detection of PWSCC cracks and that licensees had done different things to address the reduced POD including in-situ pressure testing.

The statement that there is no quantitative noise criteria present in 1997 is correct, and there is no quantitative noise criteria present today. However, industry has been aware of the NRC's concern and desire for such a criteria for a number of years. Draft NUREG 1477, dated June 1993, section 3.5.3 states relative to EC testing an d analysis guidelines that "*noise criteria should be incorporated that would require that a certain specified noise level not be exceeded, consistent with the objective of the inspection. Data failing to meet these criteria should be rejected and the tube should be reinspected. These criteria should be broken down into criteria for electrical noise, tube noise, and calibration standard noise*"

One of the criteria would be to compare the amplitude of the noise in the tubes being inspected to the voltage of the defects that are expected. The ratio of the standard voltage to the defect voltage should be determined for the appropriate defects. A signal-to-noise ratio of 3-to-1 would insure the detection of defects, while with sufficient training and care, a ratio of 2 or maybe even 1.5-to-1 would suffice. However, looking at the performance of the analysts in the 1997 outage shows that some caution is needed.

Item Number 2 - no corrective action taken after identifying PWSCC in R2C67.

The comment made by the licensee regarding the noise levels in R2C67 being bounded by the response of the samples used in the EPRI studies is believed irrelevant. The R2C67 flaw was indicated by the c-scan to be not associated with noise ridges.

The licensee should have been additionally sensitized by the fact that Dominion Engineering had predicted prior to 1997 that PWSCC would not be expected for several cycles in low radius U-bends. (See Section 1RS2.2 above)

Item Number 3 - the POD for the U-bent technique was invalid given the noise levels in the Indian Point 2 SGs.

Probability of Detection is based on the ability to identify flaws in a sample set. The number of samples containing flaws and the number of samples that contain no flaws are statistically significant. The significance is based on the confidence and probability originally established as an acceptable level of performance. For SG EC detection, using a EPRI qualified technique, the level established as an acceptable level of performance is an 80% POD, at 90% confidence level for flaws 60% thru wall. Please note that no technique is qualified for any flaws that are less than 60% through wall in accordance with the Appendix H of the EPRI Guideline.
Because the qualification is performed by EPRI for a generic population of SG flaws the sample set is chosen to represent the spectrum of tube conditions that are in the generic population. Because tube noise is an essential parameter that can have an affect on EC detection there should be a few tubes that are above and below the noise levels in the Indian Point s SGs. As any one essential parameter begins to dominate, however, it has an affect on the POD and confidence. If you demand a confidence of 90% be

maintained then as the number of noisy tubes encountered in a qualification sample set is increased the POD will correspondingly decrease. If the noise levels and numbers increase the POD will fall below the acceptable level of 80%.

Con Edison should have questioned the use of the generically qualified technique and possibly qualified a technique separately for the noise levels and population encountered in the Indian Point 2 SGs.

Items Number 4 and 5 - The correct calibration standard and probe setup was not used.

The use by the licensee of a general statement from the EPRI PWR Steam Generator Examination Guidelines, Revision 4, regarding method of manufacture and types of artificial flaws required to be present in calibration standards is not relevant.

The licensee has also stated "There is no further guidance provided for specific depths of the notches. Although the 1997 IP-2 calibration standards did not include a 40% ID notch, they met the requirements at that time." This posture totally ignores the obligation discussed above to use a technique that is qualified in accordance with the requirements of Appendix H of the EPRI Guidelines.

Paragraph H.4.3 in Appendix H of the EPRI Guidelines does permit use of alternative calibration methods without Requalification, if it can be demonstrated that the calibration method is equivalent to those described in the qualified acquisition technique or qualified analysis method The licensee has claimed that the setup used in 1997 met the then applicable ETSS probe setup guidelines/requirements. It was additionally stated that the 1997 Plus Point technique set phase such that residual probe motion was horizontal with the 20% ID notch at 0 to 5°. No information has been provided, to date, that would support a statement that a phase rotation setting of 0 to 5° was used for the 20% TW ID notch. The only guidance provided to the analysts by Analysis Technique Specification (ANTS) Sheet # IP2-97-E, Revision 0, was to adjust phase rotation so that probe motion was horizontal, with no instructions provided with respect to phase rotation criteria to be used for axial or circumferential notches. The absence of such instructions results essentially in delegation to the analyst for determination of setup requirements.

EC acquisition and analysis was performed in 1997, however, without demonstrating that the sole requirement of setting probe motion horizontal was equivalent to the requirements of ETSS # 96511Pwscc_ubend.doc.

c. Finding

There were no additional findings identified based on this information.

1RS5 Event Risk Significance Core Damage Frequence And Large Early Release (Cornerstone - Initiating Events and Barrier Integrity)

a. Inspection Scope

The team reviewed the actual consequences of the event and potential consequences of an SGTR given the performance finding discussed in section 1RS2.4 above. This analysis was conducted in accordance with the Reactor Safety SDP - Phase 3.

b. Risk Assessments

.1 Actual Consequences

There were no actual consequences of the February 15, 2000 event. 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 appropriately took those actions in the emergency operating procedures to trip the reactor, isolate the affected SG, and depressurize the reactor coolant system. Additionally, the necessary event mitigation systems worked properly. the release was kept to a minimum and there was no measurable activity offsite.

.2 Potential Consequences:

The following is a synopsis of the complete risk assessment developed by the NRC staff, and included as Attachment 3 to this report.

During the February 15, 2000, event the leakage from the apex crack in SG 24 tube R2C5 did not reach the full 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 1E-04 /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 5E-05/RY and 1E-04/RY.

The current guidance for assigning risk significance is contained in a draft NUREG/CR titled "Basis Document for Large Early Release Frequency (LERF) 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.

Therefore, the CDF/LERF increment associated for a SGTR event is considered to be clearly above the 1E-05/RY criterion for a "red" significance determination.

4. OTHER ACTIVITIES (OA)

4OA1 Management Meetings

.1 Exit Meeting Summary

On July 20, 2000, the team leader presented the team's overall findings to members of Con Edison management led by Mr. A. Alan Blind. At the exit meeting, Con Edison disagreed with the team's preliminary findings. Specifically, Mr. J. Baumstark the Vice President of Nuclear Engineering stated Con Edison's position that: 1) all 1997 SG inspection requirements were met; 2) the team had not identified any specific requirements, standards or guidelines that were not met; 3) no specific noise criteria existed relative to the probability of detection of flaws using EC examination; 4) the PWSCC indication was expected and no additional assessment was warranted after this discovery; 5) the root cause submitted was complete and accurate; and, 6) the NRC team's preliminary findings are not in agreement with NRC Inspection Report 50-247/97007, dated July 16, 1997.

Con Edison provide the Team with some Westinghouse proprietary information. This information wan to included in this report and the proprietary information will be returned to Con Edison following the Regulatory Conference.

PARTIAL LIST OF PERSONS CONTACTED

Con Edison:
J Groth, Chief Nuclear Officer
A. Blind, Vice President
J. Baumstark, Vice President, Nuclear Power Engineering
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Westinghouse:
D. Adomonis
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J. Maris

ITEMS OPENED AND CLOSED

Opened

LIST OF DOCUMENTS REVIEWED

Industry Steam Generator Guidance

EPRI SG Inspection Guidance

Rev. 4 - date June 1996

Rev. 5 - date September 1997

Performance Demonstration data base ETTS #965121 Pwscc_ubend.doc, dated May 1996

EPRI PWSCC Predication methods

NEI SG Program Guidelines 97-06, dated December 1997

NRC Generic Input

Reg Guide 1.83, Rev 1, dated July 1975

Draft Reg Guide 1.121, PWR Steam Generator Tube Plugging Limits, August 1976

Draft NUREG 1477 - Voltage -Based Plugging Criteria for SG Tubes - Dated June 1993

Generic Letter 95-03: Circumferential Cracking of SG Tubes, dated April 28, 1995

Generic Letter 95-05 Voltage Based Repair Criteria for Westinghouse Steam Generators Tubes Affected by Outside Diameter Stress Corrosion Cracking (notebook)

Information Notice 96-38: Results of SG Tube Examinations, Dated June 21, 1996

SECY 98-248: Proposed GL 98-XX SG Tube Integrity, dated October 28, 1998

Draft Reg Guide 1074 - Steam Generator Tube Integrity, dated December 1998

IN 97-26 Degradation in Small Radius U-bends , May 19, 1997

EGM 96-003, Updated June 2000 SG Tube Inspections

NRC Correspondence:

May 29, 1997 Proposed SG Inspection plan approval 1997 - Refueling Outage -

March 14, 2000 - RAI Re: Proposed SG Tube Examination Program - six questions.

March 20, 2000 - Lessons -Learned Evaluation - Includes attachments with RES response to Request

March 24, 2000 - RAI Re: Proposed SG Examination Program - 21 questions.

April 28, 2000 - Notice for May 3, 2000, meeting - 17 questions

Con Edison:

1997 IP2 Spring 1997 Inspection Evaluation - Westinghouse to Con Ed with CMOA as an attachment, dated July 24, 1997

IP-2 Steam Generator Handbook, through 1997 Outage

IP-2 Steam Generator Status Report, dated April 22, 1998, based on the results of 1997 outage

June 14, 1995 - Inservice Tube Examination 1995 Refueling Outage - TS 4.13.C.2 report

January 10, 1997 - RAI - SG Tube Acceptance Criteria TS Amendment Request. (Resident Office File)

February 7, 1997 - 1997 SG Inspection Plan

April 24, 1997 - Outage Inspection Plan - from NRC meeting

May 7, 1997 - Comparison of Cecco-5 and +point performance - (do not have a copy)

July 24, 1997 - Response to staff questions (residents office file)

July 29, 1997 - SG Tube Inservice Examination 1997 Refueling Outage - TS 4.13.C.2 Submital (Resident Office file)
December 7, 1998 - Proposed Amendment to TS Regarding SG Tube Inservice Inspection Frequency
May 12, 1999 - Response to RAI - Proposed Amendment to TS Regarding SG Tube ISI Frequency
2000 - Outage Inspection Plan
April 14, 2000 - Root caused Evaluation
April 18, 2000 - Answered Questions 2,7,17 from March 24
May 15, 2000 - Response to Request for Additional Information - proposed SG Tube Examination Program - EPRI Appendix K Report.
June 13, 2000- Response to Staff Question on Root cause Evaluation
June 15, 2000 - Response to the Staff's Questions Regarding the Root Cause Evaluation for SG Tube Rupture
June 15, 2000 - Response to RAI - Proposed SG Examination Program - NRC letters March 14 and 24, 2000
June 16, 2000 - Response RAI
June 19, 2000 - Response to RAI
June 19, 2000 - Response to RAI
June 20, 2000 - Response to RAI
LERs
    March 17, 2000 - 2000-001 - Manual Trip following SGTR
    April 24, 2000 - 2000-003 - SG 21 and 24 in C-3
Purchase Spec - MPE-72217 - Rev 10, Dated Dec 17 , 1996 - ET examination of SG tubes
Station Admin Order - 180 Administrative SG Program Plan, Rev 0 April 2000
Strategic Water (secondary) Chemistry Plan, Rev 1 March 1999 (Resident Office File)
Primary to Secondary leakage, IPC-A-110, Dated 6/4/97
Corrective Action Program
    1997 CR June 122, 1997- #2282 - IN 97-26
    March 9, 2000 - #1623 -Use of probes bigger than 0.610inches after 0.700" could not be passed
    March 20, 2000 - #1939 - SG 21 1 tube >40 % and SG 24 three tubes >40% re-review of 1997 data.
    March 23, 2000 - #2049 - SG 21 and 24 - C3
QA Surveillances
    SR 97-056 - May 12, 1997
    SR-97-105, May 21, 1997
    SR 97-106, Mat 24, 1997
QA Audits
    95-8-01-H, dated 8/31/95
    97-01-H, dated November 7, 1997
    98-01-D , dated 9/25/1998 - Chemistry Surveillance - includes the CRs generated based on the Audit.
    00-01-H, draft dated 6/16/00 - SG Inspection and maintenance

Vendor Audits
2000 - Trip Report and Associated CRs - Paul Deeds
924-34, dated April 29, 1992 - Based on NUPIC Audit
941-13, dated January 31, 1994 - Class A Vendor Evaluation
953-14, dated March 20, 1995 - Review of West. NDE Certifications
Independent QDA
Jan. 13, 1997 contact ConEd to CoreStar - IQDA 1997 Outage
Letter dated May 29, 1997 - CoreStar to Con ED
March 3, 2000 - contract Con Ed to ABB/CE- IQDA services.
Eddy Current Information
Cal Standard used in 1997
1997 Cal Groups
Reel 058 2110 - 2359, with the beginning of reel standard
Reel 060 0243 - 0613, with the beginning and end of reel standard.
1997 ANTS

Westinghouse Inputs
Team Generator Primary Side Service Module - Contract For 1997 outage
SG Tube EC Inspection Techniques
Documentation of Appendix H Compliance and Equivalency DDM-96-009
Eddy Current Low Row U-bend Examination Equivalency Qualification
May 14, 1997 - Eddy Current Probe Authorization List Rev 1
May 16, 2000 letter from Westinghouse to ConEd - Use of Appendix H Qualification Techniques at IP2 Spring Inspection - in notebook
1997 Examination Technique Specification Sheets
Analyst Training
Steam Gen Maintenance Services Memo - Copy of log book and Training schedule and information
Site Specific Test Scores
T-list & Summaries from Training & Testing Optical
Corrective Action Program
CAR 00-1076 - Missed indications in previous outages - SG 24 R34C51 in sludge pile above TTS and R2C69 U-bend
CAR 00-1075 - inconsistent implementation of analyst performance tracking.
CAR 00-1113 - tubes left off the plugging list
Analyst Procedures for assessing EC Data
1997 DAT -IP2-001 Rev 0, date 4/28/87
2000 DAT-IP2-001, Rev 0 with Field Change 001-003, dated 4/1/00
2000 - Probe Authorization sheet and Acquisition Technique Specification Sheets
Assessment of NDE Personnel Qualification Assessment - dated May 17, 2000

Dominion Engineering
SG Life Prediction Analysis
DEI- 442, Draft - Dated October 1995
DEI - 519 - Draft - Dated December 1997

Update to DEI 519 - draft - dated April 10, 2000

NRC/Con Edison/Westinghouse Meetings

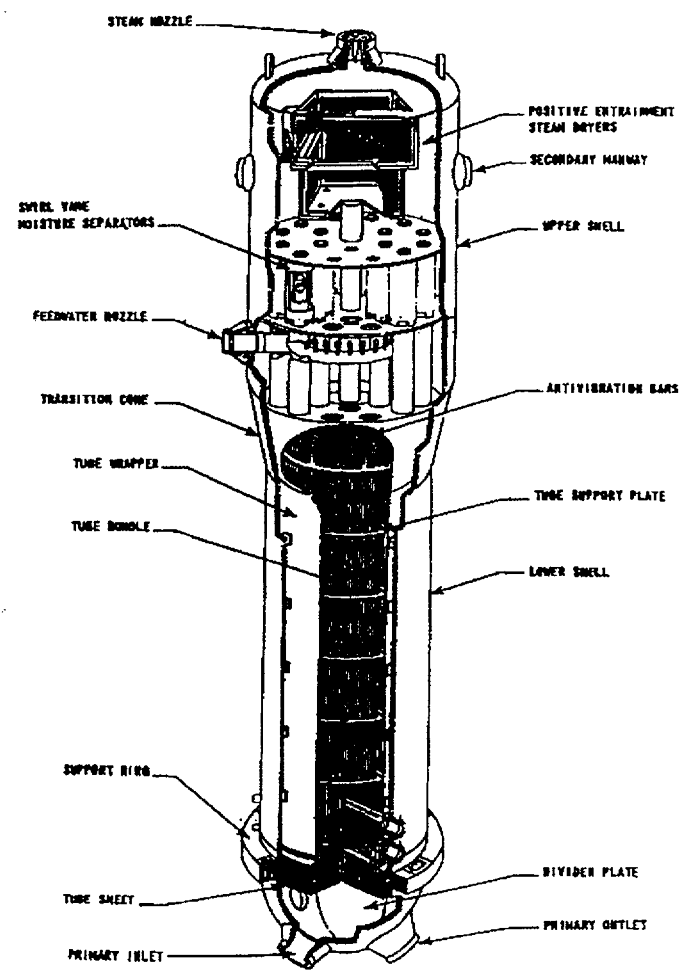
May 3, 2000 - Headquarters Handouts
Low Row U-bend Exams - ConEd
U-bend PWSCC Susceptibility Investigation - Altran
Root cause analysis Report Overview - ConEd
Condition Monitoring Operational Assessment Plan - Westinghouse
May 25, 2000 - Waltz Mill Handout (In Notebook)
2R14 SG Inspection - Westinghouse
June 6, 2000 - Waltz Mill
CMOA POD and Depth Sizing of PWSCC Indications - Westinghouse
2R14 SG Inspection - Westinghouse

LIST OF ACRONYMS USED

AIT	Augmented Inspection Team
CCR	Central Control Room
CFR	Code of Federal Regulations
Con Edison	Consolidated Edison Company of New York, Inc.
CR	Condition Report
CTS	Communication to Staff
EOP	Emergency Operating Procedure
GT	Gas Turbine
HPSD	High Pressure Steam Dump
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OD	Operability Determination
OWA	Operator Workarounds
PARS	Publicly Available Records
RCS	Reactor Coolant System
RCP	Reactor Cooling Pump
RES	Request for Engineering Services
RHR	Residual Heat Removal
SAO	Station Administrative Order
SE	Safety Evaluation
SG	Steam Generator
SGTF	Steam Generator Tube Failure
SI	Safety Injection
SL-1	Significance Level One
SOP	Standard Operating Procedure
TFC	Temporary Facility Change
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

REFERENCED FIGURES

Figure 1 - Westinghouse Model 44 Steam Generator



Mr. A. Alan Blind 37

Figures 2 thru 4 - EDDY CURRENT EXAMINATION

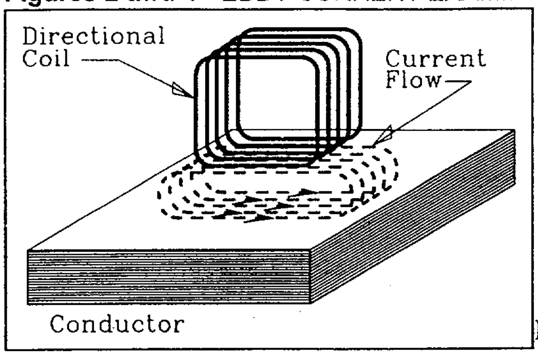


Figure 2 Directional pancake probe

Figure 3 Plus Point probe

Figure 4 Difference signal from axial and circumferential coils are amplified

Figure 5 R2C5C-scan with1997 phase setting.
Figures 5 thru 8 - Eddy Current Inspection -Tube R2C5 in SG24

Figure 6 Lissajous of defect with 1997 phase setting.

Figure 7 Noise signal that runs the length of the U-bend.

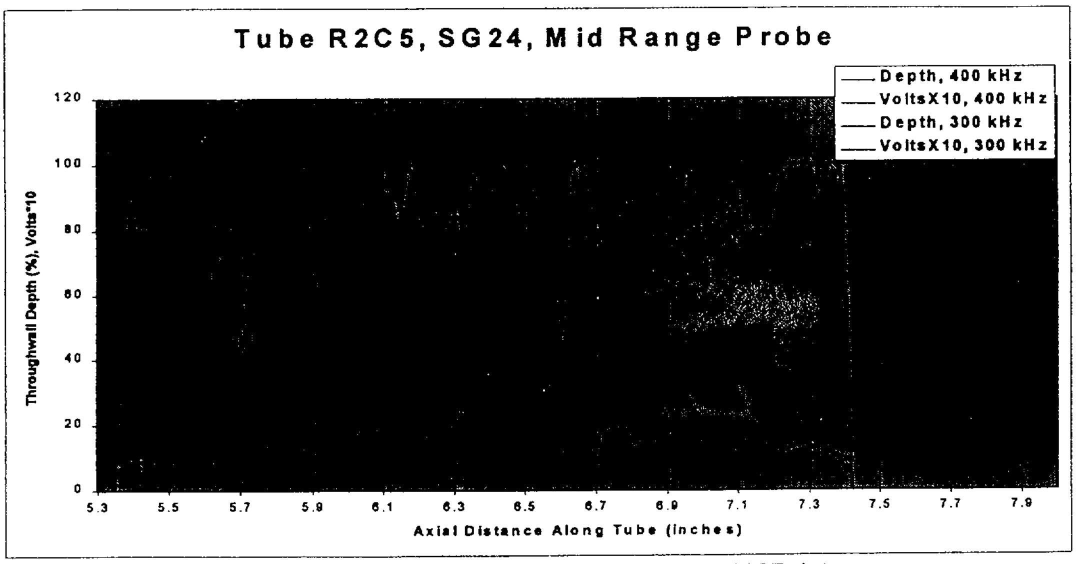


Figure 8 - Contour of the crack in tube R2C5 in SG24 using the 1997 data from the mid-range plus-point probe.

Figures 9 thru 11 - Eddy Current Inspection - Tube R2C69 in SG 24

Figure 9 - 1997Mid- range scan .

Figure 10 - 2000 mid-range scan

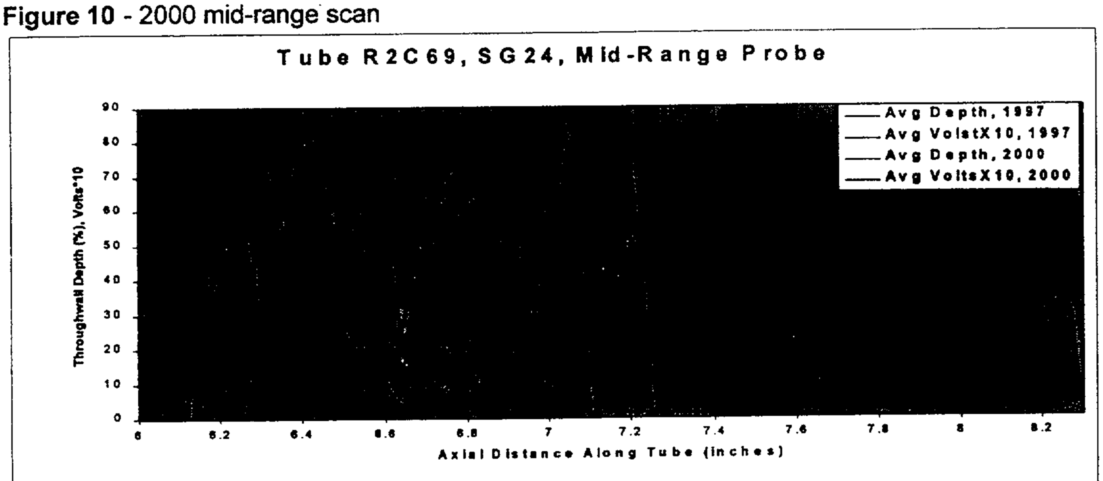


Figure 11 - Profile of Growth between 1997 and 2000

Mr. A. Alan Blind 42

Figures 12 thru -14 - Eddy Current Inspection Tube R2C72 in SG 24

Figure 13 - 2000 scan

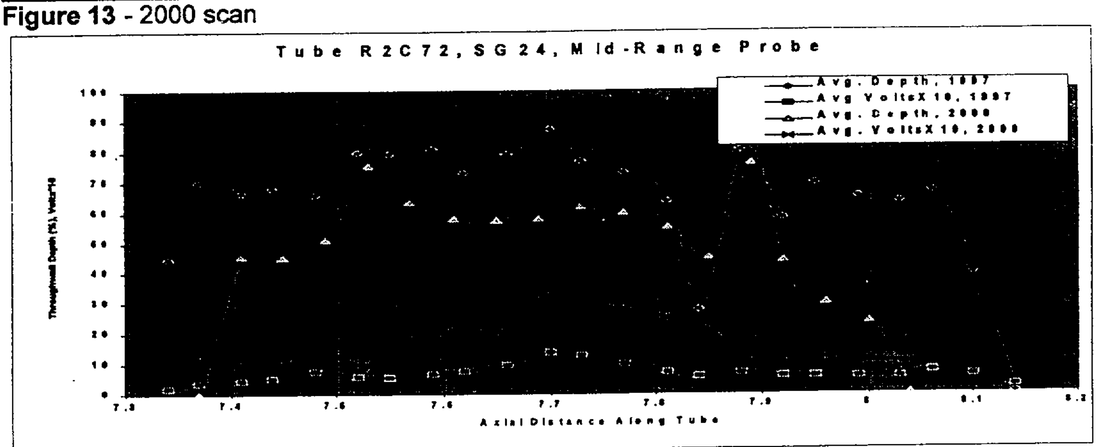


Figure 14 Profile of crack growth of tube R2 C72 of SG24 between 1997 and 2000

Figure 12 - 1997 scan

Mr. A. Alan Blind 43

Figure 16 - 2000 scan

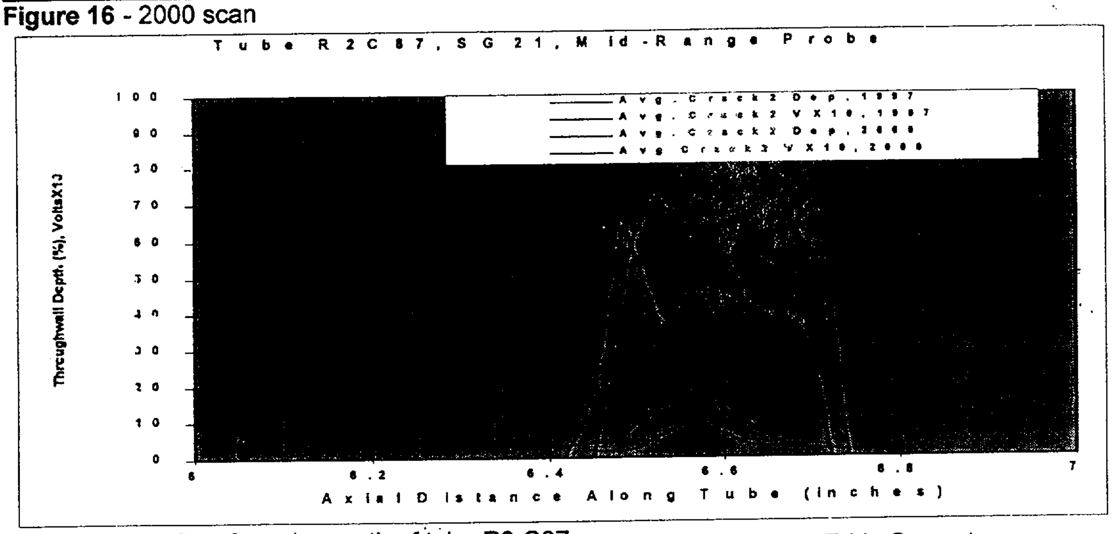


Figure 17- Profile of crack growth of tube R2 C87. **Figures 15 thru 17** - Eddy Current Inspection- Tube R2C87 in SG 21

Figure 15 - 1997 scan

Mr. A. Alan Blind 45

Figures 18 and 19 - Eddy Current Inspection- Tube R2C67 in SG 24

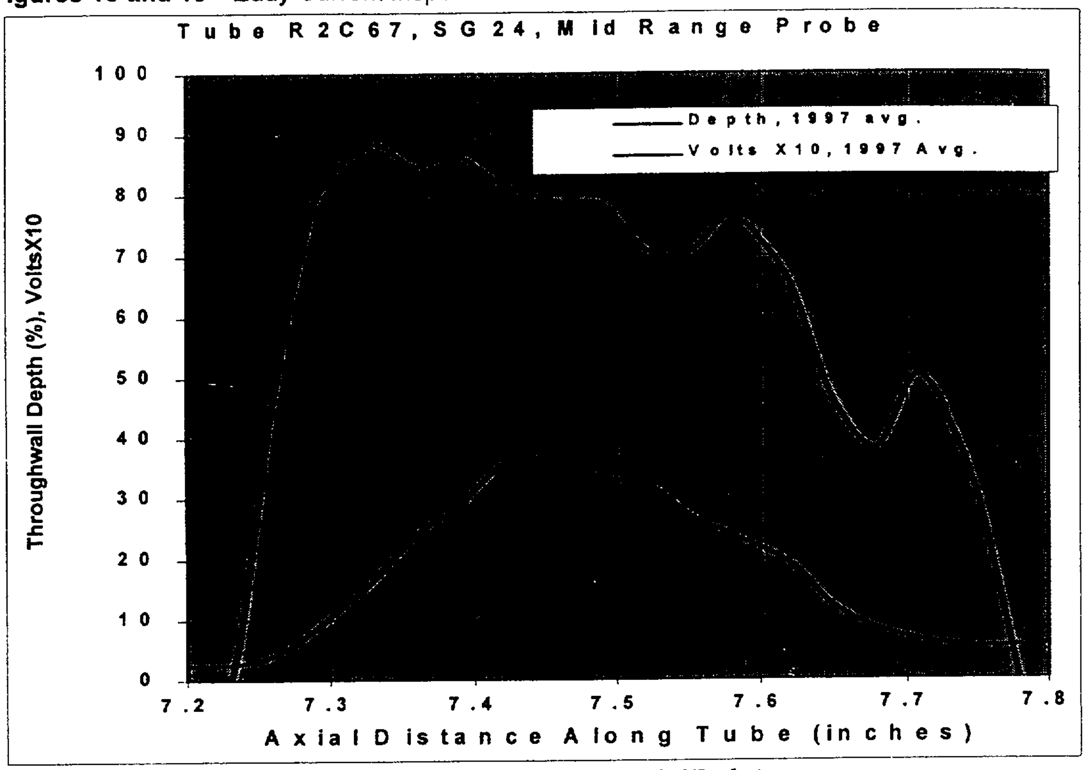


Figure 19 - Depth and voltage profile of the crack in tube 2-67 of steam generator 24 located in 1997.

Figure 18 - Crack in tube 2-67 of steam generator 24, found in 1997.

Mr. A. Alan Blind

ATTACHMENT 1
NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
• Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness	• Occupational • Public	• Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate

Mr. A. Alan Blind

protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Mr. A. Alan Blind

ATTACHMENT 2

July 20, 2000, Con Edison Presentation to the Inspection Team marked DRAFT by Con Edison

During the course of the NRC Special Inspection Team assessment of the Indian Point Unit 2 1997 steam generator inspection, the team raised a number of questions relating to the program. Additional clarification on five of the items is provided below.

Item Number 1

Con Ed did not recognize nor evaluate potential noise in the eddy current test (ECT) data. This is important as the noise could mask a 70% to 100% through-wall indication.

Discussion

In 1997 a single U-bend indication was detected in SG 24 Row 2 Column 67. At the time, a depth of 50% through-wall was estimated using a +Point probe and the tube was repaired by plugging. The indication had a signal to noise ratio of approximately 3 to 1 and the noise levels did not appear to differ appreciably from row 1 and 2 U -bend data from other plants. The inspection method used was the most advanced technique available in the industry and it appeared to us that the technique was performing as expected. Based on the information available in 1997, there was no indication that flaws between 70% and 100% through-wall would be missed due to noise. Also, there was no data available which would establish a correlation between signal amplitude and depth. It also should be noted that in 1997 there were no industry criteria to evaluate noise in a quantitative manner.

In response to the NRC's question, a current review of the 1997 data was conducted. The review of this data shows that the indication in R2 C67 had an amplitude of 3.11 volts while the background noise level was 1.04 volts peak to peak and 0.44 volts vertical maximum. This data was compared to the EPRI data for technique 96511 and the response from the calibration standards. It should be noted that the EPRI qualification data set consisted primarily of EDM notches placed in row 1 U -bend samples. It is recognized that EDM notches yield larger signal amplitudes for a given depth than PWSCC. In the absence of data from partial through -wall PWSCC specimens, the response of the calibration notches was benchmarked along with the noise levels present in the EPRI samples. The peak to peak and vertical maximum voltages are listed in the table below. All measurements were made from the 300 kHz component.

CALIBRATION STANDARD USED IN ETSS 96511

AXIAL EDM SLOTS	VOLTS PEAK to PEAK	VOLTS VERTICAL MAX
100 %	20.00	9.39
80 ID	5.40	1.96
60 ID	3.84	1.11
40 ID	2.17	0.44
20 ID	0.66	0.12

Mr. A. Alan Blind

This data suggests that, given the noise levels in R 2 C 67, flaws ≥ 40% would be detectable (i.e. signal to noise for a ≥ 40% flaw is ≥ 1 to 1).

The 1997 noise level in SG 24 Row 2 Column 5 was also evaluated. This data shows a peak to peak amplitude of 1.63 volts and a vertical maximum amplitude of 0.98 volts. The result from this assessment suggests that flaw depths of approximately 50% TW and less may not be detected (signal to noise < 1 to 1). This observation is consistent with NRC IN 97-26, "Degradation in Small Radius U-bend Regions of Steam Generator Tubes" issued May 19, 1997 which states:

"There continues to be an absence of pulled tube information to confirm that the detection threshold for these cracks is better than 40 or 50-percent through wall. In addition, available inspection techniques are not capable of reliably sizing crack depths and, for this reason, it has been industry's practice to "plug on detection" U -bend indications that are found."

The table below lists the EPRI samples, their noise levels, and the depth of the flaws in the U-bend.

ETSS 96511 FLAW MATRIX

SAMPLE	NOISE VPP	NOISE VM	DEPTH	DEPTH	DEPTH
Z5324	0.72	0.21	41	27	32
TVA-1	0.78	0.27	45	44	44
TVA-13	0.75	0.20	55	55	55
TVA-23	0.70	0.16	55	58	54
1019-I	1.26	0.29	40		
1019-III	1.39	0.61	50		
1019-IV	1.60	0.56	60		
1019-UB-I	1.22	0.41	60		
Z-5300	1.71	0.52	44	100	
TSL-126	1.19	0.19	>40		
TSL-15	1.33	0.16	>40		
TSL-2	1.03	0.20	100		
TSL-10	0.66	0.17	>40		
TSL-113	1.04	0.15	42	42	
TSL-115	1.27	0.16	62	62	
AVERAGE	1.11	0.28	N/A	N/A	N/A

The data shows that some samples had a noise level greater than that observed in R 2 C 67, while other samples were less. Specifically, 9 of 15 samples were ≥ 1.04 volts peak to peak and 3 of 15 Samples were ≥ 0.44 volts vertical maximum.

Mr. A. Alan Blind

We would conclude that, based on the information available in 1997 reviewed at the time of the 1997 inspection without the benefit of the passage of time or 2000 inspection results, there was no indication that flaws between 70% and 100% through -wall would be missed due to noise.
Data quality criteria were not in place in 1997 across the industry, and guidance was only developed following the current evaluation of R2C5. There were no criteria and no database to form a postulate that the noise effects could mask a flaw such as that present in R2C5 in 1997. It is very doubtful that any review in 1997 of the finding of a single apex flaw in row 2 at Indian Point-2 would have rationally led to consideration of a potential imminent flaw. Hindsight is very enlightening, but any review of 1997 evaluations must be put into the knowledge basis of 1997 rather than after the knowledge gained from the R2C5 evaluation.

Item Number 2

There was no specific corrective action in response to a new and significant defect at the apex of R 2 C 67. The flaw had been sized at 50% through -wall. ConEd should have recognized that corrective action was required in accordance with 10CFR Part 50 Appendix B.

Discussion

The corrective action taken in response to the detection of the R2C67 PWSCC indication was appropriate.

In 1997 Revision 4 of the EPRI Guidelines required the use of a qualified technique. We used such a qualified technique during the 1997 inspection – ETSS 96511. Moreover, the ECT response to R2 C67 was typical of those in the training materials, indicating to us that this technique was performing as was expected. A review of the EPRI ETSS shows that the noise levels in R2 C67 were bounded by the response of the samples used in the EPRI study.

The indication found in 1997 was based on the first +Point inspection of the IP2 low row U-bends following prior inspections with the bobbin coil. The first +Point inspections typically lead to an inspection transient (step increase in numbers of indications). The finding of a single U-bend indication in the +Point inspection after prior bobbin coil inspections was not considered an unusual event after about 16 EPFY of operation. In contrast, the Surry-2 tube rupture occurred in a row 1 tube after about 2 EFPY of operation when denting progression was very active with hourglassing progressing to flow slot closure, which exceeds that at the top TSP at Indian Point-2.

Based on the information available to us in 1997, reviewed at the time of the 1997 inspection without the benefit of the passage of time or 2000 inspection results, no additional corrective actions would have been required in response to the indication identified in R2 C67.

Mr. A. Alan Blind

From a programmatic point of view, during the 1997 inspection, additional analyst training was provided whenever the inspection findings were unexpected. Discovery of ODSCC/IGA in the tubesheet crevice region during the course of the Indian Point 2 1997 inspection resulted in additional analyst training and re- evaluation of data in the tubesheet crevice region. This was done as these indications were not considered "typical flaw responses" and differed, somewhat, from the materials the analysts had been trained on. This was not the case, however, with the discovery of the R2 C67 indication.

All elements of the licensee and vendor quality assurance programs were complied with in 1997, and hence the requirements of 10CFR Part 50, Appendix B were satisfied.

Item Number 3

Given that some of the samples used in the EPRI study had noise levels *above*, while others had noise levels *below* those observed in R2 C67, we should not have used the POD listed in the technique.

Response

As discussed previously, the noise level in R2 C67 was bounded by the EPRI study. In addition, the analyst experience was that similar noise levels existed at other plants that were using the same ECT technique. In 1997 there was no Industry guidance which would have directed us, or suggested that we use a POD other than that listed in the ETSS. Moreover, there are no NRC regulations, requirements or technical advisories that contain such direction or guidance.

Item Number 4

The correct calibration standards were not used.

Discussion

The calibration standards which were used in 1997 met industry standards and followed the then current EPRI guidance – EPRI PWR Steam Generator Examination Guidelines, Rev. 4.

EPRI PWR Steam Generator Examination Guideline – Revision 4 requirements for rotating probes were as follows:

Electro-discharge machining (EDM) and laser-machined notch standards are typically used to establish setup conditions for rotating probe technology. The notches should be of:

- *both axial and circumferential orientation, and*
- *standard lengths and depths on the OD and ID.*

Mr. A. Alan Blind

There is no further guidance provided for specific depths of the notches. Although the 1997 IP-2 calibration standards did not include a 40% ID notch, they met the requirements at that time.

Item Number 5

The probe setup was incorrect. Probe motion was set to horizontal.

Discussion
The setup used in 1997 met the then applicable ETSS probe setup guidelines/requirements.

ETSS 96511 establishes phase (10 Degrees) on the 40% ID notch. The plus point technique, as applied at IP-2 in 1997, set phase such that residual probe motion was horizontal with the 20% ID notch at 0 to 5 degrees. The calibration standard used in the EPRI ETSS 96511 qualification did include a 40% ID notch. A review of this data shows that when the 40% ID notch is set at 10 degrees the resultant phase for the 20% notch is approximately 1 degree with residual from probe motion horizontal.

The EPRI Revision 5 standard used at Indian Point 2 during the 2000 inspections does have a 40% ID flaw, and this signal was used to calibrate the analysis software as specified in ETSS-96511. The site specific technique sheet, ANTS IP2-00-E, specifies 15 degrees for the 40% notch, which is more conservative than the 10 degree EPRI ETSS requirement. Review of the 1997 data for R2C5 using the mid-range probe and the 2000 setup with the phase rotation set at 15 degrees, also did not show a flaw.