Holian

Slide 2
This is the regulatory conference on the inspection findings from the Indian Point 2 Special Inspection concerning the performance of Con Edison during the1997 steam generator inspections. This meeting is between the NRC and Con Edison and is open for public observation. The meeting will be transcribed. Copies of the NRC inspection report and the presentation slides can be found on the table in the back of the room.

The agenda today is as follows: (1) After this, Dan Holody of the NRC will present background information on the regulatory conference process and the relationship to the new reactor oversight program. (2) Dave Lew will then provide a summary of the steam generator tube leak event, the NRC event response, and the NRC inspection findings. (3) Jim Trapp will present the NRC assessment of risk. (4) Con Edison will then make their presentation, followed by the NRC wrap up of the meeting.

During the NRC inspection conducted earlier this year, we found that the 1997 steam generator inservice examinations were deficient in several respects. Despite opportunities, Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected the steam generator inspection program. Con Edison did not adequately account for conditions that adversely affected detectability of, and increased the susceptibility to, tube flaws.

This conference provides an opportunity for Con Edison to present information that may affect the NRC conclusions with regard to the inspection findings and risk assessment. The NRC is aware of Con Edison's disagreements with the inspection findings. Many of these viewpoints had been discussed during the inspection. The NRC considered these issues during the finalization fo the IR. The NRC decided that 10CFR50, Appendix B, Criterion XVI, Corrective Actions, is being applied appropriately. No decisions will be made by the NRC during this meeting. However, the NRC will consider this information and transmit by letter the final NRC conclusions regarding the inspection findings and risk assessment.

C\62

Holody

Slide 3

Good afternoon, my name is Dan Holody. I am the Team Leader of the Region I Enforcement/Allegation Staff. Today, the NRC is conducting a Regulatory Conference with Con Edison to discuss the risk significance of performance deficiencies, as well as an apparent violation associated with Con Edison's conduct of its inspection of steam generators at Indian Point 2 in 1997.

Today's Regulatory Conference is part of the NRC's new reactor oversight process for dealing with performance issues at nuclear facilities. Using the Significance Determination Process, the issues are assessed based on safety/risk significance and are assigned a color of green, white, yellow, or red, with green being the least significant and red being the most significant. After a potentially safety/risk significant violation is identified and characterized by the Significance Determination Process as white or greater, an opportunity for a Regulatory Conference is offered to a licensee in order to discuss significance determination, performance, potential violations, root causes, and corrective actions.

In an inspection report issued on August 31, 2000, the NRC preliminarily characterized the significance of the 1997 steam generator examination deficiencies at Indian Point 2 as highly risk significant, or red. Con Edison requested that a Regulatory Conference be held to discuss their position on the significance of this issue, including the bases for their position, and any disagreement with the apparent violation.

A Regulatory Conference is essentially the last step of the process before the staff makes its final decision on the significance of the inspection findings. The purpose of this conference today is not to negotiate the significance of the issue or any resulting enforcement action. Our purpose here today is to obtain information from Con Edison that will assist us in determining the appropriate significance determination, such as a common understanding of the facts, and a common understanding of the assumptions and factors used to determine the significance of the issue. Con Edison may also provide any other information that may be relevant to the application of significance determination in this case, including its position on the content and accuracy of the inspection report findings.

The apparent violation, Con Edison's failure to identify and adjust or modify the inspection methods and analysis to account for significant conditions that affected the quality of the 1997 steam generator inspection, is subject to further review and may be subject to change prior to any resulting enforcement action. It is important to note that the decision to conduct this conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. Rather, the NRC will evaluate the information presented today, along with our previous inspection findings, to determine a final significance determination as well as appropriate enforcement action if warranted. Normally we will issue our decision in this matter within 30 days.

Prior to turning this meeting over to Mr. Lew, I note that any statements or opinions made by NRC staff at this conference should not be viewed as a final NRC position, nor should the lack of an NRC response to a statement be viewed as NRC acceptance of that position.

Thank you.

Lew

Slide 4

On February 15, 2000, a steam generator tube failure occurred at the Indian Point Unit 2 reactor facility. This resulted in an initial primary-to-secondary leak of approximately 146 gpm. Con Edison declared an "Alert", which is the second lowest level of emergency action in the NRC required emergency response plan, and initiated a manual reactor trip before identifying and isolating the source of the leak. Con Edison successfully mitigated the event and placed the plant in cold shutdown. While there was a minor radiological release to the environment, this release was well within regulatory limits, was not detected off-site, and the event did not impact public health and safety.

Subsequent to the event, Con Edison conducted inspections of the steam generators. They found that the failure was located in the apex of tube R2C5, which is a low row tube. As they continued eddy current testing of the steam generator, the number of defects identified placed two of the steam generators in technical specification category of C-3, which required NRC approval for restart with the existing steam generators at the time.

Slide 5

The NRC immediately responded to the event through on-site follow up and monitoring by the resident inspectors and inspectors dispatched from Region I, and by NRC managers and the technical staff from Region I. An Augmented Inspection Team (AIT) was conducted from February 18 through March 3 and an AIT Follow-up was conducted from May 15 through May 26 to review the safety implications and associated licensee actions in response to the steam generator failure. The cause of the tube failure was not reviewed during the AIT and the AIT FU. Instead, the NRC conducted a special inspection from March 7 through July 20, to review the causes of the failure and the adequacy of Con Edison's performance during the 1997 SG inservice inspections. This inspection consisted of personnel from Region I and the Office of Nuclear Reactor Regulation, as well as NRC-contracted specialists in steam generator eddy current test. The findings from this inspection are the topic of this regulatory conference.

Slide 6 The overall direction and execution of the 1997 SG inservice examinations were deficient in several respects. Despite opportunities, Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected the steam generator inspection program. Con Edison did not adequately account for conditions, which adversely affected the detectability of, and increase the susceptibility of, tube flaws.	

Slide 7

More specifically, during the 1997 inspections ...

A primary water stress corrosion cracking (PWSCC) defect was found for the first time at this facility at the apex of a row 2 U-tube. The significance was not understood by Con Edison. The appearance of one defect signifies the potential for similar cracks in other low-row tubes. Such apex flaws have been associated with through wall leakage and bursting. Tube ruptures have occurred due to PWSCC. Con Edison did not review for the possibility of hour-glassing or question the adequacy of the inspection method. This issue was not entered into the corrective action system. Con Edison did not perform an adequate evaluation of the cause and susceptibility of low-row tubes to PWSCC, the extent to which this degradation existed, and the increased probability of such a defect to rupture during operation. The tube was simply plugged.

Slide 8

Tube denting in low-row tubes was identified for the first time at this facility at the upper tube support plates (TSPs). Restrictions were encountered as ECT probes were inserted into 19 of the low-row tubes. This signifies the probability of hour-glassing, or deformed flow slots at the upper TSPs. This hour-glassing increases the stresses at the U-bend apex of the tubes, which, in turn, are the leading contributor to low-row U-bend apex PWSCC. This issue was not reviewed by the corrective action system. Con Edison did not perform an adequate evaluation for the potential of hour-glassing nor did they have established procedures or examination criteria to determine if such hour-glassing was occurring.

Slide 9

Significant ECT signal "noise" interfered with the data analysis of low-row tube areas. This significant noise level reduced the probability of identifying an existing PWSCC tube defect. However, the 1997 SG inspection program was not adjusted to compensate for the negative effects of this noise in detecting flaws, particularly when conditions such as the first U-bend PWSCC defect and the upper TSP low-row tube denting indicated the increased susceptibility to PWSCC. It was the NRC determination that detailed, careful review of 1997 data could have identified four existing PWSCC defects.

These performance issues contributed to tubes with primary water stress corrosion flaws in the small radius U-bend being left in service, until the failure of one of these tubes occurred on February 15.

The issue was not reviewed by the corrective action system.

Trapp

Slide 10

Good morning, my name is Jim Trapp and I'm one of the Senior Reactor Analysts in Region I. I am going to briefly discuss the risk significance evaluation performed to determine the risk associated with these inspection findings. The risk assessment was performed in accordance with the revised oversight program inspection manual chapter 0609. The IMC provides three phases or levels of risk assessments that increase in sophistication. The phase I screen is performed to determine if additional analysis of the finding is necessary, phase II utilizes pre-established sequences from the IPE to quantify risk. Phase III evaluations are performed using available risk information to more accurately characterize the risk of findings. All three phases of the SDP were performed for these findings.

The SDP determines the potential risk associated with existing conditions. It is not limited to evaluating only the actual consequences. For example, if all the EDGs are found inoperable for a significant duration, yet offsite power is not lost during the period that the EDGs are inoperable, the actual consequences are negligible. However, the change is core damage frequency delta-CDF and overall risk of this condition would be significant. In the case of the IP2 SG findings, poor quality SG tube inspections in 1997 would increase the likelihood of a SGTR which is a significant event and therefore, these findings would be risk significant. SGTRs events are significant, because by there nature, this type of accident degrades both the RCS and containment fission product boundaries. Therefore, will increase both the probability of core damage and release of radiation to the public.

The Phase I /II SDP evaluation determined that these findings were potentially highly risk significant (Red). Therefore, a Phase III evaluation was performed by the PRA branch of NRR. The key assumptions in the phase III analysis are 1.) that the initiating event frequency for a SGTF is 1/year (assumption is based on the as-left condition of the SG tubes in 1997 and the actual SGT failure history); 2.) ½ SG tube failures will result in SGTRs (assumption is based on Surry and Doel (Belgium) SGTF events); 3.) delta-CDF is ~ delta-LERF (assumption is based on the observation made by the NRC in NUREG-1560 that most SGTR core damage events result from a stuck open secondary steam relief valve which allows a direct fission product flow path from the core to the environment).

In addition to spontaneous SGTFs, the phase III evaluation also included a review of other initiators which could induce a SGTF. These are events that increase the pressure differential across a cracked SGT which could induce the tube to rupture. The accident

initiators considered were secondary side system faults, ATWS, and severe accidents.	

Slide 11

IMC 0609 establishes 4 risk thresholds for risk significance for both delta-CDF and delta-LERF. The findings are assigned a color based on risk significance with Green being the least risk significant and Red being the most risk significant. The risk threshold for a red finding is delta-CDF of > 1E-4 or a delta-LERF > 1E-5. Each decade reduction in Delta-CDF or LERF will result in a color reduction.

The results of the NRC's phase 3 risk assessment are documented in Attachment 2 of IR 2000-007. The delta-CDF and delta-LERF were determined to be ~ 1E-4. This would be indicative of a high risk significant or RED finding. This concludes my comments on the NRCs risk determination for these findings. Thank You!