

From: Wayne Schmidt
To: A. Randolph Blough, Brian Holian, Daniel Holody, David Lew, Diane Screnci,
Gregory Cranston, Hubert J. Miller, James Wiggins(...)
Date: Fri, Aug 11, 2000 4:41 PM
Subject: IP2 SG Report

Attached is the report - Dave and I plus, the team members have reviewed it and incorporated comments. This is a large file - due to the figures.

I still need to work on the Acronyms, open item page, and the list of documents - to be complete on Monday. I will be getting some help from Caius Dodd in making the figures more legible and not distorted - but they are all there.

We are still shooting for issuance by next Friday 8/18. To accomplish this we would like to get Regional Comments back by noon on Monday with hope of getting this down to NRR COB Monday of first thing Tuesday.

A draft violation is attached - it is almost directly from the Executive Summary.

If there are any pressing questions please feel free to call me over the weekend.

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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 REPORT 05000247/2000-010
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 the small radius low row U-bend tubes (rows 1 thru 4) were not identified.

During the 1997 steam generator inspections, a new and significant degradation mechanism, i.e., primary water stress corrosion cracking (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 of low row U-bend PWSCC and of the impact on the 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 the associated tubes were left in service until the failure of one of these tubes occurred on February 15, 2000.

The report identifies that Con Edison failed to evaluate and take action to correct and account

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for signal noise, and to adjusted or modify inspection methods and analysis to account for the anomalies and other new conditions encountered. This failure is 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. Using the Reactor Safety Significance Determination Process (SDP) we characterized this issue as an apparent significant finding that was preliminarily determined to be Red. While the risk associated with the event was low to moderate due to the actual leak rate being less than that of a steam generator tube rupture (SGTR), the SDP determined that the finding was of potentially of high safety significance because of the increased risk of a SGTR, during a year of reactor operation, with the degraded tubes in service.

As discussed with Mr. John McCann of your staff, we have scheduled a Regulatory Conference for September 14, 2000, in the Region I office to discuss your evaluation and any differences with the NRC evaluation prior to our final significance determination on the 10 CFR 50, Appendix B, Criterion XVI, issue discussed above.

The NRC also identified an issue involving improper calibration and set-up of the eddy current technique used to examine the U-bend areas of low row tubes. Using the Reactor Safety (SDP) we determined that this issues to be of very low safety significance (Green). The issue involved a violation of NRC requirements, but because of the very low safety significance, this violation would normally not be cited. However, you disagreed with the violation at the exit meeting. We will be prepared to discuss this issue during the September 14, 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

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Enclosure: Inspection Report 05000247/2000-010

cc w/encl:

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U.S. NUCLEAR REGULATORY COMMISSION

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,
Division of Reactor Safety (DRS)

Team Leader: Wayne L. Schmidt, Senior Reactor Inspector, DRS

Inspectors: E. Murphy, Senior Engineer, Materials and Chemical Engineering Branch,
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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

IR 05000247-00-010, March 7 thru July 20, 2000; Consolidated Edison of New York, Inc.; Indian Point Unit 2; Special Team Inspection reviewing the causes for the February 15, 2000, steam generator tube failure, in accordance with NRC Inspection Procedure 93812. Findings in steam generator eddy current examination program corrective actions and technique qualification.

The Team members included personnel from the Office of Nuclear Reactor Regulation and Region I, and NRC-contracted specialists in steam generator eddy current testing. This inspection identified one apparent green and one apparent red issue, both of which will be the subject of a Regulatory Conference. The significance of the issues is indicated by their color (green, white, yellow and red) and was determined by the using the Reactor Safety Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (see Attachment 1). The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED).

This special inspection focused on the caused of the SGTF. The NRC also reviewed Con Edison's performance in response to the event, with an Augmented Inspection Team (AIT), Inspection Report 05000247/2000-003, and a AIT Followup Team, Inspection Report 05000247/2000-007.

REACTOR SAFETY

Cornerstone: Barrier Integrity and Initiating Events

- **TBD - Potential Green** - During the 1997 refueling outage the U-bend mid-range Plus Point eddy current test (ECT) probe, used for steam generator (SG) tube inspection, was not properly set-up 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 Electric Power Research Institute qualified technique sheet. This issue involved matters that had very low risk significance, because there was no effect on the reactor coolant system integrity. Although this violation would normally be non-cited, it is described as potential violation of 10 CFR 50, Appendix B, Criterion IX, Special Processes, because Con Edison disagreed with the team's characterization of this issue during the exit meeting (Section IR3.1)

CROSSCUTTING ISSUES

Corrective Action

- **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 SGs. However, as of

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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 ECT of SGs 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 tube support plate (TSP) of SG tubes when restrictions were encountered as ECT 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 TSP, which, in turn, indicated additional PWSCC stresses on the low row U-bend tubes.
3. significant ECT signal interference (noise) was encountered in the data obtained during the actual ECT 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 adjust or modify inspection and analysis methods during the inspections process to account for the anomalies and other new conditions encountered. As a result, four tubes with indications were not promptly identified in the 1997 outage and were left in service until the failure of one of these tubes occurred on February 15, 2000.

These matters had a high safety significance with a significant reduction in safety margin since the potential for a SG 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 40A1.1)

4. The team concluded that Con Edison's root cause analysis for the SGTF, dated April 14, 2000, did not identify and address significant SG inspection program performances issues as they related to the failure of tube R2C5 in SG 24 on February 15, 2000. While the root cause analysis attributed the SGTR to a flaw that was obscured by ECT signal noise, it did not identify, nor address, deficiencies in the processes and practices during the 1997 SG inspection.

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Report Details

BACKGROUND

Summary of Plant Event

Following the steam generator tube failure (SGTF) 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 steam generator (SG) 24. This small radius (low row) tube had cracked at the apex of the U-bend due to primary water stress corrosion cracking (PWSCC). Con Edison conducted an eddy current test (ECT) examination of the SG tubes and conducted visual inspections of the secondary side of the SGs. During these ECT 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 before restarting the plant in accordance with the technical specifications (TSs). At the conclusion of the inspection, the unit remained in cold shutdown pending NRC restart approval.

The NRC previously 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 system (RCS) to the secondary water. This heat causes the secondary water to boil, and the resulting steam is used to turn the turbine which turns the electrical generator. Figure 1 shows a Westinghouse Model 44 SG, like those installed at Indian Point 2. The four SGs are identified as SG 21 thru SG 24.

Each SG was built with 3,260 tubes. These tubes have the reactor coolant running inside them with 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 identified by its 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 row 1 tubes were removed from service prior to initial operation. 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 (TSPs). In each TSP, there are holes cut to allow water/steam flow around the tubes. Also there are six evenly spaced flow slots that run 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 about 3 inches wide. The U-bend area is located above the upper TSP.

During operation, the RCS is pressurized to approximately 2235 psig to prevent boiling of the reactor coolant. Normal SG pressure varies with plant load between approximately 100 psig at no load to approximately 700 psig at 100% power. The pressure difference between the RCS and the SGs can cause leakage from radioactive RCS water to the secondary side of the SG. This is referred to as primary to secondary leakage. (See Technical Specifications)

Technical Specifications

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 TS require that a representative sample of the SG tubes be examined using ECT 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 normally 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.

The primary to secondary leakage rate is limited by the plant technical specifications to 0.3 gallons per minute (gpm). Primary to secondary leakage can result from several sources, including 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).

The TS also contain a requirement to report significant deformation of the upper TSP (hour-glassing) since it can have a significant effect on the integrity of tubes beyond row 1. (See Applicable Steam Generator Degradation Mechanisms below)

Eddy Current Examination Technique

ECT is a method of inspecting SG tubes by passing a probe that generates an electromagnetic field and the probe senses the disturbance of the field due to defects in the tubing. The technique is based on the principle of electromagnetic impedance of a coil in an alternating current 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 as variations in the voltage across the coil and by the degree that the voltage and current are out-of-phase (referred to as the phase angle).

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

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 ECT probes can classify defects as axial cracks, circumferential cracks or both.

The frequency of the alternating current sent to the probe and the size of the probe affect 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 the voltage 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 a multifrequency probe, operating at 10, 100, 300, and 400 kHz.

Noise in ECT 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 ratio, the easier it is to detect a defect.

The Electric Power Research Institute (EPRI) qualifies the ECT techniques for use during SG inspections. This qualification includes the verification that the technique can identify known defects with a probability of detection (POD) of greater than 80% with a 90% confidence. The POD of flaws is calculated based on the detectability in a sample of tubes with known flaws (defects). These defects may be actual flaws in tubes removed from SGs across the industry or man-made notches in tubes using laser-machining or a process called electro-discharge machining (EDM). 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.

The ECT techniques are calibrated, as with any measurement instrument, to known calibration standards during their use. These calibration standards include notches of known depth and length against which the analyst calibrates the instrument.

ECT information may be displayed in numerous forms, several of which are shown in the figures in the back of the report. During an ECT examination, the data and the analyses conducted are electronically stored and maintained as part of the plant inspection record. The c-scan plot is a topographical picture, as if the tube was split and laid out flat, of the changes in probe impedance. The signal shows a voltage reading that has been adjusted for phase angle (referred to as the vertical component). The strip chart is a look at the high and low values shown on the c-scan, as if the c-scan was viewed from the side. The lissajous is a graphical view of the voltage and phase angle effects at a specific point in the tube.

ECT signals may be affected by deposits that collect on the OD surface of the tubes. Different types of flaws within the tube wall, deposits outside the tube, and SG structures, such as TSPs and the tube roll transitions, all have an effect on the ECT signal and have a characteristic lissajous signal.

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Through extensive training and qualification, the ECT 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 OD, it is referred to as ODSCC.

Based on the crack characteristics, a PWSCC defect (and a SCC defect in general) may not yield an ECT signal of the same amplitude of a similarly sized calibration standard EDM defect. Further as stated in NRC Information Notice 97-16, *"There continues to be an absence of pulled tube information to confirm that the detection threshold for these cracks is better than 40 - 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."*

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 low row tubes, and in the TSP intersections especially if the tubes are dented.

Denting of the tubes is the direct result of secondary side corrosion of the carbon steel TSP. 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, the tube, as it tries to expand during heat up, 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 ECT 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 TSP. In the area of the flow slots where the structural resistance is low enough, deformation and/or cracking of the TSP can occur. If this happens on both sides of the flow slot, the sides of flow slot are forced inward at the middle, causing the previously rectangular shaped flow opening to develop the shape of an hour-glass. This is referred to as hour-glassing. In the low row U-bends, PWSCC is significantly more likely to occur if hour-glassing forces the tube legs closer together, since a small movement of the tube legs will concentrate sufficient tensile stress at the apex of the U-bend.

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Steam Generator History

In a review of plant history, the team found that the Indian Point 2 SGs 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. To allow inspection of the upper TSP region Con Edison, prior to the 1997 outage, installed inspection ports on SG 21 and SG 23.

Due to the composition of some secondary system components at Indian Point 2, deposits on secondary wall of the tubes contain hematite (Fe_2O_3), interspersed with metallic copper. These deposits generally do not promote severe tube corrosion. However, they can have an effect of increasing the noise in an ECT 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 RFO 13 refueling outage. The SG inspection plan included a 100% Plus Point probe examination of the low row U-bends. The examination, completed in June 1997, identified 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 performed. Also during this examination, Con Edison identified the first instances of 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, 2000, 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 between 400 and 600 gpm.

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1. REACTOR SAFETY - Cornerstones: Initiating Events, Barrier Integrity

**1R1 Initial Review of Eddy Current Data Following The Tube Failure
(Cornerstone - Barrier Integrity)**

a. Inspection Scope

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

b. Observations and Issues:

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

The year 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 noise, 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 the ECT Analysis Technique Specification Sheet # IP2-97-E (ANTS # IP2-97-E), Rev. 0, dated 5/8/97, that was used during the 1997 outage, and found that it had been set-up incorrectly not in accordance with the ERPI qualification of the probe Examination Technique Specification Sheet # 96511Pwscc_ubend.doc (ETTS # 96511), dated May 1996 (see Section 1R3.1). Con Edison and its contractor subsequently corrected this problem during the reevaluation phase of stored 1997 data.

Initially for the 2000 outage the U-bend Plus Point phase set-up, ANTS # IP2-00, Rev. 1, dated 2/27/00, was not properly set-up, and had not changed from the erroneous set-up in 1997. In early March 2000, the Con Edison revised ANTS # IP2-00-E, by Rev. 2, dated 3/4/00, to conform with ETTS # 96511. All the year 2000 U-bend examinations that had previously been completed were repeated using the corrected set-up.

c. Findings

The team identified no findings during this review. The U-bend probe calibration and set-up issue is discussed relative to 1997 performance in Section 1R3.1.

**1R2 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 ECT data collected on eight tubes that were identified as possibly having detectable flaws in 1997, including tube R2C5 in SG 24, the tube that

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failed on February 15, 2000. During this review, the team used the actual data collected in 1997 and assessed the detectability of these flaws and their potential size based on techniques used in 1997.

The team also reviewed the information on this topic that Con Edison provided prior to the exit meeting on July 20, 2000.

b. Observations and Issues

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 TSP) 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 - Figure 5 is a c-scan plot of the vertical component of the ECT 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. This ridge makes both the detection and sizing of this defect more difficult. Figures 6 and 7 are the lissajous plots for the flaw area and the noise ridge, respectively. These figures show that a flaw signal is distinguishable from the characteristic noise signal, as was the case in the other tubes profiled. No year 2000 data is available since the tube failed.

The indication has been profiled for both the 300 and 400 kHz inspection frequencies, as is shown in Figure 4. The signal-to-noise is slightly better for the 400 kHz frequency than for the 300 kHz. The voltage above 1.2-volts indicates the defect; however, there is considerable noise. The depth estimate based on 1997 data is 87% 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

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crack. The depth estimate based on 1997 data is 63.7% TW.

Con Edison correctly stated that there was no quantitative noise criteria present in EPRI Steam Generator Examination Guidelines, Rev. 4, used in 1997. However, the 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 ECT testing and 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"*

The team determined that it was possible to compare the amplitude of the noise in the tubes being inspected to the size of a defect that it could be masking. The ratio of the noise voltage to the defect voltage should be determined for the appropriate defects. In the documentation provided to the team on July 20, 2000, Con Edison compared the 1997 noise voltage in tube R2C5 to the voltage from the standard EDM notches and stated that the flaw depths would have been about 50%TW.

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. EPRI determined the qualification of the mid-range Plus Point probe using a generic population of SG flaws with a sample set chosen to represent the spectrum of tube conditions consistent with a generic population. If the proportion of noisy tubes to non-noisy tubes is greater in a SG than in the qualification sample the POD could be affected.

The 1997 data contained significant noise, possibly due to OD deposits on the U-bends tubes, that making detection more complicated. However, the defects were detectable using the c-scan and lissajous signals available at the time. Con Edison did not identify the possible effect that the noise could have on flaw POD as a significant condition adverse to quality and did not enter the issue into their corrective action program. Techniques to minimize the effects of the noise on data quality were not used and/or criteria for rejecting data based on high noise was not provided.

c. Findings

The additional information provided on July 20, 2000, did not change the team's findings. The team's findings in this area are relative to the Revised Reactor Oversight Program (RROP) Crosscutting Issue of Corrective Action and are discussed in Section 4OA1.1.

.2 Review of the 1997 U-Bend Primary Water Stress Corrosion Cracking Indication

a. Inspection Scope

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

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was also generated.

The team reviewed the 1995 Indian Point 2 Steam Generator Life Predictions with respect to U-bend PWSCC. These predictions were completed by another contractor in 1995 following the 1995 outage and in 1997 following completion of the 1997 outage.

The team also reviewed the information on this topic that Con Edison provided prior to the exit meeting on July 20, 2000.

b. Observations and Issues

The c-scan plot of the 1997 data from R2C67 is shown in Figure 18. 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.

While the flaw was identified and the tube plugged, neither Con Edison nor its ECT contractor recognized the discovery of the low row U-bend apex indication as a significant condition adverse to quality and did not enter the issue into their corrective action program. Identification of this flaw was significant, because it was the first observation of this type of degradation in the U-bend area in SG tubes at Indian Point 2. There was no specific review as to the significance of this flaw nor the possible extent of the condition.

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°F, 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 one PWSCC U-bend crack at the end of the last cycle of operation (EOC 21)

The contractor's 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 the

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contractor 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.

c. Findings

The additional information provided on July 20, 2000, did not change the team's findings. The team's findings in this area are relative to the RROP Crosscutting Issue of Corrective Action and are discussed in Section 4OA1.1.

.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 SGTF and Con Edison subsequent responses, the Indian Point 2 Steam Generator Data Book, dated December 1, 1997, and the 1995 Indian Point 2 Steam Generator Life Prediction, to assess SG conditions in 1997 relative to tube denting and hour-glassing. (See Applicable Steam Generator Degradation Mechanisms above under BACKGROUND)

b. Observations and Issues

Early in the inspection, the team questioned whether TSP hour-glassing could have contributed to the development of PWSCC leading to the failure of tube R2C5 in SG 24. 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 criterion for when hour-glassing was significant. As such, Con Edison never reported any significant hour-glassing.

During the outage, 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 deflection of the row 1 tubes, finding that 0.46 inches 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 apart, and have larger radius U-bends. The critical movement for row 2 tubes was 0.1 of an inch. This calculation showed that the stress in R2C5 was above the threshold for PWSCC..

The 1997 SG inspection identified 37 tubes that needed to be plugged due to denting at TSPs. Of significance, 19 tubes 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 (15 in row 2, three in row 3, and one in row 4) at the upper TSP caused the 19 U-bend restrictions, not allowing examination of the upper TSP area. These tubes were plugged.

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Neither Con Edison nor its ECT contractor identified this first identification of 19 low row tube restrictions due to denting at the upper TSP and the potential for flow slot hour-glassing as a significant condition adverse to quality that could impact the integrity of tubes beyond row 1. This issue was not entered into the corrective action program.

Further, the total of 37 dented tubes was above the 1995 SG life prediction best estimate of 25 such tubes during the 1997 outage and was a significant increase above the numbers of restrictions identified in the last several outages (one during RFO-15, zero during RFO-14, and one during RFO-13) .

c. Findings

The team's findings in this area are relative to the RROP Crosscutting Issue of Corrective Action and are discussed in Section 4OA1.1.

**1R3 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 ECT 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) and specified the use of EPRI Steam Generator Examination Guidelines, Rev. 4, by the ECT contractor.
- EPRI Steam Generator Examination Guidelines, Rev. 4, (EPRI Guidelines) Appendix H.
- "Eddy Current Low Row U-bend Examination, MIZ-18A and TC6700, Non-Mag. Bias and Mag. Bias Equivalency Qualification." The purpose of this equivalency qualification was to demonstrate that the magnetic bias Plus Point probe (which was used for examination of the Indian Point 2 low radius U-bends) had comparable detection capability to the non-magnetic bias Plus Point probe.
- ETSS #96511, 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.
- ANTS # IP2-97-E, Rev. 0 - documentation of the analysis method of SG low radius U-bends at Indian Point 2 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 at Indian Point 2.

The team also reviewed the information on this topic that Con Edison provided prior to

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the exit meeting on July 20, 2000.

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 ECT 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 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 # 96511 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 # 96511 were not met.

- The calibration standard ACGT-006-97 manufactured in accordance with Westinghouse Drawing 1B79882 did not include the required 40% TW inside diameter axial and circumferential EDM notches.
- The required phase rotation set-up was not used. This ANTS sheet instructed the analyst to adjust phase rotation so that probe motion was horizontal. This was not in accordance ETSS # 96511. The team considered this technically deficient, due to the insensitivity of the Plus Point probe to probe motion resulting in too small of 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 that were unclear and not in accordance

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with the EPRI qualified standard. 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 EDM notch was utilized in the qualification process. The team estimated that this resulted 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 # IP2-97-E, Rev. 0, was not prepared, however, to comply with the phase rotation requirements of the equivalent qualification.

With respect to the ETSS # 96511 requires set-up, the team 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 have a small negative impact the ability to detect small PWSCC flaws.

c. Findings

The additional information provided on July 20, 2000, did not change the team's findings.

The 1997 analysis of SG low radius U-bends at Indian Point 2 was performed in accordance with the requirements of ANTS # IP2-97-E, Rev. 0, was not consistent with the requirements of either ETSS # 96511 or the Westinghouse equivalency qualification.

While this issue had a small effect on the POD of low row U-bend indications, there was no effect on the RCS integrity. In accordance with the Reactor Safety Significance Determination Process (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 ECT techniques for U-bend inspection since the Plus Point ECT 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, Criterion IX, Special Processes for ECT 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 05000247/2000-010-01, EA 00-179-1)

.2 Data Analysis Guideline Review

a. Inspection Scope

The team reviewed the data analyst guidelines requirements for use of the Plus Point probe contained in Westinghouse Procedure DAT-IP2-001, "Data Analysis Technique Procedure," Rev. 0, and compared them with the EPRI SG Guidelines. Eddy Current Probe Authorization List, Revision 1, dated May 14, 1997, which provided the specific

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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 The Eddy Current Probe Authorization List, Revision 1, dated May 14, 1997, indicated that these probes were qualified to Appendix H of the EPRI Guidelines and were authorized for use in characterization of indications in dented intersections and restricted tubes.

c. Findings

The team identified no findings during this review.

.4 Analysis Training Review

a. Inspection Scope

The team reviewed the training provided to the data analysts in accordance with the criteria contained in the EPRI Guidelines, Section 6.2 (Site-Specific Performance Demonstration) which 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 [inservice inspection vendor or ECT contractor], 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 the 1997 refueling outage ECT 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) set-up 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 ID flaws at free span locations. Due to the lack of identification at Indian Point 2 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

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The inspectors considered the incomplete status of the ECT analyst training and testing information to be an indicator that the site-specific performance demonstration requirements of the EPRI Guidelines, 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 ECT analysts was not established. The team characterized this as a minor violation not subject to enforcement action.

c. Findings

The team identified no findings during this review.

1R4 Risk Significance - Event and Core Damage Frequency 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 4OA1.1. 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. 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.

.2 Potential Consequences:

The following is a synopsis of the complete risk assessment developed by the NRC staff, and included as Attachment 2 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:

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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 determine a yearly incremental CDF/LERF for an SGTR of approximately $1\text{E-}04$ /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 $5\text{E-}05$ /RY and $1\text{E-}04$ /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 in Attachment 2.

Therefore, the LERF increment associated for a SGTR event is considered to be above the $1\text{E-}05$ /RY criterion for a "red" significance determination.

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4. OTHER ACTIVITIES (OA)

4OA1 Steam Generator Program CROSSCUTTING ISSUE - Corrective Action

.1 1997 Steam Generator Inspection Program

a. Inspection Scope

The team reviewed the 1997 SG inspection program and identified performance issues as documented in Sections 1R2.1, 1R2.2, and 1R2.3. The team assessed these issues relative to the standards established by 10 CFR 50, Appendix B.

The team also reviewed the information on this topic that Con Edison provided prior to the exit meeting on July 20, 2000.

b. Issues and Findings

The additional information provided on July 20, 2000, did not change the team's 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 tubes that contained detectable PWSCC indication in the low radius U-bend area. The significant noise present in the ECT data for the low radius U-bends, hampered the capability to detect flaws in this region. However, the team found that four such defects were detectable using the 1997 data and available techniques. Further, the 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 the potential for 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 ECT 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., 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. (See Section 1R2.2)
2. recognize the significance of, and fully evaluate, the flaw masking effects of the

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high noise encountered in the ECT 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. (See Section 1R2.1)

3. appropriately establish procedures and implement practices to address the potential for hour-glassing in the upper TSP 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 TSP hour-glassing were not assessed following the identification in 1997 of ECT probe restrictions at the upper TSP and the identification of a PWSCC indication at the apex of a SG tube. (See Section 1R2.3)

Using the Reactor Safety SDP, as documented in Section 1R4, 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. In accordance with NRC Enforcement Policy and Reactor Safety SDP this matter is considered an apparent violation. (AV 50000247/2000-010-02; EA 000-179-1)

.2 Review of Con Edison's Root Cause Analysis for the Tube Failure

a. Inspection Scope

The team reviewed Con Edison's root cause analysis for the SGTF, dated April 14, 2000, and the corrective action system condition reports generated on SG issues.

b. Issues and Findings

The team observed that Con Edison's root cause analysis did not identify and address the SG program performance issues identified above in Section 1R2 and 1R3 as they related to the SGTF on February 15, 2000. While the root cause analysis attributed the failure to a flaw that was obscured by ECT signal noise, it did not identify, nor address, deficiencies in the processes and practices during the 1997 SG inspection.

On March 20, 2000, Con Edison initiated CRS 200001939 which documented that four tubes had 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; R2C5 - 87% W, R2C69 - 53%TW, R2C72 - 75%TW and n SG 21 R2C87 - 53%TW. This review compares well with the teams independent depth estimates. (See Section 1R2.1) However, 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 which comments on the detectability and plugging on detection of PWSCC flaws as justification for not complying with the Indian Point 2 TSs. The team characterized this as a minor violation

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not subject to enforcement action.

4OA2 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. J. Groth. 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 ECT 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 contractor proprietary information. This information was not included in this report and the proprietary information will be returned to Con Edison.

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PARTIAL LIST OF PERSONS CONTACTED

Con Edison:

J Groth, Chief Nuclear Officer
A. Blind, Vice President
J. Baumstark, Vice President, Nuclear Power Engineering
J. McCann, Manager, Nuclear Safety and Licensing
A Spaziani, Nuclear Safety and Licensing Engineer
J. Mark, SG Program
J. Parry, SG program
G. Turley, Independent, Quality Data Analyst

Westinghouse:

D. Adomonis
R. Maurer
S. Ira
J. Maris

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ITEMS OPENED AND CLOSED

Opened

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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)

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- ▶ July 29, 1997 - SG Tube Inservice Examination 1997 Refueling Outage - TS 4.13.C.2 Submittal (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 - ECT 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

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- ▶ 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 ECT 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 ECT 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

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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

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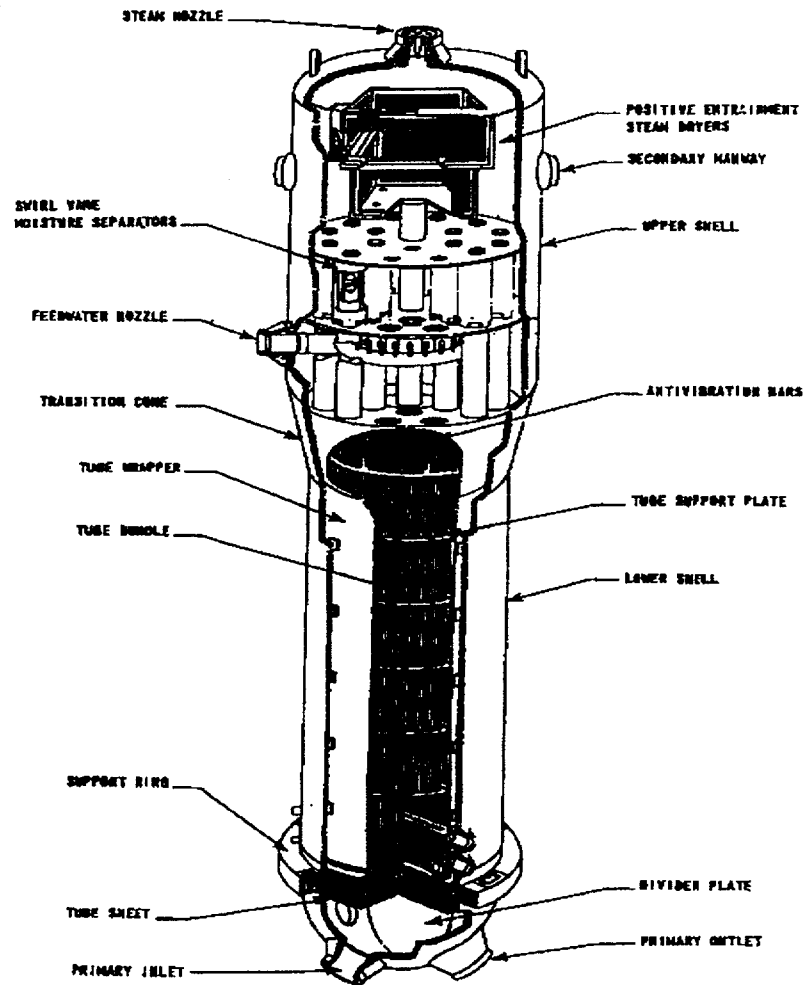
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LIST OF ACRONYMS USED

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
PWSCC	Primary Water Stress Corrosion Cracking
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
TSP	Tube Support Plates
TS	Technical Specification

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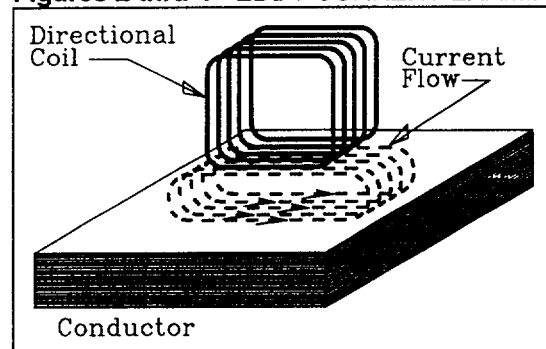
REFERENCED FIGURES**Figure 1 - Westinghouse Model 44 Steam Generator**

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Figures 2 thru 4 - EDDY CURRENT EXAMINATION**Figure 2** Directional pancake probe**Figure 3** Plus Point probe

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Figure 4 Difference signal from axial
and circumferential coils are amplified

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Figure 5 R2C5C-scan with 1997 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.

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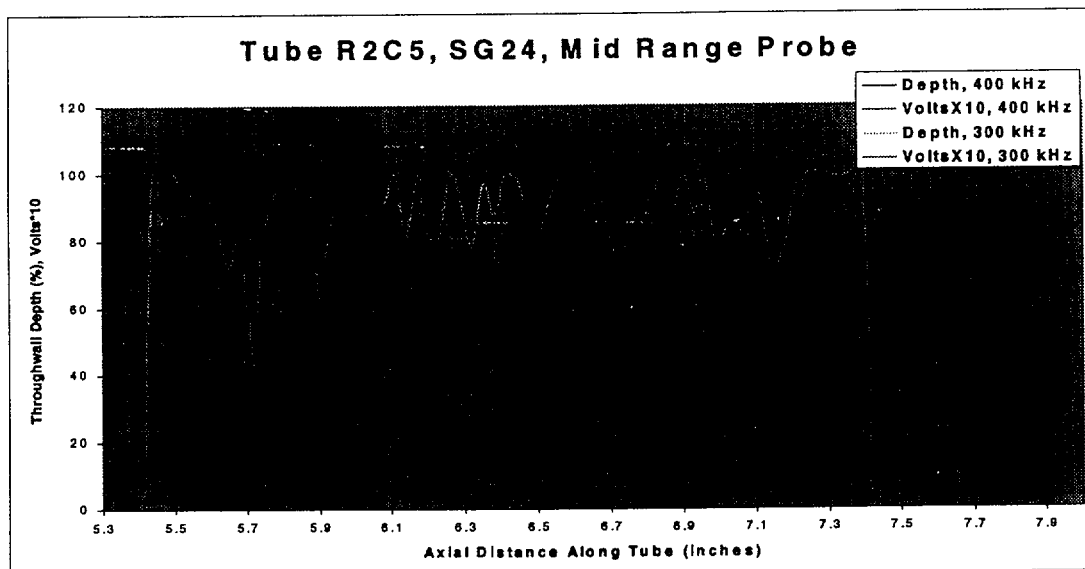


Figure 8 - Contour of the crack in tube R2C5 in SG24 using the 1997 data from the mid-range plus-point probe.

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

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Figure 9 - 1997Mid- range scan .

Figure 10 - 2000 mid-range scan

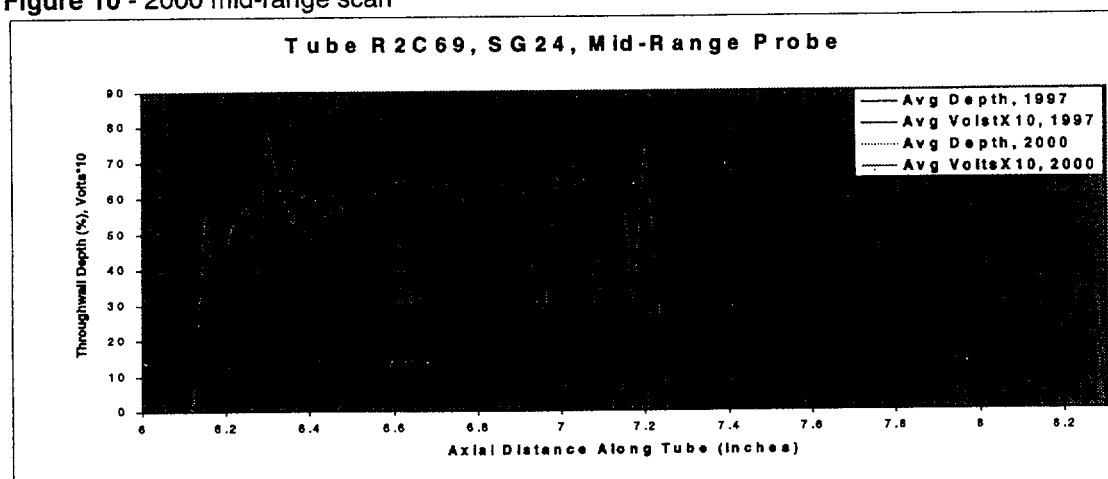
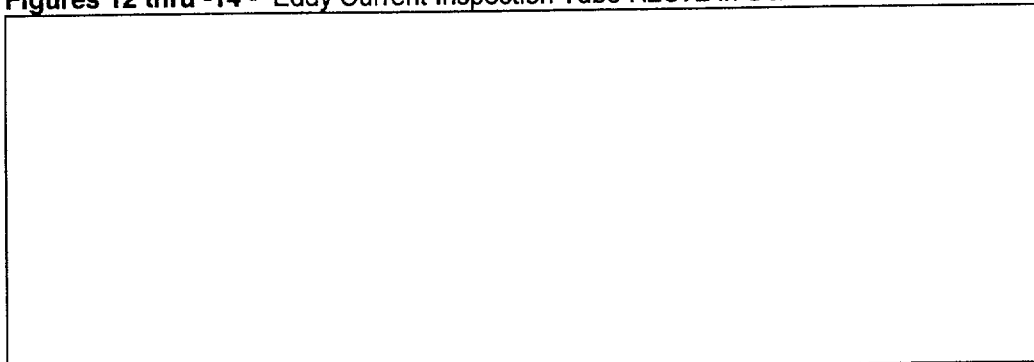
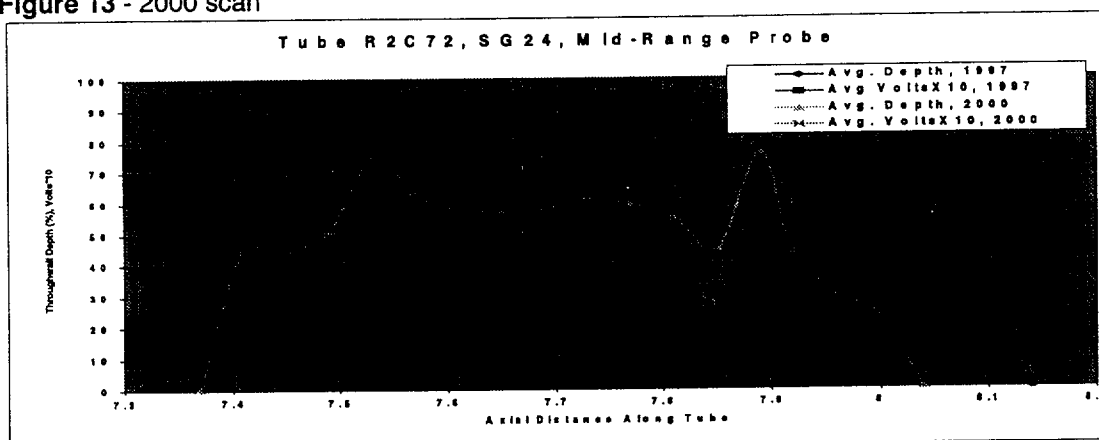
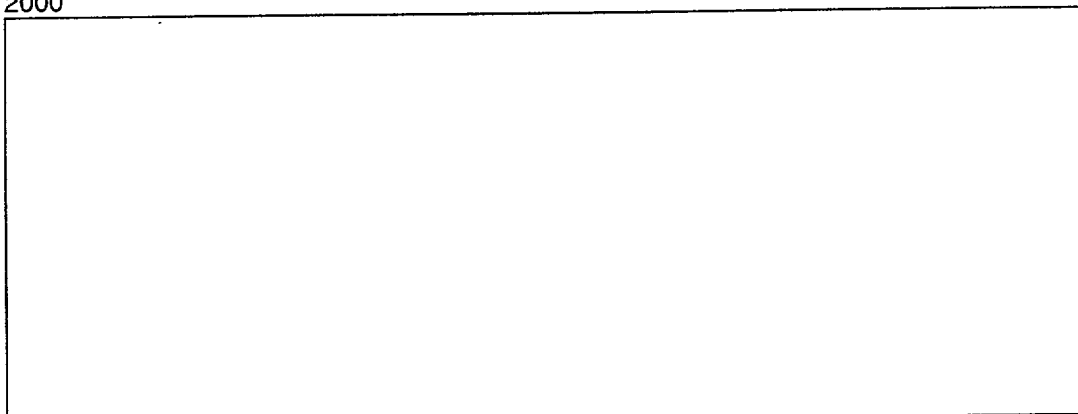


Figure 11 - Profile of Growth between 1997 and 2000

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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**

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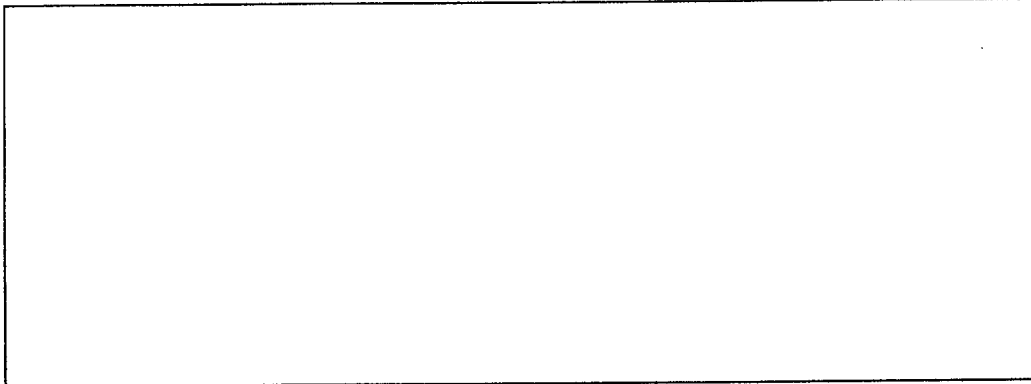


Figure 16 - 2000 scan

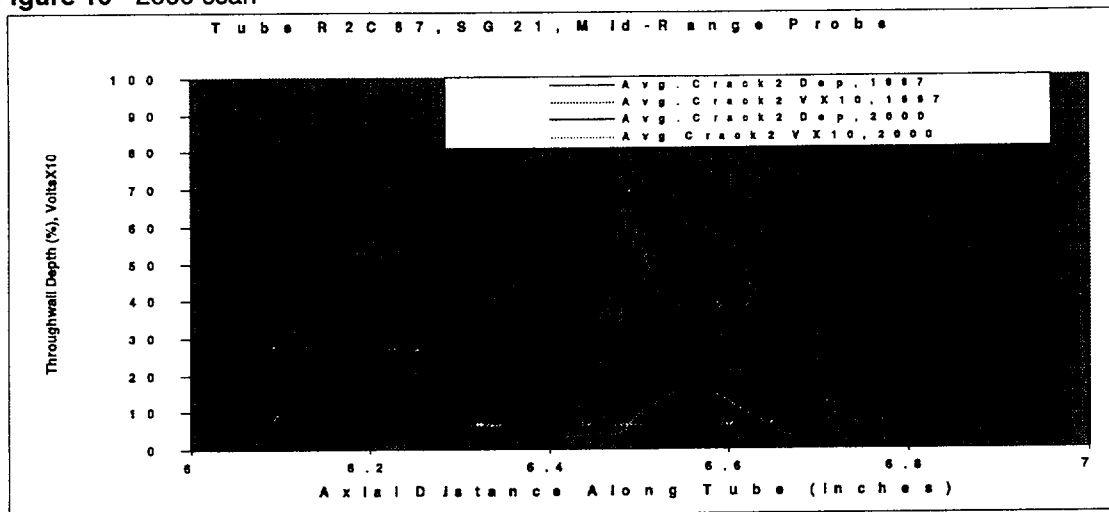


Figure 17- Profile of crack growth of tube R2 C87. Figures 15 thru 17 - Eddy Current Inspection- Tube R2C87 in SG 21

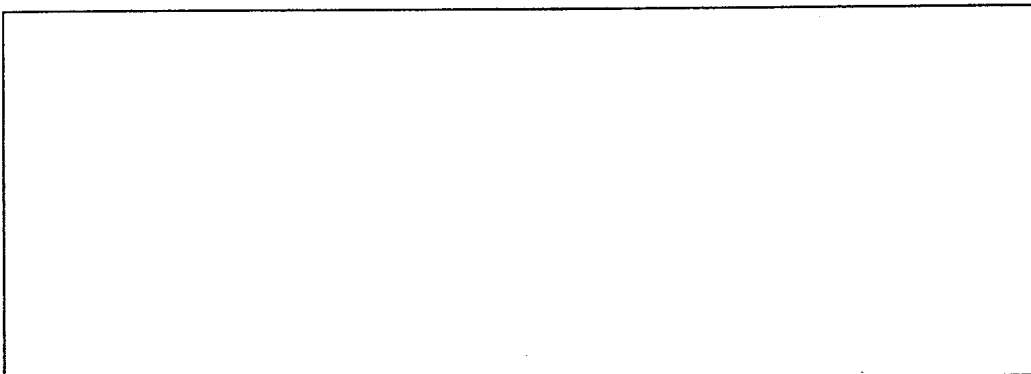


Figure 15 - 1997 scan

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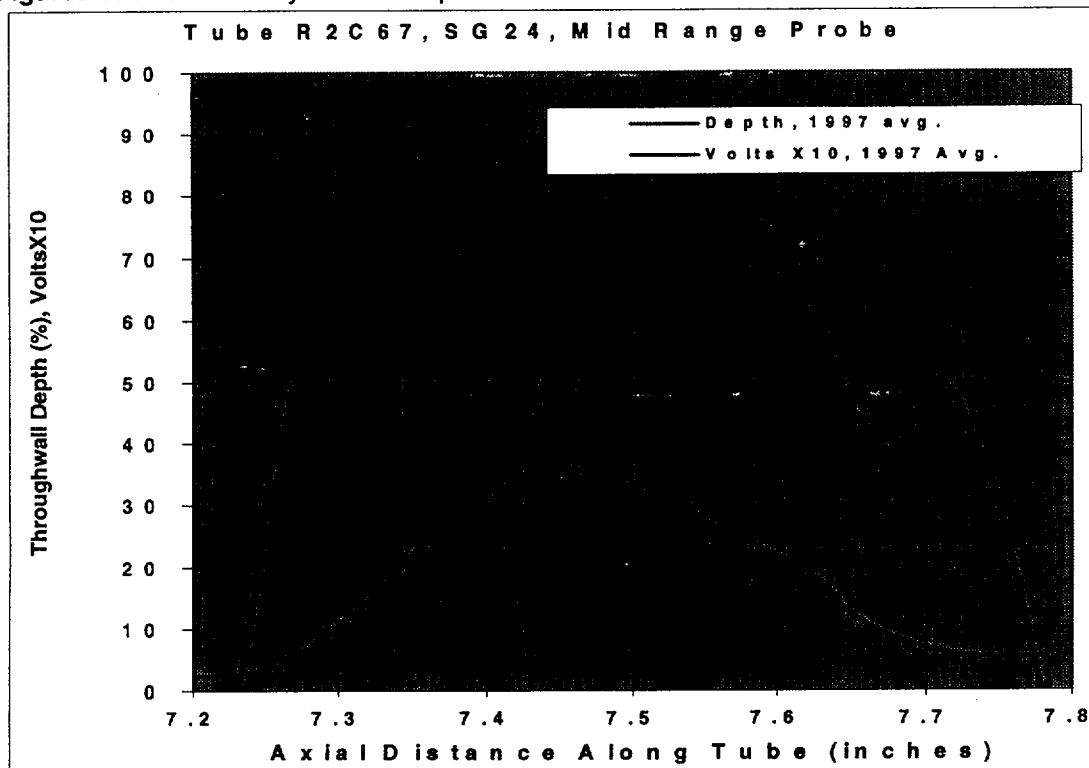
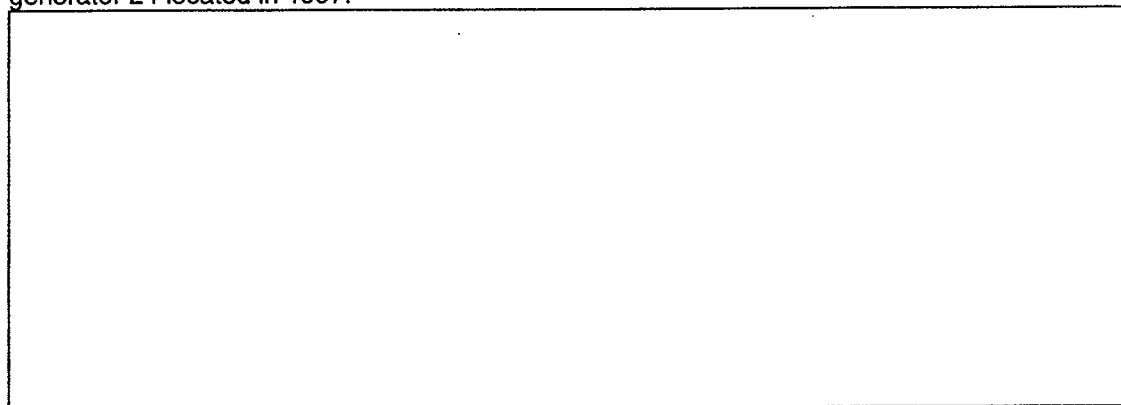
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Figures 18 and 19 - Eddy Current Inspection- Tube R2C67 in SG 24**Figure 19 - Depth and voltage profile of the crack in tube 2-67 of steam generator 24 located in 1997.****Figure 18 - Crack in tube 2-67 of steam generator 24, found in 1997.**

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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
<ul style="list-style-type: none"> • Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness 	<ul style="list-style-type: none"> • Occupational • Public 	<ul style="list-style-type: none"> • 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 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

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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>.