3 & 4

From: Robert Summers
To: Trisha Haverkamp
Date: Mon, Apr 10, 2000 1:44 PM
Subject: Fwd: IP 2 Qs & As Rev 6

This was the last Q&As document used to brief NRC senior managers prior to the public meeting at the site (mid-March). Much has changed in our knowledge on the answers, but we haven't updated the q&as (and don't intend to unless we have some big activity where we'll have to use this to brief sr. managers).

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From: Robert Summers
To: IP2 Comm Team
Date: Fri, Mar 10, 2000 6:51 PM
Subject: IP 2 Qs & As Rev 6

rev 6 is final

it has a lot of answers in the rad release area that are very technical

we will use this as briefing material in prep for the Tuesday meeting with the public.

CC: IP2 Comm Team CC List

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IP 2 Questions & Answers Paper

Sections

This document contains background materials for NRC staff in preparation for meetings/discussions with external stakeholders regarding the tube failure event at Indian Point Unit 2 on February 15, 2000. The contents are not expected to be used as verbatim answers to questions; however, familiarization with these materials should allow a consistent response regarding the Agency position on a wide variety of related topics.

This document discusses issues currently under review by the AIT, and therefore, should be treated as containing pre-decisional information.

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Yes, but "small"?

Key points:

- The plant ~~was safely shut~~ down and all safety systems operated as designed.
- The leakage in the steam generators prior to the event was well within industry ~~guidance and~~ a small fraction of ~~technical~~ specification allowable levels.
- NO radioactivity was measured or detected outside of the plant either on-site or off-site; although small releases occurred.
- Required, immediate notifications were made appropriately; although, some problems occurred in implementing emergency response activities.

General Guidance to be used when appropriate:

- NRC has safety limits intended to protect the public health and safety
- NRC has limits on radiation releases from the plant
- NRC has limits on operation and maintenance of steam generators
- Some RCS leakage in the steam generator may occur - limits on this operational Leakage are set at a level that provide ample margin to any safety or health limits
- IP-2 was operating well within specified limits
- IP-2 staff were monitoring conditions appropriately, following NRC and industry guidelines
- A steam generator tube failure is an analyzed condition . . . there are pre-established procedures for handling a tube leak/failure
- Operators acted promptly to shut down the reactor, classify the event, and make immediately-required, off-site notifications
- Operators were able to isolate the affected steam generator and bring the plant to a stable, 'cold shutdown' condition
- ConEd dispatched personnel to sample and monitor conditions around the plant . . . no radioactivity above normal background conditions were identified

- NRC is confirming all of the above information through the AIT
- Resident and other inspectors have been onsite since the event
- NRC promptly staffed incident response centers in both Region and Headquarters

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A. Questions related to IP-2's performance:

1 How is IP-2's performance?

ANSWER: They have had to face a number of issues at this plant and they have brought in a number of new managers to address these issues.

2 How did they perform in this event in comparison to the past?

ANSWER: Overall, they were successful at responding to this event in shutting the plant down, isolating the leak, and bringing the plant to a safe, cold shut down condition. Also, the operators performed well in identifying the problem and immediately shutting down the plant; in classifying the emergency condition; and making the required immediate notifications.

3 How is Con Ed's performance overall?

ANSWER: ConEd has come under fire for a number of problems last year resulting in blackouts affecting 250,000. A recent state report blamed ConEd's faulty planning and equipment for the blackouts affecting customers from Yonkers to the Bronx to upper Manhattan.

The state report calls on ConEd to take steps to improve reliability, including redesigning the company's distribution system, developing ways to identify equipment vulnerable to heat stress failures, **improving crisis communications with the public**, and increasing the amount it pays customers for losses.

After the tube failure event, a number of state legislature representatives have called for ConEd to replace the steam generators at IP 2 prior to resuming operations.

4 Didn't ConEd put profit before safety by not replacing the steam generator, equipment that they had already purchased several years ago?

ANSWER: **[Give operational limits answer.]** The NRC establishes operational limits for key parameters at each nuclear plant. ConEd was well within those

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parameters in the days prior to the event.

5 Why is ConEd operating this old plant with old equipment?

ANSWER: **[Give Operational limits answer.]** The NRC establishes operational limits for key parameters at each nuclear plant. ConEd was well within those parameters in the days prior to the event.

6 Was the plant safe?

ANSWER: All indications are that our defense in depth approach proved effective. The leak was small; the operators shut down the plant. We are trying to confirm all the details.

7 Is it safe now?

ANSWER: Yes, the plant is now operating within its shutdown operational limits. We will continue to monitor ConEd's actions to maintain the plant shutdown and investigate the steam generator condition.

8 Was my family ever in any danger?

ANSWER: No.

9 Did ConEd fail (for any reason) during the event?

ANSWER: That is why we are conducting an AIT, to confirm that ConEd followed their procedures. [Add to Answer with, "They did ..." only if you know the answer won't change after the inspection.]

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B. Questions related to plant restart, NRC oversight and AIT activities:

1 Will the NRC use the 0350 process to approve restart?

ANSWER: No.

2 How long before we know what Con Ed's plan of action will be?

ANSWER: This question is best answered by Con Edison. However, we do have an Augmented Inspection Team that has begun work to follow Con Ed's actions and to conduct an independent investigation of the event. However, I can say that the initial step was completed, and that was to place the plant in a safe, cold shutdown condition. Also, ConEd has identified a tube in the number 24 steam generator that is the most likely cause of this event. They are about to complete the rest of their steam generator tube inspections and we need to assess their results. Further, the AIT has completed its onsite investigation, but they are still assessing their findings.

3 How long until the NRC knows what its course of action will be?

ANSWER: Our course of action is somewhat dependent on the licensee's. First we have to complete our independent investigation of the event. This should take no more than a couple of weeks. We also need to complete our assessment of the licensing bases of the Indian Point 2 steam generators, to make sure that any actions taken by Con Edison to recover from this event are acceptable.

4 Why is the NRC performing an Augmented Team Inspection? What exactly will it look at?

ANSWER: **DRS - L. Doerflein should answer this question refer to AIT Charter.**

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5 Why aren't you shutting down the plant permanently or until the steam generator(s) are replaced?

ANSWER: The NRC licensed the Indian Point 2 plant to operate safely within specified license conditions and technical specifications. We will continue to monitor Con Ed's actions to ensure that they live up to their end of that agreement. It would be premature though, at this time, for us to conclude that the plant can or cannot safely operate with these steam generators.

6 You are conducting an AIT now. Doesn't the need for that inspection confirm that ConEd did not handle this event right?

ANSWER We decide to send an Augmented Inspection Team based on a number of factors, including the need to understand the causes of the event. AITs are fact-finding inspections, and that's what we intend to do: find the facts about the event.

7 Will the NRC hold a public meeting prior to allowing the plant to restart to give the public an opportunity to voice concerns about its continuing to operate?

ANSWER: Vital information has been provided to the public through the media, meetings such as this, and through our press releases on the NRC's Web site. The plant is shutdown to examine the steam generators and more information will be provided when we have it and it is reliable. As long as they provide appropriate corrective actions, ConEd is authorized to operate IP2. Regarding a restart decision, the NRC has not placed any hold on ConEd. However, we are assessing ConEd's performance overall and have not yet made a final decision regarding the appropriate regulatory action that we may take.

8 Why is the NRC AIT not looking at the steam generator replacement issue?

ANSWER: The intent of the AIT is to find out what happened during the event. The review of the steam generators, and their acceptability, requires a different type of expertise and level of review that is being handled by the NRC Office of Reactor Regulation.

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9 What is the NRC's response to 'Cong. Kelly's request . . . for a public meeting?

ANSWER: This meeting was arranged in part, due to Congressman Kelly's request. In addition, the AIT will have an exit at the conclusion of that effort which will be open for public observation. In the meantime, we are working very hard to assess the facts and we fully intend to provide our assessment to the public when that is done. We will continue to provide as much information as we can to the public, such as through attendance at various local meetings, when we can. For example, we are planning to attend a meeting in Rockland County on March 16, similar to the meeting that was held in Westchester County last month.

10 What is the NRC's oversight of the reactor? [DRP-P.ESELGROTH]

ANSWER:

11 How will the new oversight program impact this ? What will change in the inspection program? How would our response to this event have been different under the new program? [WHO SHOULD HANDLE THIS?]

ANSWER:

12 The GAO report said that 60% of the NRC staff didn't understand or agree with the oversight change. The concern was that declining performance can't be detected. Please comment.

ANSWER: The GAO report did identify concerns among the staff that the changes to the revised reactor oversight process (RROP) may not identify degrading performance. As noted in the NRC's comments on the draft GAO report, the agency was not surprised by the results of the survey. The agency is only two years into a sustained change effort, and significant progress has been achieved. We believe the questions and concerns of the NRC staff at this point in the process are appropriate and constructive. The NRC staff is trained to bring a questioning attitude to licensee proposals, and thus it is not surprising that the staff approaches changes within the agency in the same fashion. As we undertake these new initiatives, including the RROP, we want the staff to participate in identifying, evaluating and resolving potential issues prior to full

implementation.

The agency is aware of the specific concern of detecting declining performance before significant reductions in safety margins occurred. This issue was addressed in SECY-00-0049, "Results of the Revised Reactor Oversight Process Pilot Program," dated February 24, 2000. In that paper, the staff highlighted this concern and recognized that a lack of experience in dealing with pilot plants using the RROP has contributed to the concern.

In developing the RROP, the staff used a graded approach to establish risk-informed thresholds, which are intended to provide reasonable assurance that safety margins are maintained, and allow sufficient time for both the NRC and licensees to address noted performance deficiencies before there is an undue risk to public health and safety. The pilot program increased staff confidence that the combination of performance indicators and inspection findings can provide appropriate indications of licensee performance. The appropriate implementation of these processes should provide reasonable assurance that safe plant operation is maintained. Initial implementation of the RROP, scheduled to begin this April, will enable the staff to confirm the capability of the RROP to identify declining safety performance trends in a timely manner, and to determine if threshold adjustments are warranted. During the pilot program, no significant aspects of licensee performance related to the cornerstones of safety were identified that were not adequately covered through the combination of performance indicators and inspection. Additional experience and insights gathered through the implementation of the RROP at all sites will permit more extensive execution of all aspects of the process and generate greater confidence in the appropriateness of the RROP.

Thus, while there may currently be considerable skepticism among the staff of the changes in the oversight program, including concerns with the ability to detect declining performance, this skepticism was not unexpected by the agency. The agency believes that additional experience with implementation of the RROP at all facilities will demonstrate that the RROP is capable of identifying degrading performance, and that staff skepticism of the changes will decline.

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C. Questions related to the sale of the IP-2 plant:

1 What review does NRC do of a proposed sale? What are NRC's criteria?

ANSWER: The NRC uses section 13.2 from the Standard Review Plan and the regulations (10 CFR50.80, 50.33(f), and 50.47) to determine the technical and financial qualifications of the buyer. The primary criteria are the ability of the buyer to provide the proper organizational and management needs to support the plant and the availability of the buyer's resources to safely operate and decommission the plant through the period of the new license.

2 What if the state or county objects to the sale?

ANSWER: Prior to the sale of the plant, the proposed license transfer is published in the Federal Register. The public, including individuals and organizations, can comment on the sale at that time. These comments are considered when evaluating the sale of the plant.

3 How does NRC evaluate the financial qualifications of the proposed new owners? How is the NRC qualified to evaluate these complex financial and legal matters?

ANSWER: The regulations (10 CFR 50.33(f)) provide the criteria to evaluate the financial qualifications of the buyers. We have financial analysts who have evaluated the financial aspects of over 60 license transfer applications over the past 5 to 6 years. These have included the sales of the TMI-1, Clinton, and Pilgrim plants (already approved) and the Nine Mile Point 1 & 2 , Vermont Yankee, and Oyster Creek plants (in process), where the financial qualifications issues are similar to those expected for IP-2. In addition to this experience, the staff has academic training as well as other work experience in the financial and economics areas.

4 How does the NRC make sure the funds will be available for decommissioning? What if decommissioning costs more than expected - who pays, or does the clean-up stop?

ANSWER: If the buyer is an electric utility, then the utility is allowed to continue collecting decommissioning funds while licensed. However, if the buyer is not an electric utility, then the buyer has to demonstrate, prior to the purchase, that the decommissioning funds are guaranteed according to our criteria. If the seller

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currently does not have enough funds to transfer to the buyer, then the buyer must compensate for the difference.

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D. Questions related to steam generators:

1 When is the NRC going to conduct its review of the Indian Point 2 steam generators? Which part of the agency will be doing that?

ANSWER: The NRC has already begun its review of the Indian Point 2 steam generators. The Region I Office has formed an Augmented Inspection Team, or AIT, that will conduct this review over the next few weeks.

In addition, the Office of Nuclear Reactor Regulation, or NRR, is reviewing the licensing bases for the Indian Point 2 steam generators. NRR is developing an information base providing information on the IP-2 steam generators. This information base will include information on observed tube degradation, inspection process and frequency, tube plugging history, and history of similar steam generators. NRR is also requesting an independent review by the Office of Nuclear Regulatory Research of the license amendment that allowed a one-time extension of the steam generator inspection interval.

2 Have we calculated what the total amount of leakage was yet?

ANSWER: No, we have not yet calculated the total leakage. We believe that the leak rate into the steam generators was a total of about four to five gallons a day just prior to the tube failure, with most of the leakage being associated with the Number 24 steam generator. This was well within the operational limit on leakage in the technical specifications. The technical specifications at IP 2 limit the leakage to 0.3 gallons per minute (or 432 gallons per day). There is a lower operational limit of 150 gallons per day in any steam generator containing tube sleeves; however, none of the IP 2 steam generators have tube sleeves at this time. It appeared that the leak rate increased to about 100 gallons per minute at the start of the event. That leak rate declined as the operators took action to shut the plant down and to lower the pressure in the reactor coolant system.

ConEd's emergency response staff in the Technical Support Center estimated that about 5500 gallons of RCS leaked into the number 24 steam generator, based on a consideration of the level change in the steam generator and the makeup from the charging system. The NRC AIT has not yet assessed the accuracy of this value. Based on a sample

of the reactor coolant taken at about 10:30 p.m. on February 15, the coolant activity at the time of the event was:

RCS total activity - 3.86XE-1 uCi/ml (same a normal level prior to event)

RCS Iodine activity - 2.02XE-2 uCi/ml (about 10X pre-shutdown level and typical for a small defect - single pin - in the fuel).

3 What do you base that on?

ANSWER: First, the results of the licensee's routine leakage monitoring, using a radiation monitor measuring N-16 indicated that some leakage, as high as about five gallons per day, was occurring in the number 24 steam generator prior to the February 15 tube failure. Also, this value was consistent with the licensee's routine sampling and monitoring of the condenser offgas.

The results of a condenser offgas sample shortly after the event began, estimated the amount of coolant leakage to be about 100 gallons per minute. Also, during the event the operators had to place a second charging pump in service to keep up with the leak rate. This would also indicate that the leakage rate was about 90 to 100 gallons per minute. This leak rate would decrease as the operators reduced the pressure in the reactor coolant system in response to the leak.

4 If Con Ed has had replacement steam generators on site since some time in the 1980s, why didn't the NRC make the company install them, especially in light of the fact that the current steam generators were leaking?

ANSWER: In the first place, the decision to replace the steam generators is Con Edison's decision, and therefore it may be best to ask them of their plans to replace the steam generators. Steam generators are not replaced for minor levels of leaking. Leaking tubes can be fixed in other ways, especially if the number of affected tubes is small. The tubes can be repaired by sleeving or taken out of service by plugging the tube. When the amount of tubes removed from service becomes large, replacing the steam generators is the viable option. However, we do have operational limits for steam generators, in the Technical Specifications, that would prohibit the plant from operating with excessive leakage. The NRC ensures that these limits are met when the plant operates.

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But as I said earlier, the decision to replace steam generators is Con Ed's.

5 How long until Con Ed will be able to get a look at the steam generator tube in question and determine what its course of action will be? Will it be done robotically?

ANSWER: ConEd is in the process of testing the tubes in all the steam generators at this time. While making the preparations for this testing, ConEd identified water leaking through one of the tubes in the No. 24 steam generator. They examined this tube with a boroscope and were able to identify small perforations in the tube structure near the top of the 'U-bend". At this time, this is the most likely cause of the February 15th event.

ConEd estimates that the steam generator tube inspections will be completed by:
SG #21- March 15
SG#22 - March 14
SG#23 - March 14
SG#24 - March 15
The above dates include actions to perform eddy current inspections, analyze data and identify the tubes to be plugged.

6 What were the results of the most recent steam generator tube inspection?

ANSWER: During the plant outage completed on 6/13/97, full length examination of all steam generator tubes was performed by the licensee. Based on this inspection, 173 tubes were plugged. The results were summarized in a letter from Con Ed to the NRC dated July 29, 1997. **[See Question 40 for additional detail]**

7 Are steam generator tube inspections done under the new inspection program? If so, by how much?

ANSWER: The steam generator tube inspections are completely unaffected by the changes to the NRC's inspection program. The tube inspections are required by the license and are conducted by the utility. Therefore the planned change to the NRC inspection program will have no affect on this.

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8 How are steam generator tubes inspected?

ANSWER: Typically, while the plant is shutdown for refueling, the licensee uses special test gear that sends a probe through the steam generator tubes that measures the wall thickness of the tubes and looks for potential weak spots or defects using an eddy-current measurement technique.

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9 **What could be the indications of an imminent steam generator tube failure?**

ANSWER: If the plant is operating, like Indian Point 2 was the other night, the earliest indication of tube failure is usually an alarming condition in the process radiation monitoring system. The steam generator produces steam that goes to the main condenser after passing through the turbine. Unless the tubes are leaking, increases in the radiation levels in the steam would not be expected. Any abnormal radiation in this steam could indicate that a tube is leaking excessively and may eventually fail.

10 **What kind of inspection of the steam generator tubes will Con Ed have to perform? Will it have to look at all four generators and all of their tubes?**

ANSWER: Until the nature of the tube failure is known, it will be difficult to predict the scope of the inspections. After the failure analysis is completed, Con Ed will develop an inspection plan, which the NRC will discuss with the licensee to ensure that adequate inspection and testing is performed to completely address the cause of the failure and to ensure that structural and leakage integrity of the tubes will be maintained until the next steam generator tube inspections are performed.

11 **Does it look like the steam generator can be fixed, or will it have to replaced? If it has to be replaced, what kind of work and time will that involve?**

ANSWER: Its too early to answer a question of this nature.

12 **Doesn't Indian Point 2 have Alloy 600 steam generators, which are considered inferior to other steam generators? Why would the NRC allow the plant to use inferior steam generators?**

ANSWER: Alloy 600 is a commonly used material for steam generator tubing. The licensee is responsible for designing and operating the plant in a manner consistent with maintaining the health and safety of the public. Through inspections and plugging of the tubes during plant shutdowns, the licensee can evaluate the structural and leakage integrity of the tubes until the next steam

generator tube inspections. The licensee relies on this information to ensure that the plant can continue to operate safely within the specified license conditions and technical specifications, so that health and safety of the public can be maintained until at least the next inspection.

13 What is the typical life expectancy for a steam generator?

ANSWER: The life expectancy for a steam generator is affected by the choice of materials, material processing history, mechanical working of the material to form the component, service environment, impact from loose parts, etc. It is difficult to predict an absolute life expectancy because of the various factors that affect the service life, but experience has shown that about half of the steam generators in service have been replaced after 20 years. Many of the replacement generators use improved alloys, so the typical life expectancy is expected to rise.

14 Historically, what was the worst tube leak? What was the nature of the event?

ANSWER: Based on data in NUREG/CR-6365, the worst tube leak was at Ginna, with a maximum leak rate of 760 gpm. This was also a W-44 SG. Reportedly, it was due to loose parts wear and fretting, due to baffle plate debris left in SG, wear of peripheral tubes, and fretting of inner tubes.

15 What was the release and the exposure as a result of the Ginna tube rupture?

ANSWER: The release was reportedly 90 Ci of noble gases, 0.4 Ci of dose equivalent I-131 (total of about 5 Ci of all isotopes of iodine); 1.3 Ci of Cobalt, Molydebnum, Barium, and Cesium; and 25 Ci of tritium may have been released.

16 What was the impact of the event?

ANSWER: Corrective measures at Ginna included removing the debris and 24 "structurally degraded" tubes, upgrading the quality control practices used for maintenance work, and installation of a loose parts monitoring system. NRC issued IN 83-24 and 88-06 discussing the effects of loose parts. GL 85-02

requested that licensees perform visual inspections on their steam generator secondary sides in the tubesheet area to identify and remove any foreign material and identify and tube damage.

17 How many steam generator tube rupture events have occurred? When and where?

ANSWER: Seven steam generator tube ruptures have occurred in the U.S. commercial nuclear power plants over the past 20 years (from NUREG/CR-6365):

2/26/75 - Point Beach 1
9/15/76 - Surry - 2
10/2/79 - Prairie Island 1
1/25/82 - Ginna
7/15/87 - North Anna 1
3/7/89 - McGuire 1
3/14/93 - Palo Verde 2

18 What was the impact of these events on the plant and surrounding area/public?

ANSWER: In all cases listed above, the plants were properly cooled down and the radioactive material releases were small and well below regulatory limits.

19 Can you identify examples of plants where leakage increased and there were no tube failures or burst?

ANSWER: Yes. Defected tubes caught by leakage before rupture occurred at the following plants:

10/23/93 - Braidwood Unit 1, 12.5 gph leak
3/9/92 - Arkansas Nuclear One, Unit 2, 15 gph leak
1/16/92 - McGuire Unit 1, 10 gph leak
12/17/90 - Maine Yankee, 84 gph leak
3/6/90 - Three Mile Island Unit 1, 30 gph leak
6/21/89 - Beaver Valley Unit 2, 21 gph leak
10/19/88 - Indian Point Unit 3, 120 gph leak

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5/16/84 - Fort Calhoun, 112 gpm leak*
* although the Fort Calhoun steam generator leak was listed in NUREG/CR-6365 as a tube rupture, the tube leak rate was less than the makeup capacity. Tube leakage in excess of makeup capacity is the usual definition of a tube rupture.

20 Did any of the tube failure events involve multiple tubes?

ANSWER: In each of the tube failure/rupture events listed in NUREG/CR-6365, only one tube ruptured in each event (although substantial degradation was noted in other tubes).

21 Is the steam generator leak in a new location?

ANSWER: The steam generator tube failure on February 15, 2000, occurred in a Row 2 tube in steam generator 24. This is the first occurrence that we aware of regarding a leak in a Row 2 tube at Indian Point 2. However, this was not the first occurrence of a steam generator tube leak from an inner row tube.

22 Is this evidence of a new type of steam generator tube degradation?

ANSWER: In the 1997 inspection, the licensee for Indian Point 2 found an axial indication in the apex, or top of the U-bend region, of a Row 2 tube in steam generator 24. This was the first evidence of a Row 2 indication in this location at Indian Point 2. This type of degradation has been previously experience by steam generators at other facilities.

23 What percentage of the other steam generator tubes will be inspected prior to startup?

ANSWER: ConEd plans to inspect all of the steam generator tubes prior to restarting the plant.

24 What was the steam generator leakage before the event?

ANSWER: Shortly before the event leakage was small, about a couple of gallons per day and trending up very slowly. The leakage rate was calculated to be as high

as about five gallons per day from the number 24 steam generator just prior to the event. This leakage before the event was well below license limits and the lower industry administrative limits.

25 What was the trend of the leakage before the event? Why didn't the NRC take some action to have ConEd shut down the plant before the event? Was NRC notified of the leak increase from 2 to 2.5 GPD?

ANSWER: **[Can DRP - P Eselgroth answer the last part of this question?]** The NRC establishes operational limits for key parameters at each nuclear plant. ConEd was well within those parameters in the days prior to the event. Otherwise, they would have been required to shut down. The leakage limits at IP 2 limit the leakage to 0.3 gallons per minute in any steam generator which does not contain tube sleeves. Leakage through the steam generator tubes and/or sleeves is limited to 150 gallons per day in any steam generator containing sleeves. This lower limit does not now apply at IP 2, since it does not use tube sleeves. Engineering data has shown that it is difficult to predict future steam generator tube behavior based on tube leakage trends. Yes, the steam generator leakage had been slowly increasing but was well within limits.

26 Aren't those leakage limits wrong, based on this event?

ANSWER: Our Office of Nuclear Reactor Regulation in Rockville regularly reviews the data from events like this to see if any changes need to be made to operational limits. They will be reviewing this data also. However, similar limits to those at IP 2 have been followed throughout the industry and have been found to be effective in allowing plants to shut down prior to large leaks or ruptures, depending on the nature of the leaking tube defect.

27 Is there an accepted leakage rate for steam generators? If so, what is it?

ANSWER: The accepted leakage rate is not the same at every plant, but the criteria used to develop an accepted leakage rate are intended to ensure that structural and leakage integrity will continue to be maintained with an acceptable level of margin consistent with the applicable GDCs of 10 CFR Part 50, Appendix A and the limits of 10 CFR Part 100. As discussed in question 26, the operational leakage performance criterion specified in NEI 97-06 is intended to ensure that

the operational primary-to-secondary leakage limit for any one steam generator does not exceed 150 gpd at room temperature conditions. By letter dated December 16, 1997, the NRC staff was informed that the industry, through the NEI Nuclear Strategic Issues Advisory Committee, had voted to adopt NEI 97-06. Each licensee will evaluate its existing steam generator program to ensure that it is consistent with NEI 97-06 no later than the first refueling outage starting after January 1,1999.

28 Why did you allow ConEd to delay inspecting their steam generator tubes in 1997? Wasn't that unsafe?

ANSWER: Based on the information submitted by the licensee with the proposed amendment to the technical specifications, we concluded that there was reasonable assurance that the steam generator tubes would maintain structural and leakage integrity for the current operating cycle. We have asked the Office of Nuclear Regulatory Research to perform an independent review of the safety evaluation.

29 Are there any other plants out there that have this same problem and could have a leak tomorrow?

ANSWER: The other plants also have operational limits on leakage in their technical specifications. As the tubes in the steam generators degrade, there is a possibility that leaks will develop. The operational limits on leakage allow the plants to detect leakage at small levels, much lower than normal reactor coolant system make-up capacity. Until we know the root cause of the problem, we cannot assess whether other plants have this same problem.

30 Isn't the NRC changing our SG leak limits to less conservative levels? Describe the 40% criteria. Why does NRC allow this?

ANSWER: Flaw acceptance criteria, termed "plugging" or "repair limits," are specified in the plant technical specifications (TSs). Traditionally, the TSs required that flawed tubes be removed from service by plugging or be repaired by sleeving, if the depths of the flaws exceeded the repair limit, which was typically 40 percent through-wall. The TS repair limits in combination with the operational assessment are intended to ensure that tubes accepted for continued service will

retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits.

The traditional strategy for achieving the objectives of the GDCs related to steam generator tube integrity has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Development of minimum wall thickness requirements to satisfy RG 1.121 was governed by analyses for uniform thinning of the tube wall in the axial and circumferential directions. The assumption of uniform thinning results in development of a repair limit that is conservative for all flaw types occurring in the field. The resultant 40-percent depth-based repair limit typically incorporated into the technical specifications is conservative for highly localized flaws such as pits, short cracks, and in particular outer diameter stress corrosion cracking.

Alternate repair criteria (ARC), used in place of the 40% depth-based repair limit, have been approved for limited application in certain plants. In the safety evaluations of the approved ARC, the staff has recognized that although the total margin may be reduced following application of the ARC, the criteria ensures that structural and leakage integrity will continue to be maintained with an acceptable level of margin consistent with the applicable GDCs of 10 CFR Part 50, Appendix A, the plant licensing basis, and the limits of 10 CFR Part 100. Because of the increased likelihood of cracks left in service, the staff has included provisions for augmented steam generator tube inspections and more restrictive operational leakage limits in the guidance for the ARC.

For some plants, a document from Nuclear Energy Institute, NEI 97-06, specifies more restrictive operational leakage limits. Some plants already use the same leakage limits as proposed in NEI 97-06. By letter dated December 16, 1997, the NRC staff was informed that the industry, through the NEI Nuclear Strategic Issues Advisory Committee, had voted to adopt NEI 97-06. The chief objective of the industry initiative is for PWR licensees to evaluate their existing steam generator programs, and where necessary, to revise or strengthen program attributes to meet the

intent of the NEI 97-06 guidelines. The NEI 97-06 guidelines are intended to improve both the quality and the consistency of steam generator programs throughout the industry. The guidance is provided in documents developed by the Electric Power Research Institute, such as the PWR Steam Generator Examination Guidelines and the PWR Primary-to-Secondary Leak Guidelines. The operational leakage performance criterion specified in NEI 97-06 is intended to ensure that the operational primary-to-secondary leakage limit for any one steam generator does not exceed 150 gallons per day at room temperature conditions.

31 Is there a limit to the number of tubes that can be plugged? Where is that limit stated?

ANSWER: The Updated Final Safety Analysis Report for IP-2, Revision 15 dated December 1999, states that the upper limit for tube plugging is 25-percent.

32 Are plants designed to accommodate a steam generator tube rupture/failure?

ANSWER: Yes, pressurized water reactors are required to analyze the effects, and design specific equipment and operating procedures to handle the consequences of a steam generator tube rupture/failure event.

33 Please explain whether the DPO that the NRC is reviewing is related to this event?

ANSWER: The differing professional opinion (DPO) from Joram Hopenfeld concerns can be grouped into five broad issues: (1) limitations of nondestructive methods, (2) primary-to-secondary tube leakage during postulated main steam line break conditions, (3) increased risk due to steam generator tube degradation and implementation of alternate repair criteria, (4) iodine spiking assumptions for radiological analyses and (5) steam generator tube integrity under severe accident conditions. Although the DPO issues are more germane to the voltage based criteria, they are related to the IP-2 event only in the general sense that there are aspects of the issues that are related to tube integrity. In addition, IP-2 has not requested an amendment to use the voltage based criteria.

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34 Shouldn't the DPO be resolved prior to allowing IP 2 to restart?

ANSWER: The focus of the DPO is different than the issues surrounding the IP-2 tube failure. Given that, there is no basis or need to tie the DPO resolution to restart.

35 Who is on the DPO Panel?

ANSWER: Jim Wiggins, Deputy Regional Administrator for Region I; Wayne Hodges, Deputy Director for Technical Review Directorate, Spent Fuel Project Office; one other panel member to be assigned.

36 Are these the oldest SG in service? The last model 44s in service?

ANSWER: IP 2 has the oldest Westinghouse Model 44 steam generators in service but they are not the oldest SGs in service. The following plants have older generators:

Plant	Date of Commercial Operation	Model SG
Oconee1	7/15/73	B&W Once-Through-SGs
Fort Calhoun	9/26/73	Combustion Engineering
Prairie Island	12/16/73	Westinghouse Model 51
Kewaunee	6/16/74	Westinghouse Model 51
Indian Point 2	8/1/74	Westinghouse Model 44

37 What type of SGs do they have on site for replacement? How many?

ANSWER: The licensee has indicated that they are thermally treated Alloy 600 Model 44F Westinghouse SGs. ? We believe there are 4.

38 What inspections are planned for the steam generators? [DRS - L. DOERFLEIN]

ANSWER: **[can DRS or DRP answer this question?]**

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39 What type of inspection was done in 1997? Did they use two types of probes?

ANSWER: They performed a combination Cecco-5/bobbin probe examination for the majority of the eddy current testing. A 700 mil diameter probe was used to perform the initial eddy current testing. Tubes that did not pass the 610 mil diameter probe were plugged. A rotating pancake probe was utilized to examine the U-bends if the narrow radii of the bends precluded passage of the Cecco-5/bobbin probe.

40 What were the results of the 1995 steam generator tube inspection and what do they mean?

ANSWER: Based on the inspection, tubes with indications measured by eddy current greater than 40% were plugged. The 1995 examination revealed Primary Water Stress Corrosion Cracking (PWSCC) in the roll transition region, which provides a tube to tubesheet interface. Since cracking was also found in this region in 1993, Con Edison had qualified an F* distance and a rerolling procedure in anticipation of additional PWSCC roll transition cracking. The F* distance is defined as the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tube sheet. This distance is equal to 1.25 inches and is measured down from the bottom of the roll transition. The tubes with PWSCC indications that did not meet the F* distance criteria were rerolled or plugged. Since the tubes are roll expanded for only a portion of the tubesheet, rerolling above the existing roll provides a new, sound, tube to tubesheet interface capable of the design criteria. 1995 Inspection Results Summary:

Steam Generator 21

2122 tubes (71.0% of all active tubes) were examined over their full length
9 Tubes Were Plugged Based on Eddy Current Indications
13 Tubes Had Distorted Roll Indications: 13 Tubes Were Rerolled Based on Distorted Roll Indications

Steam Generator 22

1475 tubes (50.5% of all active tubes) were examined over their full length
6 Tubes Were Plugged Based on Eddy Current Indications
2 Tubes Had Distorted Roll Indications: The Tubes Were Plugged Rather Than Rerolled

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Steam Generator 23

2103 tubes (70.0% of all active tubes) were examined over their full length

1 Tube Was Plugged Based on Eddy Current Indications

542 Tubes Had Distorted Roll Indications: 542 Tubes Were Rerolled Based on Distorted Roll Indications

Steam Generator 24

2991 tubes (100% of all active tubes) were examined over their full length

5 Tubes Were Plugged, 2 Based on Eddy Current Indications and 3 Due to Tube Restrictions

37 Tubes Had Distorted Roll Indications: 25 Tubes Were Rerolled Based on Distorted Roll Indications; 12 Met the F* Length

41 Regarding the 1995 SG inspection results, one SG had significantly more cracks than the others, was it SG 24?

ANSWER: No, it was steam generator 23 that had the most distorted roll indications that required re-rolling and steam generator 21 had the most eddy current indications that required plugging.

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E. Questions related to radioactive releases/environmental monitoring:

1 What was the radiation monitoring data for the on-site and off-site conditions? Did they use any TLDs in the field for assessing the release?

ANSWER: Onsite

Initially at 2000 hrs, at various onsite locations radiation and contamination surveys were performed. No evidence of a release were detected.

On 2/16 at 0050 hrs, miscellaneous inplant auxiliary building and support buildings were surveyed. No elevated readings were noted.

On 2/16 at 0315 hrs, a survey of the AFB roof indicated 24 S/G atmospheric steam dump (ASD) had elevated readings (500 cpm direct), while no contamination was detected on the roof. The 0315 hrs survey result was not communicated to personnel manning the EOF. A later, 1000 hrs, survey of the 24 ASD was poorly performed and failed to detect the presence of radioactivity. This survey was utilized in the EOF and conclusions were drawn that there was no actual release.

On 2/17 from 0300 to 0500 hrs, various inplant areas were surveyed indicating some elevated readings in the secondary plant areas (up to 2 mrem/hr).

On 2/17 at 1030hrs an NRC inspector resurveyed the 24 ASD and reconfirmed the 0315 hrs survey that there was evidence of a release. During the night shift, chemistry performed a gamma scan of the ASD and confirmed the presence of Xenon 133 and Xenon 135 (5 day and 9 hour half-lives, respectively) indicating a recent release had occurred.

On 2/18 at 1100 hrs, a manway leading inside the 24 ASD was opened and no detectable smearable contamination was found. Some elevated fixed readings were found.

On 2/19 comprehensive plant area surveys were conducted. The S/G blowdown flash tank and the secondary boiler blowdown indicated elevated readings. On 2/20 additional plant area surveys revealed elevated readings on the secondary boiler blowdown heat exchanger and flash tank.

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Offsite

On 2/15 at 2200 hrs, a radiation survey was conducted due North of the plant at the Hudson River shoreline (1.8 miles NNW). Also at 1.3 miles NE and 1 mile SW of the plant radiation and direct contamination measurements were conducted. No evidence of a release was detected.

On 2/16 from 0910 hrs to 1430 hrs, direct radiation readings and air samples (particulate and iodine) were taken at 8 offsite locations between 0.9 to 1.8 miles from the plant. No readings above background were detected.

On 2/16 at 1000 hrs, the North and South plant fence line was surveyed. All areas were <0.1 mrem/hr and no detectable contamination was found.

On 2/16 at 1700 hrs, the North and South sector pressurized ion chamber (PIC) instruments were polled for data (15 minute readings in tenths of uR/hr). Only 2 out of 8 instruments provided measurements during the release time period. No elevated readings were found.

On 2/16 IP3 roof and onsite plant areas were surveyed for contamination. None was detected.

On 2/16, 7 fixed environmental air samplers were changed and counted. They were changed out again 24 hours later and counted. No readings above background (particulate and iodine) were detected.

On 2/17 during the morning, 6 off site soil samples were taken. There were 3 North of the plant and 3 South of the plant at distances of between 1/4 and 1.8 miles. Also the NRC and State of New York took soil samples near the plant. All soil samples indicated only background activity.

On 2/18, 4 more PIC instruments were directly downloaded to provide both the North and South sector direct gamma radiation readings during the event. No elevated readings were found.

On 2/19, all of the environmental and onsite TLDs were exchanged and processed. The 82 environmental TLDs indicated background levels of 4-10 uR/hr and the 20 onsite TLDs indicated values of 7-15.2 uR/hr (18.7-47.5 uR/hr near a radwaste storage tank). None of the TLD data indicated any readings above background.

2 Do we know if the atmospheric steam dump valve lifted? In relation to that, do we have any clearer idea of what the level of radioactive release was?

ANSWER: A slight opening may have occurred for a short period of time, or its possible that only valve seat leakage may be the extent of the release through this pathway. Through a comprehensive review of the plant conditions at the time of the event, several other pathways have been identified and based on chemistry samples and flow conditions, conservative releases of radioactivity have been calculated for gaseous and liquid releases. The current draft estimate for total gaseous releases are: 1.695 Curies resulting in 0.01 mrem to the whole body at the site boundary and 0.036 mrem to the thyroid. Total liquid releases were very minor: 0.0076 Curies resulting in 7.76E-5 mrem to the whole body and 8.7E-6 mrem to the thyroid. From a review of all the environmental measurements taken, no actual exposure to the public was measured.

The Augmented Inspection Team inspection will characterize the level of the release.

3 When will the NRC know exactly how much radioactive steam was released? How will that be determined?

ANSWER: At the conclusion of the AIT inspection, the actual plant conditions and a comprehensive listing of all environmental releases of radioactive steam and radioactive liquids will be established and subsequently published in the early April timeframe.

4 Was environmental monitoring adequate?

ANSWER: Yes.

5 What was monitored/sampled to provide assurance that no releases were measurable on or off site? Who did the monitoring/sampling?

ANSWER: (1) ConEd verified that there was no elevation of the ambient radiation levels around the plant by evaluating TLD and PIC data. (2) ConEd found no deposition of fission/activation products in soil samples.

6 What is monitored, and from what parts of the plant? What type of

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radioactivity? Plutonium? [DRS - J.WHITE/J.NOGGLE]

ANSWER: (1) Airborne Releases: Noble gases and particulates were monitored at various plant vents (by RMS or gamma spec. at the laboratory). (2) Waterborne Releases: Fission/activation products (Cs-134, Cs-137, Co-80, Iodines, etc) were measured (by gamma spec.). (3) No transuranics (plutonium, americium, etc) were detected in waterborne or airborne samples.

7 Why can't we monitor the Atmospheric Relief Valve effluent?

ANSWER: We normally don't monitor the atmospheric relief valve effluent because: (1) it is not a normal effluent release pathway by design; (2) it has too many uncertainties since it isn't a normal release point; and, (3) it results in a negligible amount of projected dose to the public.

However, even though it is not monitored, the dose contribution from this pathway can be calculated using samples of the activity in the coolant to determine the amount of radioactivity in the steam generator, the design flow conditions for the valve when its open and the duration of the time the valve is open.

8 Who sets release limits? What is the basis? Has it changed over time?

ANSWER: The NRC in its regulation, 10 CFR Part 20, sets the release limits for routine radioactive gaseous and liquid effluents. The limits in this regulation specify that the radiation dose to members of the public shall not exceed 100 mrem in a year from the operation of the NRC licensed facility. The 100 mrem limit is consistent with the recommendation of the U.S. Environmental Protection Agency. Prior to 1994, a value of 500 mrem in a year was allowed.

However, for nuclear power reactors, the NRC, in 1979, started imposing special license conditions that required nuclear power reactors to limit the amounts of radioactive material discharged in to the air and water to levels that are as low as is reasonably achievable (ALARA). To be ALARA, the nuclear power reactors are to limit the amount of radioactive material they discharge in order to maintain estimated doses to members of the public to the following: 3 mrem to the whole body and 10 mrem to any organ from radioactive liquid effluents in a year and 5 mrem to the

whole body and 15 mrem to the skin from radioactive gaseous effluents in a year. The ALARA license conditions are designed to keep the doses from radioactive effluents from nuclear power plants to levels that are well below the 100 mrem limit in 10 CFR Part 20.

9 Why do we let them release any amount of radiation?

ANSWER: Nuclear power reactors are allowed to release radioactive material because they are licensed by the NRC in accordance with the Atomic Energy Act of 1954, as amended.

10 Does the State have oversight of the effluents?

ANSWER: No, the NRC in accordance with the Atomic Energy Act of 1954, as amended, has the authority to regulate radioactive effluents discharged from nuclear power plants. The States do not have the authority to regulate the discharge of radioactive material from nuclear power plants.

11 Why did we recently lower our worker exposure limits?

ANSWER: In 1994, the NRC modified its radiation protection standard, 10 CFR Part 20 (Part 20), to reflect developments in the principles and scientific knowledge underlying radiation protection that occurred since Part 20 was originally issued more than 30 years before. These developments included updated scientific information on radionuclide uptake and metabolism, but also basic philosophy of radiation protection. The changes were made to ensure that Part 20 continued to provide adequate protection of radiation workers and public health and safety.

12 How do our limits compare with international limits? Compare the release limits with existing radiation levels.

ANSWER: Each country sets its own standards; however, the standards are generally based on recommendations from the International Commission on Radiological Protection (ICRP). In developing its regulations the NRC has generally followed the basic radiation protection recommendations of the ICRP and its U.S. counterpart, the National Council on Radiation Protection and Measurements (NCRP). The ICRP recommended annual public dose limit of

100 mrem is the same limit in used Part 20. Thus, while there may be some differences in specific limits, the level of protection to members of the public is identical.

Radiation or radioactivity potentially detected in the Indian Point environment can be grouped into two main categories. The first is " naturally-occurring" radiation and radioactivity. This is the major source of human radiation exposure. A comprehensive review of the sources of natural background radiation and the resultant doses received by the population of the United States was performed by the National Council on Radiation Protection and Measurements (NCRP, 1987). The study showed that the public received an annual average dose of 300 millirem.

The second source of radioactivity is due to releases from the Indian Point power plant. The average projected dose to the public (individual) from radioactive liquid and gaseous effluents released in 1998 were 0.000016 millirem and 0.000022 millirem, respectively. These values are well below the limits that are required by the NRC.

13 What exactly is being released to the Hudson River?

ANSWER: Radioactive material that is produced from the nuclear fission process is routinely released from the nuclear power plant in a controlled manner. The radionuclides typically include cobalt, cesium, iron, tritium, nickel, silver, strontium, antimony, xenon, and iodine. The radioactive material released will vary from year to year based on plant power levels as well as the types of clean-up systems used.

In 1998, the licensee made 68 batch releases which contained 0.28 curies of fission and activation products to the Hudson River. This amount, 0.28 curies, represents less than 0.01% of the operational limits permitted in the IP 2 technical specifications.

14 If the radiation leak was too small to be measured, how did the SJAE detect it?

ANSWER: The SJAE system has a process radiation monitoring system (RMS) as part of its design. This RMS is calibrated during each refueling outage. Based

on the calibration results, ConEd sets various alarm setpoints for this equipment. When the radiation level reaches any of these setpoints, it causes alarms or actuations of equipment that is monitored by the plant operators. This radiation monitor has an excellent sensitivity to detect a very low noble gas concentration.

15 Was it below EPA limits?

ANSWER: Yes, it was well below EPA limits.

	Licensee's Data	EPA Limits
Total Body Dose	1.02E-2 mrem	25 mrem
Organ Dose	3.59E-2 mrem	25 mrem

16 Were we aware of the release? Was NRC on site at the time of the release?

ANSWER: **[Can DRS - J White OR DRP - P. Eselgroth answer this question?]**

17 They heard that we didn't arrive until after 10:30. Is that true?

ANSWER: **[Can DRP - P. Eselgroth answer this question?]**

18 What were all the sources of the leaks?

ANSWER: Based on the AIT inspection review a draft estimate of all the postulated release paths for gaseous and liquid releases has been compiled as listed below:

<u>Indian Point Steam Generator Tube Failure 2/15/00 Event Reported Releases</u> (3/9/00)

1. <u>Gaseous Releases</u> (Noble Gas: Xenon 133, 135; Argon 41; Krypton 85, 88, 87)

A. Containment vent (24 hrs)	1.50 Ci	4.04E-5 mrem (avg X/Q)
B. Evacuate condenser (3 hr)	9.80E-2 Ci	4.5E-3 mrem
C. 24 ASD (10 hr design leak)	3.64E-2 Ci	2.19E-3 mrem
(Also includes Iodine fraction, 1E-2)		3.44E-2 mrem organ
D. Gland Seal Exhaust	5.77E-2 Ci	3.21E-3 mrem

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E. Secondary steam leaks	2.53E-3 Ci	4.20E-4 mrem
F. Initial SJAE (45 sec)	7.1E-5 Ci	1.59E-6 mrem
G. 21, 22, 23 ASD (2.7 hr)	2.60E-5 Ci	1.45E-3 mrem, organ
H. S/G 24 Blowdown (Also includes I - 131, 132, 133)	3.25E-5 Ci	1.8E-6 mrem, organ
I. S/G 21, 22, 23 Blowdown (Also includes I - 131, 132, 133)	3.88E-7 Ci	2.07E-5 mrem, organ
TOTAL:	1.695 Ci	1.04E-2 mremWB 3.59E-2 mrem organ-thyroid

Gaseous Release Comparisons

Total gaseous releases represent: 0.2% of TS limit (5 mrem/qtr) Total Body
0.5% of TS limit (7.5 mrem/qtr) Organ

Noble gases are routinely released from containment. Prior to last refueling outage (Jan 1997), 80.5 Ci were vented resulting in 7.9E-2 mrem.

2. Liquid Releases

A. Groundwater inleakage released through contaminated piping from the event

2/21/00	2.78E-3 Ci	2.01E-5 mrem
2/22/00	3.37E-3 Ci	5.75E-5 mrem
B. S/G 24 Blowdown	5.59E-4 Ci	2.30E-6 mrem (organ)
C. S/G 21, 22, 23 Bldn	2.82E-4 Ci	3.00E-6 mrem (organ)
D. Secondary side leaks	5.80E-6 Ci	1.30E-6 mrem (organ)
E. Gland Seal Exhaust	5.71E-4 Ci	2.10E-6 mrem (organ)
TOTAL:	7.57E-3 Ci	7.76E-5 WB, 8,7E-6 mrem organ

Liquid Release Comparison

Total liquid releases represent: 0.005% of TS limit (1.5 mrem/qtr) Total Body
0.0002% of TS limit (5 mrem/qtr) Organ

19 How did this leak compare with past leaks? What was the largest leak in the past?

ANSWER: There have been a number of notable steam generator tube leak events over the past approximately 20 years including Palo Verde, McGuire, Surry, North Anna, Maine Yankee, and Ginna. Relatively speaking, the IP 2 steam generator tube leak was considered small in nature and of no radiological consequence. The Ginna tube rupture event in 1982 is believed to be the largest leak in terms of gallons per minute from the tube and total mass released from the reactor coolant system. In January 1982, a steam generator tube rupture occurred at the R.E. Ginna Station (Scriba, New York) resulting in 117,000 pounds of steam and water being released from a steam generator. The maximum calculated release rate was 760 gallons per minute (NUREG -0909). Although estimated airborne radioactivity released to the owner controlled property exceeded 10 CFR 20 effluent airborne concentrations for a period of time during the event, the dose consequences associated with this release were of no significance. A release of radioactivity outside the owner controlled area occurred; however, the release resulted in less than 25 % of the unrestricted area dose limits. All releases were well within 10 CFR Part 100 guidelines.

By comparison, the IP 2 steam generator tube failure resulted in an estimated, initial primary to secondary leak of about 140 gallons per minute for about the first 15 minutes that decreased after the operators tripped the plant and initiated actions to isolate the affected steam generator and reduce reactor system pressure.

20 Why do some licensees make NO liquid releases? What do they do with the effluents if they don't release them?

ANSWER: The decision to not discharge radioactive liquid effluents is made by a licensee based on the geographic location and the physical design of the facility and the cost comparison for discharge into the environment against the collection, processing, packaging and disposal in a licensed low level waste disposal facility.

When a licensee chooses to not discharge radioactive liquid waste into a river, lake, or ocean, the licensee must process the liquid waste through a radioactive waste treatment facility to covert the waste into a form suitable for disposal in a licensed low level disposal facility. Typically this means solidifying the liquid waste with concrete and packaging it in a strong durable container. The solidified waste is then shipped, in accordance with Department of Transportation and NRC regulations, to a licensed low

level waste disposal facility for burial.

21 Is there some liquid waste that was generated by this event? What kind of radioactivity is in this liquid? What is the activity level? What are they going to do with it? What are the other options?

ANSWER: The radioactive liquids were in general, drained into the liquid radwaste processing system to be treated as other primary reactor coolant liquid wastes. Due to some residual radioactive liquids in transfer piping, a small amount of radioactive liquid was released. Also four other small liquid release pathways were identified. Total radioactive liquid released was estimated to be 7.57mCi with an estimated whole body exposure to the public of 0.00008 mrem (7.57E-5 mrem). The radioactive liquids in the affected steam generator that were drained to the liquid radwaste processing system will be filtered, demineralized and sampled prior to release.

22 Why wouldn't a licensee use these options instead of releasing radiation into the river?

ANSWER: The decision to release the radioactive liquid waste into the river versus packaging it and disposing of it in a licensed low level waste facility is up to the licensee. The answer will be based on several input parameters for either option chosen; the amount of waste material, radioactivity levels, waste processing equipment capacity, manpower, dose to plant workers to process the waste, the potential dose to members of the public, availability of disposal space at the disposal facility, and cost.

23 Would you eat a fish that was caught outside the plant?

ANSWER: Yes, I would. Because:

(1) concentration at the discharge canal is so low (dilution);
(2) bioaccumulation factor is low; and
(3) projected dose will be negligible.

24 Would you take a loved one fishing there?

ANSWER: Yes, I would. Not only for fishing but also boating and swimming.

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25 Does the licensee sample the fish?

ANSWER: Yes. The licensee is required to maintain a radiological environmental monitoring program (REMP). The REMP is designed to assess the impact of the release of radioactive material into the environment. As part of the program, samples of air, water, milk, fish, and food crops from areas near and far away from the facility are collected and analyzed for radioactive material. In addition to these environmental samples, measurement of direct radiation is also conducted using special radiation detection devices located around the plant site.

The licensee is required to report the results of this monitoring to the NRC on an annual basis (typically in May of each year). This information is available to the public. As the NRC moves into its new system called " Agencywide Documents Access and Management System" (ADAMS), all new public documents will be available electronically through http://www.nrc.gov on NRC's Internal web site. This will include licensee submittals, such as the annual radiological environmental monitoring report. For help in using this system, please call the Public Document Room at 202-634-3273.

26 Isn't the Unit 1 spent fuel pool leaking 20 gallons per day?

ANSWER: Yes. Current estimates indicate that approximately 20 gallons per day of (1E-3 uCi/ml) contaminated water continues to leak into the ground around Indian Point Unit 1. Approximately 5 years ago, ConEd provided for collection of the ground water inleakage around the Unit 1 plant and processes the liquids in the liquid radwaste processing system. These liquids are filtered, demineralized, and sampled prior to discharging into the Hudson River.

ConEd last quantified leakage from the West Fuel Storage Pool in March 1998. The leak rate calculated at that time was 21.5 gpd.
Leakage from the Unit 1 pools is collected in the north curtain drain (NCD), which also collects ground water (20 gpm input) and rain water (input varies). The water collected in the NCD continues to show low levels of radioactive contamination (e.g.,Cs-137 @ 5E-5 uCi/ml), which has remained about the same as it was when the leak rate was last quantified in 1998.

In 1999, ConEd drained the water storage pool, which was one potential source of leakage to the NCD. ConEd plans to start a measurement in March 2000 to quantify the leakage from the West Fuel Pool using a boron mass balance calculation.

ConEd processes the water collected in the NCD to remove PCB contamination, and discharges the water via the normal effluent paths using a discharge permit. The water could be further processed to remove radioactivity, if needed, but usually has low levels of contamination so that no further processing is needed.

27 What are the normal release paths from the plant? Are they monitored or unmonitored? (airborne, water, etc.)

ANSWER: All routine release pathways were reviewed and approved by the NRC. (These pathways are different by each plants.) All routine release pathways are listed and monitored as required by the plant's license and described in the Offsite Dose Calculation Manual. Typical release pathways are:

(1) Airborne; plant stack, and reactor and radwaste buildings; and
(2) Waterborne; radwaste release and service water discharge

Routine and unmonitored release pathways are reviewed during NRC inspections.

28 What are the release and exposure limits for the plant?

ANSWER: The Technical Specifications provide the gaseous and liquid release limits and they are based on exposures to the public. These are:

Gaseous release limits at the site boundary: gamma air dose (noble gas) - 5 mrad/qtr and 10 mrad/yr, beta air dose (noble gas) - 10 mrad/qtr and 20 mrad/yr, iodine-131, H-3, and more than 8-day half lif particulates - 7.5 mrem/qtr and 15 mrem/yr to any organ.

Liquid release limits at the site boundary: 1.5 mrem/qtr and 3 mrem/yr total body, and 5 mrem/yr and 10 mrem/yr to any organ.

29 Are the IP2 and IP3 limits separate and independent? Is there a combined

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limit on releases/exposures?

ANSWER: Release and projected dose limits are independent. Dose limits established in Appendix I of Part 50 are based on a single unit. (e.g. radioactive liquid: 3 mrem/year for IP2 and 3 mrem/year for IP3)

However, IP2 and IP3 have a Memorandum Of Understanding, which contains radioactive liquid discharge practices, such as coordinating release time and duration. This practice implements radioactive effluent ALARA.

30 Are there different NRC and EPA release or exposure limits? What is the difference? Why is there a difference?

ANSWER: NRC's effluent/exposure limits are based on projected doses for all exposure pathways (such as inhalation, ground deposition, drinking and irrigating water supplies, fish, milk, meat, swimming, boating, etc).

EPA's dose limits (25 mrem for total body and organ, 75 mrem for thyroid) are based on other pathway, which is direct shine dose from the plant. Therefore, EPA's public dose limits play as a ceiling limits.

Historically, when the licensee met the NRC's dose, then the licensee met the EPA's dose limits automatically until some changes happened in the plant operation. Hydrogen water chemistry for the BWR is the classical example. Injecting hydrogen to the reactor coolant causes massive production of N-16 (because of ammonia) which has a high gamma energy. Shine dose from turbine building due to N-16 is the function of hydrogen flow rate into the system. The shine dose could be a controlling public dose, 25 mrem/year.

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31 How much has actually been released from the plant regularly? How does that compare to the release from this event?

ANSWER: Radioactive Material Release

	This Event	1997	1998
Airborne Release	1.695 Ci	547.0 Ci	5.2 Ci
Waterborne Release	7.57E-3 Ci	4.42E-1 Ci	2.75E-1 Ci

Total Body Dose (mrem)

	This Event (a)	1997(b)	1998 (b)
Airborne Release	1.04E-2	1.12E-2 (c)	4.76E-4 (c)
Waterborne Release	7.76E-5	2.08E-2	1.23E-2

(a) Projected dose was calculated using an accident dispersion factor. (χ/Q: 2.50E-4 sec/m^3).

(b) Projected dose was calculated using an annual average dispersion factor. (χ/Q: 2.43E-6 sec/m^3).

(c) Noble Gas Immersion Dose

32 What are the effects of these releases? Has anyone ever studied the number of cancer cases that occur around nuclear plants (IP 2) ?

ANSWER: The average radiation exposure to an individual in the United States is about 360 millirem per year. About 300 millirem of this is from natural sources, including radon from the ground, cosmic, terrestrial, and internal sources from the foods we eat. The largest man-made source is medical diagnosis, accounting for about 50 millirem per year. Consumer products such as smoke detectors, signs, like exit signs on the highways, and luminous watch dials contribute about 10 millirem per year. Radiation from releases at nuclear power plants contribute very little to this average dose to the public. The NRC limits the maximum radiation dose to the public around the plants to 100 millirem per year.

The effects of radiation on living cells may result in three outcomes: (1) cells repair themselves, resulting in no damage; (2) cells die, much like millions of body cells do every day, and are replaced through normal body processes; or (3) cells change their reproductive structure.

Although radiation is known to cause cancers at high doses and high dose rates, currently there are no data to unequivocally establish the occurrence of cancer following exposure to low doses and dose rates, those below about 20,000 millirem. Studies of a population exposed to chronic low-levels of radiation above normal background have shown no biological effects. This population includes radiation workers and people living in areas having high levels of background radiation (above 1000 millirem per year).

In the absence of sufficient data to the contrary, the radiation protection community assumes that any amount of radiation may pose some risk and that the risk is higher for higher dose levels. The NRC's dose limits for members of the public were developed on that basis.

Regarding your question about studies of cancer cases around IP 2, the New York Department of Health says that any information on increased health problems around a nuclear plant can be addressed to them at:

New York State Department of Health
Bureau of Chronic Disease and Epidemiology

I can provide you with their phone number if you like:

(518) 474-2354

33 Were most of the gas releases vented from containment? How much?

ANSWER: Yes. Approximately 1.5 Curies of the total 1.695 Curies released was due to containment venting after the event.

34 What do the secondary plant surveys show in radioactivity?

ANSWER: Radiation surveys conducted in the secondary plant areas indicate that the secondary boiler blowdown and heat exchanger were contaminated. Other areas may become identified during later evaluations.

35 What is the difference between a normal noble gas release (via waste gas

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decay tank holdup) and the prompt release of noble gases?

ANSWER: Noble gases are produced either by fission or activation processes. Production rate is different for each noble gas. Total activity is also function of gamma abundance and its energy, and operating conditions. Therefore, it is not a simple task to estimate noble gas activity. The simple answer to this question is that noble gases with a short half-life are not normally present in a normal gas release and a prompt release may not have sufficient hold-up time to allow these short-lived noble gases to decay. Therefore, a prompt release would include a mix of both short-lived and long-lived, radioactive noble gases.

36 Was the containment atmosphere sampled prior to venting? What was that release?

ANSWER: Yes. 1.5 Curies of noble gases were batch released. During the previous January 1997 venting of containment for a refueling outage 80.5 Curies were vented according to normal TS permits.

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F. Questions Related to Emergency Response:

[ALMOST ALL OF THE FOLLOWING ANSWERS ARE ISSUES THAT THE AIT IS REVIEWING - CAN WE PROVIDE DETAILS OF WHAT WE KNOW OR SUSPECT PRIOR TO HAVING THE AIT EXIT?]

1 Did the NRC keep state and federal response organizations informed of the event and its status? How was this done?

ANSWER: The NRC contacted NY State to share information on the event and status of response by each organization. The federal agencies were notified by the HQ response organization per procedure. The region also contacted the DOE Rad Assistance Program team at Brookhaven National Lab to notify them of the situation and maintian information exchange periodically. Region I and HQ coordinated contacts with NYS and other federal responders throughout the event.

2 Why didn't the counties order an evacuation?

ANSWER: The State and Counties were following the event and concluded that there never was any danger to the public and the plant was stable and proceeding to an orderly cold shutdown.

3 Did Con Ed properly notify everyone it was supposed to during the event and within the prescribed times?

ANSWER: We know that Con Ed made the immediately-required notifications after classifying the emergency at the Alert level as required by its emergency plan in a timely manner.

4 Will the NRC be reviewing how that process went?

ANSWER: Yes. This is a significant item for the AIT to review and assess.

5 What is the NRC's overall view of the plant's performance in that area?

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ANSWER: Give the "That is why we are conducting an AIT" answer.

6 Did the operators take timely actions to prevent or minimize any releases and bring the plant to cold shutdown?

ANSWER: This is a significant item that is being reviewed by the AIT.

7 Did the operators follow a defined plan/procedures in this process?

ANSWER: This is a significant item that is being reviewed by the AIT.

8 Were the plant's Emergency Operating Procedures properly followed (exited and entered)?

ANSWER: This is a significant item that is being reviewed by the AIT.

9 Why was the ADV setpoint not "dialed up" prior to isolating the steam generator?

ANSWER: This is a significant item that is being reviewed by the AIT.

10 What were the difficulties in properly adjusting Residual Heat Removal cooling, and why were these not anticipated?

ANSWER: Overall, the plant operators achieved their objective to safely shutdown the plant. Our AIT will determine if there are some details that could have been done differently.