Petitioners Presentation to NRC Staff Regarding Steam Generator Safety Issues at Indian Point Unit 2

#5

INTRODUCTION

The central issue in the petition pits useable life vs. loseable life. Con Ed acquired replacement steam generators years ago. The company does not want to swap out the existing steam generators until their useful life is completely exhausted. That's a business decision based primarily on economics.

There's a larger issue involved. Steam generator are the final barrier between highly radioactive material and the environment. When that barrier is breached, concerns about useable life are superceded by concerns about loseable life - the health and well-being of plant workers and members of the public.

The petitioners will show that efforts by Con Ed and NRC to control steam generator tube degradation at Indian Point Unit 2 failed. There is no reason to believe that these efforts will be any more successful in the future. Because Con Ed and the NRC are not infallible, and because the people working at and living around the plant are not immortal, the petitioners request the NRC to require the following three actions to be taken before allowing Indian Point Unit 2 to resume operation:

1. Replace all four steam generators (see page 2)

2. Resolve Dr. Joram Hopenfeld's formal concerns about steam generator tube ruptures (see page 12)

3. Provide potassium iodide (KI) tablets for the surrounding population (see page 17)

Dr. Hopenfeld's concerns are over eight (8) years old. Sadly, this is the "youngest" of our three items. Con Ed acquired replacement steam generators for IP2 in 1989 and has waited longer than a decade to put them in. Nearly two decades ago, inquiries following the Three Mile Island core meltdown accident recommended that KI tablets be provided as a prudent public health measure. Thus, our petition deals with public health and safety issues that are 8, 10, and 20 years old.

The NRC and the nuclear industry are fond of saying that safety is their top priority. Actions may speak louder than words, but in this case inactions speak the loudest. Dr. Hopenfeld's concerns have not been resolved. The replacement steam generators have not been installed. KI tablets have not been provided.

The NRC must put the loseable lives of plant workers and members of the public ahead of the useable life of four large pieces of metal and grant all three items requested in our petition.

A127

ITEM # 57

Nuclear Information & Resource Service
Public Citizen's Critical Mass Energy Project

PACE Law School Energy Project
Union of Concerned Scientists

(1a)

PETITION ITEM 1: REPLACE IP2'S STEAM GENERATORS

The petitioners request that the NRC not permit Indian Point Unit 2 (IP2) to be restarted with the existing steam generators. While the NRC cannot order Con Ed to replace the steam generators at IP2, the NRC has the statutory ability - and the moral obligation - to prevent this facility from restarting with the existing degraded steam generators.

The existing steam generators at Indian Point Unit 2 were purchased from Westinghouse Electric Corporation. They are Westinghouse Model 44 steam generators with tubes made from Alloy 600 metal. This material fact is important because:

> Steam generator tubes made of a particular metal alloy, known as Alloy 600, have exhibited widespread degradation as a result of a variety of corrosion and mechanical factors. This has contributed to seven steam generator tube rupture events, numerous forced reactor shutdowns, extensive tube repairs and outage extensions, significant occupational exposure of personnel to radiation and steam generator replacement at 22 plants. Eleven plants are planning to replace their steam generators in the next five years. Steam generator tube degradation also contributed to the decision to permanently shut down the Trojan nuclear power plant in Oregon, and other licensees may choose to close plants in cases where repair or replacement of the components proves economically prohibitive.[1]

Eight (8) nuclear power reactors operated in the United States with Westinghouse Model 44 Steam Generators. Seven (7) of these reactors have replaced their steam generators. The only nuclear power reactor in the United States still operating with Model 44 steam generators with Alloy 6000 tubes is Indian Point Unit 2:

History of Westinghouse Model 44 Steam Generators[2]			
Plant	Commercial Date	Replacement Date	SG Lifetime
Ginna	07/70	06/96	25 yr 11 mo
Point Beach 1	12/70	03/83	12 yr 4 mo
H. B. Robinson	03/71	10/84	13 yr 7 mo
Point Beach 2	10/72	12/96	14 yr 2 mo
Turkey Point 3	12/72	04/82	9 yr 5 mo
Turkey Point 4	09/73	05/83	9 yr 8 mo
Indian Point 2	08/74	N/A	N/A
Indian Point 3	08/76	06/89	12 yr 10 mo

Unless common sense prevails, Indian Point Unit 2 will set a dubious record in August of this year - the most aged Model 44 Steam Generators ever used in a United States nuclear power plant.

Plant owners replaced the steam generators at Point Beach, Turkey Point, and the other sites because the Alloy 600 tubes experienced more degradation than anticipated.

What is degradation, specifically tube degradation?

> Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

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[1] Nuclear Regulatory Commission, TIP:27, "Steam Generator Tube Issues," September 1999. [http://www.nrc.gov/OPA/gmo/tip/tip27.htm]

[2] http://www.nrc.gov/NRC/REACTOR/IP/historymodel44.html, March 20, 2000.

> Degraded Tube is a tube, or sleeved tube, that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.[3]

Twenty percent degradation means that the imperfection affects twenty percent of the tube's wall thickness. In other words, if a tube's wall was 100 things wide, it could have an imperfection of up to 19 things deep and still not be considered degraded.

Nuclear power plants can continue to operate with degraded tubes, but only up to a certain point:

> Tubes shall be considered acceptable for continued service if depth of degradation is less than 40% of the tube wall thickness.[4]

For the example of the 100 thing wide tube wall, a nuclear plant could continue operating as long as the indicated imperfection is less than 40 things deep.

What happens when degraded tubes are found?

> Once tube degradation has occurred, the goal is defect management - that is, to ensure that damaged tubes that could leak or rupture during the next operating cycle are identified and then either repaired or removed from service.[5]

Why is "defect management" important?

> Steam generator tube degradation needs to be controlled to prevent a significant increase in the risk profile of a pressurized water reactor.[6]

To significantly increase the risk profile of a nuclear power plant is to significantly increase the chances of an accident or the adverse consequences from an accident, or both. Thus, the danger to the public is significantly increased when steam generator tube degradation is not properly controlled.

Indian Point Unit 2's operating license requires the steam generator tubes to be periodically inspected for degradation. When tube degradation reaches or exceeds 40 percent, the tube must be repaired or plugged. Plugging a tube removes it from service. Con Ed attempts to control steam generator tube degradation by periodically inspecting the tubes and plugging degraded tubes.

Indian Point Unit 2 entered its 1993 refueling outage with 1072 tubes already plugged.[7] Fifty nine (59) other tubes were plugged during the 1993 outage.[8] The 1995 outage saw twenty one (21) additional tubes

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[3] Stephen B. Bram, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Sleeving and Acceptance Criteria," April 13, 1994.

[4] Stephen E. Quinn, VicePresident, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Sleeving Commitments," August 29, 1995.

[5] John Douglas, "Solutions for Steam Generators," *EPRI Journal*, May/June 1995.

[6] P. E. MacDonald, V. N. Shah, L. W. Ward, and P. G. Ellison, Idaho National Engineering Laboratory, "Steam Generator Tube Failures," NUREG/CR-6365, April 1996.

[7] Stephen B. Bram, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Steam Generator Tube Inservice Examination 1993 Refueling Outage," May 13, 1993.

[8] Stephen B. Bram, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Steam Generator Tube Inservice Examination 1993 Refueling Outage," May 13, 1993.

plugged.[9] During the refueling outage in 1997, 173 more tubes were plugged. Thus, IP2 entered the new millenium with 1,325 tubes plugged.

In addition to inspecting tubes and plugging degraded tubes, Con Ed monitors steam generator tube integrity while Indian Point Unit 2 is operating:

> The licensee stated that, should unforeseen circumstances cause SG [steam generator] tube leakage, there are multiple methods available to monitor primary-to-secondary leakage through the SGs. They employ radiation monitors in the condenser air ejector, the SG blowdown line, and the main steamline (MSL). In addition, MSL N-16 monitors are installed, which significantly enhance monitoring of MSL activity. In addition TS 3.1.F.2.a.(1) limits the primary-to-secondary leakage to 0.3 gallons per minute (gpm) for any one SG. However, the licensee maintains an administrative limit of 0.1 gpm.[10]

Have Con Ed's efforts to control steam generator tube degradation at Indian Point Unit 2 been successful? The success criteria as defined by Con Ed and by the Nuclear Regulatory Commission are:

- The Indian Point Unit No. 2 steam generator inservice inspection program is based upon the guidance contained within Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, dated July 1975. The purpose of this surveillance is to provide reasonable assurance of equipment integrity necessary to operate without experiencing tube rupture or tube leakage in excess of specified limits.[11]

- The RG 1.121 criteria for establishing operational leakage rate limits require a plant shutdown based upon a leak-before-break consideration to detect a free span crack before a potential tube rupture.[12]

- In addition to the steam generator inspections required by their technical specifications, both Indian Point Nuclear Generating Units 2 and 3 are required to monitor primary-to-secondary leakage to ensure that, in the event that steam generator tubes begin to leak, operators will be able to bring the plant to a depressurized conditions before a tube ruptures. [13]

- The staff finds the licensee's leakage monitoring program provides assurance that should a leak develop during the operating cycle it would be quickly detected allowing immediate mitigating actions to be taken before tube rupture occurs.[14]

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[9] Stephen E. Quinn, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Steam Generator Tube Inservice Examination Program 1997 Refueling Outage," July 29, 1997.
[10] Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 201 to Facility Operating License No. DPR-26 Consolidated Edison Company of New York, Inc. Indian Point Nuclear Generating Unit No. 2 Docket No. 50-247," June 9, 1999.
[11] A. Alan Blind, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency," December 7, 1998.
[12] Stephen E. Quinn, VicePresident, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Sleeving Commitments," August 29, 1995.
[13] William T. Russell, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Director's Decision Under 10 CFR 2.206," DD-96-06, June 10, 1996.
[14] Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 201 to Facility Operating License No. DPR-26 Consolidated Edison Company of New York, Inc. Indian Point Nuclear Generating Unit No. 2 Docket No. 50-247," June 9, 1999.

Successful control of steam generator tube degradation means that inspection (including degraded tube repairing and plugging) and monitoring activities prevent steam generator tube ruptures. Judged against the company's and the NRC's standards, Con Ed failed to control steam generator tube degradation at Indian Point Unit 2:

> On February 15, 2000 at 19:29, Eastern Standard Time (EST), with Indian Point Station, Unit 2 operating at 99 percent reactor power, operators manually shut down the unit and declared an Alert due to a primary to secondary leak in 23 steam generator. ... Operators began cooling down and depressurizing the reactor coolant system as required by procedures. 24 steam generator was isolated at about 20:34.
>
> Prior to this event, primary to secondary leakage was approximately 3.5 gpd. Leakage was being closely monitored. The R-49 steam generator blowdown monitor showed an upward trend. At about 19:19, the pressurizer level started to decrease. With alarms received from R-61D and 24 steam generator secondary system radiation monitor (R-55D), indications were that there was a substantial primary to secondary leak in 24 steam generator. Operators entered Abnormal Operating Instruction (AOI)-1.2, "Steam Generator Tube Leak." At 19:19 a second charging pump was started to maintain pressurizer level.
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> By 19:22, steam generator blowdown was isolated. At 19:29, the primary to secondary leakage was beyond the capacity of a single charging pump, and the reactor was manually tripped. [15]

According to Con Ed, an already leaking, degraded tube ruptured at or around 19:19pm. Operators had to manually start another charging pump because water level inside the pressurizer (a large metal tank connected to the primary side) was dropping. The pressurizer level was dropping because primary system water was leaking through the ruptured tube inside the steam generator.

This event occurred many years after Con Ed sued Westinghouse, the supplier of the Model 44 steam generators. On what grounds did Con Ed sue Westinghouse?

> Despite identification and continuing acknowledgement by Westinghouse of the degradation of the steam generators caused by corrosion and other factors and requests by Con Edison that Westinghouse correct these defects in the steam generators by stopping the process of deterioration, corrosion, denting, closing and cracking, which by the terms of the IP 2 Agreement Westinghouse was required to do at no cost to Con Edison, Westinghouse has failed to do so or has been unable to do so.[16]

Substitute "Con Edison" for "Westinghouse" and either "NRC" or "the public" for "Con Edison" and this indictment is as valid today as it was years ago. Con Ed knows that the steam generators at Indian Point Unit 2 are degraded. They sued Westinghouse over their degraded conditions and acquired replacements. But after collecting money from the ratepayers for the replacement steam generators[17] and settling with Westinghouse for the old steam generators, Con Ed has failed to or has been unable to replace the degraded steam generators.

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[15] Consolidated Edison Company of New York, Inc., Licensee Event Report No. 2000-001-00, "Manual Reactor Trip Following Steam Generator Tube Rupture," March 17, 2000.

[16] Consolidated Edison Company of New York, Inc., against Westinghouse Electric Corporation, Complaint filed in United States District Court Southern District of New York, Civil Action No. 82 Civ. 3504 (MEL).

[17] David A. Schlissel, Schlissel Technical Consulting Inc., "Indian Point 2 Steam Generator Issues," March 10, 2000.

The petitioners have identified six (6) reasons why the steam generators at Indian Point Unit 2 must be replaced prior to restart:

1. The nuclear industry and the NRC have a poor track record of controlling steam generator tube degradation.

2. Radiation exposures to workers at IP2 will be reduced.

3. Safety margins will be significantly increased by improved heat transfer capabilities.

4. Safety margins will be significantly increased by reducing dependence on operator actions.

5. Safety margins will be significantly increased by clarifying whether Con Ed conforms with accepted industry practice.

6. A financial analysis concluded that the company lost money by not replacing the steam generators during the 1997 opportunity and could lose more money if they are not replaced now.

Any one of these six reasons provide ample justification for replacing the steam generators. Combined, they provide irrefutable, overwhelming evidence of the absolute need for replacement.

The nuclear industry and the NRC have a poor track record of controlling steam generator tube degradation.

The NRC recently admitted that the basis for waiving a requirement to inspect the steam generator tubes at Indian Point Unit 2 last year was unsound:

> Based on the information we have reviewed, we believe the licensee's assessment of two forms of degradation found in their generators was inadequate: (1) ODSCC above the top of the tubesheet location (sludge pile); and (2) PWSCC at a row 2 U-bend. We believe that a more thorough operational assessment for these forms of degradation would have predicted an increased probability of tube leakage or rupture by the end of cycle 14.[18]

Con Ed inspected the steam generator tubes in 1997, plugged 173 tubes, and concluded that the remaining tubes were acceptable for operation:

> The 1997 steam generator tube inservice examination demonstrates that the Indian Point Unit No. 2 steam generators are acceptable for continued service at full power.[19]

Last year, Con Ed was required to inspect the steam generator tubes again. The company sought and obtained permission from the NRC to defer that inspection requirement until this year:

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[18] Ashok C. Thadani, Director - Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, to Samuel J. Collins, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Request for Independent Reviews of May 26, 1999, Safety Evaluation Regarding Steam Generator Tube Inspection Interval and February 13, 1995, Safety Evaluation Regarding F* Repair Criteria for Indian Point Station Unit 2," March 16, 2000.

[19] Stephen E. Quinn, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Steam Generator Tube Inservice Examination 1997 Refueling Outage," July 29, 1997.

> Based on a review of this portion of the licensee's assessment, the staff expects the SG [steam generator] tubes will continue to satisfy structural and leakage integrity requirements under normal and accident conditions through the end of the current operating cycle (14). This conclusion is based on: 1) the licensee's comprehensive eddy current examination and plugging practice at EOC [end of operating cycle] 13; 2) the growth rates of the degradation mechanisms are expected to be similar to what was seen for cycle 13 operation; and 3) the licensee's acceptable in-situ testing results on the limiting EOC 13 indications.[20]

Hindsight is always 20/20 vision. The NRC's Office of Nuclear Regulatory Research reviewed the data available to Con Ed and to the NRC staff from the 1997 inspections:

> For the first time, a Row 2 U-bend PWSCC [primary water stress corrosion cracking] indication was found. The dimension of the indication by the +Point characterization was below the in-situ screening threshold for Row 2 U-bend flaws. ... As this represents the first detected U-bend indication after approximately 23 years of operation, any growth rates associated with this indication would be considered minimal.[21]

Which tube ruptured during the February 15th event at IP2?

> The location of the tube leak in 24 steam generator has been identified at Row 2, Column 5 near the top, outer radius of the U-bend. ... Preliminary analysis indicates that the cause of the tube failure is primary water stress corrosion cracking (PWSCC).[22]

Thus, Con Ed's control of steam generator degradation was, in the NRC's words, "inadequate." The NRC staff failed to catch Con Ed's inadequate work and mistakenly approved the inspection deferral. As it often the case, two wrongs did not make a right. Plant workers and people living near the plant faced the needless risk of a nuclear plant accident. Hindsight may have revealed the mistakes, but they were caused by shortsightedness by Con Ed and the NRC staff.

IP2's tube rupture event is but the most recent example of the nuclear industry and the NRC failing to adequately control steam generator tube degradation. Two other examples occurred in 1996:

> During refueling outage 2R12 [at Arkansas Nuclear One Unit 2] in May and June of 1997, eddy current data from Steam Generator tube eddy current testing was compared with data obtained in late 1996 in outage 2F96-1. It was determined that up to 25 tubes could have been dispositioned as distorted support indications (DSIs) and further characterized with motorized rotating pancake coil (MRPC) but were not. This hindsight analysis to evaluate flaw growth rate revealed that the indications in these 25 tubes could have been greater than 40 percent through-wall during startup from outage 2F96-1. The tubes remained in service for approximately five months until 2R12 when they were removed from service by mechanical plugging.[23]

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[20] Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 201 to Facility Operating License No. DPR-26 Consolidated Edison Company of New York, Inc. Indian Point Nuclear Generating Unit No. 2 Docket No. 50-247," June 9, 1999.
[21] James S. Baumstark, Vice President - Nuclear Engineering, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Response to Request for Additional Information - Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency," May 12, 1999.
[22] Consolidated Edison Company of New York, Inc., Licensee Event Report No. 2000-001-00, "Manual Reactor Trip Following Steam Generator Tube Rupture," March 17, 2000.
[23] Dwight C. Mims, Director - Nuclear Safety, Entergy Operations, to Nuclear Regulatory Commission, "Licensee Event Report 50-368/97-008-00," October 27, 1997.

> In 1996, an inappropriate repair of the cold leg sentinel plugs for tubes R9C60 and R10C60 in the Salem Unit 2 , 24 Steam Generator was performed. The repair installed a Plug-a-Plug)(PAP) behind each of the existing sentinel plugs. This repair eliminated the ability of the sentinel plugs to allow a small amount of primary-to-secondary leakage as an indication of tube fatigue cracking.[24]

As a result of the Arkansas Nuclear One mistake, twenty five degraded tubes remained in service for five months when they should have been removed from service. This was not the first such mistake made at Arkansas Nuclear One:

> The Arkansas Power and Light Company, the licensee for ANO-2, found three tubes to be degraded to the point where they no longer retained adequate structural margins to sustain the full range of normal operating, transient, and postulated accident conditions without rupture.[25]

As a result of the Salem mistake, two tubes were improperly plugged such that further degradation would not have been detected. Shortly after Salem reported finding two improperly plugged tubes, Con Ed found a problem plug at Indian Point 2:

> During the normal scanning examination of the tubesheet it was noted that an explosive plug (SG 23 CL Row 2 Column 5) was missing the "skirt" portion of the plug. The "skirt" region is the portion of the plug located below the plug's pressure sealing surface. Westinghouse stated that this was a first-time occurrence. This explosive plug was installed in 1978.[26]

Replacing the steam generators at Indian Point 2 will essentially reset the clock on tube degradation. Replacement will not magically cure the poor track record of the nuclear industry and the NRC in controlling steam generator tube degradation. But it will provide greater safety margins and make public health and safety less dependent on mistake-free performance.

Radiation exposures to workers at IP2 will be reduced.

In the late 1980s, the New York Power Authority (NYPA) replaced all four steam generators at their Indian Point Unit 3 facility. One of the primary benefits from that replacement, according to NYPA, was the reduction in radiation exposures to plant workers. NYPA told the NRC that workers at Indian Point Unit 3 averaged more than 269 person-rem on steam generator related activities each year from 1982 to 1987 inclusive.[27]

Con Ed reports that its workers also receive sizeable exposures from the steam generators:

> Activities associated with a steam generator inspection for Indian Point Unit No. 2 typically incur a radiation exposure of approximately 40 person-rem per inspection.[28]

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[24] A. C. Bakken III, General Manager - Salem Operations, to Nuclear Regulatory Commission, "LER 311/98-003-00," June 29, 1998.
[25] Nuclear Regulatory Commission, Information Notice 92-80, "Operation With Steam Generator Tubes Seriously Degraded," December 7, 1992.
[26] Stephen E. Quinn, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Steam Generator Tube Inservice Examination 1997 Refueling Outage," July 29, 1997.
[27] Marylee M. Slosson, Nuclear Regulatory Commission, "Summary of Meeting Held on July 30, 1987 to Discuss Replacement of Indian Point 3 Steam Generators," August 11, 1987.
[28] A. Alan Blind, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency," December 7, 1998.

Replacing the steam generators at Indian Point 2 will essentially reset the source term associated with the steam generators. The contaminated steam generators will be removed and replaced with radioactively clean steam generators. The scope of future tube inspections should be less for the new steam generators than for the old steam generators. The reduced inspection scope and the lower source term will result in lower radiation exposures to plant workers.

Safety margins will be significantly increased by improved heat transfer capabilities.

Six years ago, Con Ed sought permission from the NRC to sleeve degraded tubes instead of plugging them. Sleeving allows degraded tubes to remain in service using a "patch" over the degraded area. Con Ed justified its request, in large part, on the fact that sleeving improves the heat transfer capabilities of the steam generators:

> The heat transfer capabilities of Indian Point 2 Steam Generators will be improved by utilizing the proposed sleeving process or implementing the F* criteria rather than the currently required tube plugging and subsequent loss of heat transfer area.[29]

Replacing the steam generators at Indian Point Unit 2 will maximize their heat transfer capabilities. More than one thousand plugged tubes will be returned to service, recovering that lost heat transfer area. Restoring the steam generators to their original, non-degraded condition is far, far better than sleeving.

Safety margins will be significantly increased by reducing dependence on operator actions.

Following most design basis accidents, the control room operators have a relatively passive role as the plant's engineering safety features automatically function. They monitor plant conditions and verify proper actuation of automatic actions. The operators' role following a steam generator tube rupture is significantly more active.

> During a tube rupture transient, the reactor operators are expected to (a) maintain the primary coolant subcooled, (b) minimize the leakage from the reactor coolant system to the faulted steam generator secondary side, and (c) minimize the release of radioactive material from the damaged steam generator.[30]

Instead of monitoring plant conditions and taking compensatory measures when automatic equipment actions have not occurred, the operators must manually take actions following a steam generator tube rupture that would not otherwise be taken automatically. According to an independent assessment performed by the Idaho National Engineering Laboratory, the performance of operators during actual steam generator tube rupture events has not always been exemplary:

> The success of the reactor operators has been mixed, some were slow to understand what was occurring, slow to start reducing power, and slow to isolate the defective steam generator. Others reduced power and isolated the faulted steam generator promptly. Some operators were slow to cool and depressurize the primary system, other took prompt action. The result was that the

[29] Stephen B. Bram, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Sleeving and Acceptance Criteria," April 13, 1994.
[30] P. E. MacDonald, V. N. Shah, L. W. Ward, and P. G. Ellison, Idaho National Engineering Laboratory, "Steam Generator Tube Failures," NUREG/CR-6365, April 1996.

> faulted steam generators were overfilled in a number of cases and more radioactive material was released to the environment than necessary.[31]

According to Con Ed, operators identified the faulted steam generator at Indian Point Unit 2 and isolated it around 20:34 pm on February 15, 2000, or 75 minutes after the tube ruptured at 19:19 pm. That's only thirty minutes *longer* than assumed in the safety analyses performed for Indian Point Unit 2:

> Operator action time required to terminate break flow is 45 minutes or less, as demonstrated by simulator testing.[32]

Unfortunately, the February 15th event was not a simulator exercise but a bona fide emergency situation. The operators' delay meant that much more radiation would have been released had it not been for luck.

The slow response to the February 15th steam generator tube rupture is not the only recent example of operator performance problems at Indian Point Unit 2. Just last year, the NRC reported:

> The [NRC] inspectors reviewed Con Edison's response to a rod insertion event. The operators did not recognize the event for approximately two hours because of inadequate control board monitoring, incorrect record-keeping, and inadequate audible cues that automatic control rod motion was occurring.[33]

The silver lining in IP2's cloud is that operator performance during the February 15th event was better than during the prior steam generator tube rupture event. At 04:34 am on March 14, 1993, a tube ruptured at Palo Verde Unit 2 in Arizona. The leak rate through the ruptured tube was approximately 250 gpm, or more than twice the rate encountered at IP2. It took the control room operators at Palo Verde Unit 2 over two hours just to figure out they were dealing with a steam generator tube rupture event.[34]

Replacing the steam generators at Indian Point Unit 2 will reduce the degradation level of the tubes. In turn, the plant's safety level will increase because the likelihood of a tube rupture is reduced. By taking action to reduce the likely challenge to the operators, the public is better protected.

Safety margins will be significantly increased by clarifying whether Con Ed conforms with accepted industry practice.

Following the February 15th event at Indian Point 2, the NRC created an IP2 event page on its website, www.nrc.gov, and posted relevant information there. According to this information:

> NEI 97-06, "Steam Generator Program Guidelines," provides guidance on performing condition monitoring and operational assessments to evaluate tube integrity. In 1997, the commercial nuclear industry committed to following NEI 97-06. The chief objective of NEI 97-06 is for pressurized water reactor (PWR) licensees to improve the quality and consistency of their steam generator programs by evaluating their programs, and where necessary, revising or strengthening

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[31] P. E. MacDonald, V. N. Shah, L. W. Ward, and P. G. Ellison, Idaho National Engineering Laboratory, "Steam Generator Tube Failures," NUREG/CR-6365, April 1996.

[32] James S. Baumstark, Vice President - Nuclear Engineering, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "NEI Pilot Program for Use of NUREG-1465," February 14, 2000.

[33] A. Randolph Blough, Director - Division of Reactor Projects, Nuclear Regulatory Commission, to A. Alan Blind, Vice President -- Nuclear Power, Consolidated Edison Company of New York, Inc., "Mid_Cycle Plant Performance Review - Indian Point Unit 2," September 30, 1999.

[34] J. B. Martin, Regional Administrator, Nuclear Regulatory Commission, to W. F. Conway, Executive Vice President - Nuclear, Arizona Public Service Company, "NRC Inspection Report 50-529/93-14," April 16, 1993.

> program attributes to meet the intent of the NEI 97-06 guidelines. To confirm that adequate steam generator tube integrity has been maintained since the previous inspection, the licensee performs a condition monitoring assessment as described in NEI 97-06 and associated EPRI guidance. This assessment details the "as found" condition of the tubing relative to the performance criteria. Information for the assessment is gathered from tube inspections and other tests such as one to measure the pressure capacity of the tube. Based on the information found in the condition monitoring assessment, the licensee can perform an operational assessment to provide assurance that the performance criteria for the steam generator will not be exceeded during the next operating cycle.[35]

As this information was found on the IP2 event page, the petitioners assume that Con Ed committed to following NEI 97-06 in 1997 along with the rest of the commercial nuclear industry. However, during our review of publicly available information between Con Ed and the NRC regarding steam generators, the petitioners did not find a single reference to NEI 97-06. For example, when Con Ed sought permission in late 1998 to defer the steam generator inspection required during 1999, here's what they reported:

> The Indian Point Unit No. 2 steam generator inservice inspection program is based upon the guidance contained within Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, dated July 1975. The purpose of this surveillance is to provide reasonable assurance of equipment integrity necessary to operate without experiencing tube rupture or tube leakage in excess of specified limits.[36]

The 1975-era guidance in Reg Guide 1.83 might be compatible with the 1997-era guidance in NEI 97-06. But it is curious to the petitioners that the NRC cited NEI's guidance document while Con Ed cited NRC's guidance document.

Replacing the steam generators will provide Con Ed and the NRC opportunity to get on the same page as to the appropriate guidance for the steam generator inspection program at Indian Point Unit 2.

A financial analysis concluded that the company lost money by not replacing the steam generators during the 1997 opportunity and could lose more money if they are not replaced now.

One of the petitioners (Smeloff) will present this information at the April 7, 2000, public meeting.

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[35] http://www.nrc.gov/NRC/REACTOR/IP/index.html

[36] A. Alan Blind, Vice President, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency," December 7, 1998.

PETITION ITEM 2: RESOLVE DR. HOPENFELD'S DPO AND GSI-163

By letter dated March 22, 2000,[37] the petitioners formally requested permission from the NRC for Dr. Joram Hopenfeld to attend the April 7, 2000, public meeting and discuss the technical concerns documented in his Differing Professional Opinion (DPO). Our letter clearly stated that Dr. Hopenfeld, if permitted to speak, would only talk about his technical concerns and that the petitioners would explain why we felt the DPO must be resolved prior to restart of Indian Point Unit 2.

By letter dated March 31, 2000,[38] the NRC denied our request. In a private conversation prior to the rejection letter, Mr. Collins of the NRC advised one of the petitioners (Lochbaum) that the NRC viewed the DPO and 2.206 processes as separate and distinct. Whatever.

The petitioners present this sampling of Dr. Hopenfeld's technical concerns excerpted from publicly available documents:

> The DPO was initiated in 1991 because the NRC had begun allowing plants to operate with through-wall cracks in steam generator tubes. I felt that the NRC failed to recognize the fact that leaving cracked tubes in service could, during design and severe accidents, result in primary to secondary leakage which would exceed the leakage from a single tube rupture. The plants were not designed for such large leakages and therefore public safety was compromised.[39]
>
> My main concern is that a Main Steam Line Break (MSLB) outside containment could trigger a multiple steam generator tube failure which would than result in a core melt because of depletion in coolant inventory.[40]
>
> The Executive Director for Operations has been assuring the Commission, the ACRS and the public that the DPV/DPO will be addressed as part of the regulatory approach for solving steam generator tube integrity issues. For nine years this had been the excuse given for not resolving the DPV/DPO in accordance with established procedures.[41]
>
> A key provision of NRC Management Directive 10.159 is that a review of DPV/DPOs is to ensure "full consideration and prompt disposition of DPVs and DPOs by affording an independent impartial review by qualified personnel." Since the present DPV/DPO process has been under "consideration" for nine years, obviously the intent of "prompt disposition" has not been met.[42]

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[37] David A. Lochbaum, Union of Concerned Scientists, to Samuel J. Collins, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "2.206 Petition on Indian Point 2," March 22, 2000.
[38] Samuel J. Collins, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, to David A. Lochbaum, Union of Concerned Scientists, "Petition Pursuant to 10 CFR 2.206 - Indian Point Nuclear Generating Unit 2 (TAC No. MA8449)," March 31, 2000.
[39] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.
[40] J. Hopenfeld, "Differing Professional View," December 23, 1991.
[41] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.
[42] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.

> After six years of failed attempts to obtain a resolution to the DPO I requested that an ad-hoc panel from outside the agency be selected to address the DPO issues. My request was rejected on the ground that it will take a long time to obtain a resolution if an outside panel is involved. It is now more than three years since that request was made, and the DPO remains unresolved.[43]

[Petitioners' Note: Dr. Hopenfeld submitted three candidates to the NRC to represent him on the DPO panel. One of the petitioners (Lochbaum) was among these three candidates. The other two candidates were Mr. Donald C. Prevatte, a frequent subcontractor to the NRC and Mr. Ivan Catton, formerly of the NRC's ACRS. Lochbaum volunteered to assist the NRC resolve the DPO matter free of charge. The NRC rejected all three of Dr. Hopenfeld's candidates and put their own person on the DPO panel to "represent" Dr. Hopenfeld's interests. The petitioners question the NRC's logic in rejecting Dr. Hopenfeld's request. How can "no progress in nearly ten years" be slowed down by an outside panel?]

> In response to your November 1, 1999 request regarding the final staff Differing Professional Opinion Consideration Document (DPO Consideration), undated, I must state that none of the DPO issues has been resolved to my satisfaction. It misstates material facts, ignores major DPO documents, and focuses on minor issues instead of addressing all concerns in an objective and professional manner.[44]

> In 1997, the [NRC] staff informed the Commission that they had recently discovered that the replacement of the 40% through-wall plugging criteria would significantly increase susceptibility to tube failure during certain severe accident sequences. ... Conspicuously, the staff failed to inform the Commission that, five years earlier in 1992, a DPV [differing professional view] analysis already existed which showed that lifting the 40% plugging criteria would significantly increase the risk from severe accidents. The staff knew or should have known that such as analysis already existed.[45]

> To be credible, risk-informed regulation mandates statistically valid and scrutable data, competent insights of accident scenarios and their consequences, and of accident prevention strategies, as well as meaningful public involvement. In reality, the [NRC] staff examines accident scenarios and their consequences in a superficial manner; accident prevention is apparently dictated primarily by financial consideration, and the public is being excluded from meaningful participation in the NRC deliberation process. This situation is exemplified in the recent granting of an inspection waiver to Farley Nuclear Power Plant Unit 1. Considering that "staff beliefs" were used as a sole justification, the inspection waiver shows that public risk from aging nuclear power plants has never been greater.[46]

The petitioners regret that the NRC denied Dr. Hopenfeld the opportunity to voice his concerns publicly. We hope that we fairly conveyed his concerns with the above quotes from publicly available information.

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[43] Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "DPO Panel Review of Steam Generator Integrity," September 28, 1999.
[44] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.
[45] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.
[46] Dr. Joram Hopenfeld, Nuclear Regulatory Commission, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," December 16, 1999.

Perhaps the best evidence of the need to resolve Dr. Hopenfeld's DPO prior to restart of Indian Point Unit 2 exists in a prior Director's Decision issued by the NRC related to steam generators at Indian Point.

> In the NUREG-0844 assessment, the staff concluded that the probability of simultaneous multiple tube failures was small (approximately 10E-5), and the risk resulting from releases during steam generator tube ruptures with loss of secondary system integrity was also small.[47]
>
> I further informed the Petitioner that her request for a public meeting to explain the denial of her request for license suspension [at Indian Point 2 and Indian Point 3] was denied, primarily because the NRC assessment of risk associated with steam generator tube rupture events has already been articulated in public documents.[48]

NUREG-0844 was issued by the NRC in September 1988. Eight (8) years later, the NRC issued NUREG/CR-6365.[49] Table 21 of NUREG/CR-6365 provides the frequency of tube ruptures for various initiating events:

Initiating Event Identifier	Frequency of Rupturing for Various Numbers of Tubes			Total Tube Rupture Frequency
	1 Tube	2 to 10 Tubes	> 10 Tubes	
Loss of normal feedwater	9.5E-4	9.5E-4	2.0E-5	1.9E-3
Turbine generator trip	8.7E-4	8.7E-4	1.8E-5	1.8E-3
Reactor coolant system flow loss in one loop	9.3E-5	9.3E-5	2.1E-6	1.9E-4
Alternating current power loss secondary side	2.4E-4	2.4E-4	5.0E-6	4.9E-4
Inadvertent opening of a secondary side safety relief valve	4.4E-5	4.4E-5	9.0E-7	8.9E-5
Steam line rupture	6.6E-6	6.6E-6	1.4E-7	1.3E-5
Main feedwater line rupture	6.6E-6	6.6E-6	1.4E-7	1.3E-5
Feedwater failure that results in a flow increase in one loop	2.8E-6	2.8E-6	5.6E-8	5.7E-6
Transient with failure to scram	1.8E-6	1.8E-6	3.6E-8	3.6E-6
Loss of offsite power	1.7E-6	1.7E-6	3.5E-8	3.4E-6
Inadvertent opening of a PORV	3.6E-7	3.6E-7	7.4E-9	7.3E-7
Large or medium loss of coolant accident	7.4E-8	7.4E-8	1.5E-9	1.5E-7

Thus, the NRC staff denied the Indian Point steam generator 2.206 petition in June 1996 without mentioning or articulating why information from NUREG/CR-6365, issued in April 1996, was not considered. That information was relevant in 1996 and remains relevant today.

The reason that this information is relevant to Indian Point Unit 2 is because Con Ed calculated the core damage frequency from a steam generator tube rupture to be 1.62E-6.[50] Con Ed defined a steam generator tube rupture as:

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[47] William T. Russell, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Director's Decision Under 10 CFR 2.206," DD-96-06, June 10, 1996.
[48] William T. Russell, Director - Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Director's Decision Under 10 CFR 2.206," DD-96-06, June 10, 1996.
[49] P. E. MacDonald, V. N. Shah, L. W. Ward, and P. G. Ellison, Idaho National Engineering Laboratory, "Steam Generator Tube Failures," NUREG/CR-6365, April 1996.

> Single steam generator tube rupture from any cause.[51]

NUREG/CR-6365 Table 21 reported that the probability of rupturing ten tubes following all of the postulated events analyzed is identically equal to the probability of only rupturing a single tube. Further, this NRC-issued document reported that the probability of rupturing more than ten tubes is only an order of magnitude (factor of 10) less for many of the postulated events.

The information contained in NUREG/CR-6365 Table 21 clearly supports Dr. Hopenfeld's about multiple tube failures. Con Ed defines a steam generator tube rupture to be limited to the failure of a single tube. Therefore, Con Ed's safety analysis may not be bounding and associated implementing documents, such as the operator response times to SGTR events, may be non-conservative. The petitioners seek resolution of Dr. Hopenfeld's DPO to determine whether Indian Point Unit 2 is operating outside its design bases, a condition prohibited by federal regulations.

The NRC often tells the public that its first objective is to maintain safety. If the public is to accept "maintain safety" without appending "on the back burner" to it, the NRC staff has to expeditiously resolve Dr. Hopenfeld's concerns. He raised the concerns in December 1991. George Bush was President. Bill Clinton has since been elected President, nearly impeached, and will leave office in less than a year after his second 4-year term, yet Dr. Hopenfeld's safety concerns remain unresolved.

In September 1998, the NRC announced:

> The Nuclear Regulatory Commission staff has issued a final design approval to Westinghouse Electric Company for its AP600 standard nuclear reactor design. Issuance of the final design approval completes the NRC staff's technical review of the application for design certification received in 1992 and reflects the advice of the Commission's independent Advisory Committee on Reactor Safeguards.[52]

The NRC staff received, processed, and approved an application for an advanced reactor design that no utility in the United States has expressed an interest in building, yet Dr. Hopenfeld's safety concerns remain unresolved.

In March 1996, George Galatis was featured on the cover of *TIME* magazine for a story about safety culture problems at the Millstone nuclear plant. The NRC staff received, processed, and approved a request to restart Millstone, yet Dr. Hopenfeld's safety concerns remain unresolved.

In April 1998, BG&E submitted a license renewal application to the NRC for the Calvert Cliffs nuclear power plant. Despite having initially planned a 30-month review period, the NRC approved this first-of-a-kind request in March 2000, just 23 months after receipt. The NRC staff received, processed, and approved a request to relicense a nuclear power plant -- and one having steam generators -- ahead of schedule, yet Dr. Hopenfeld's concerns remain unresolved.

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[50] Consolidated Edison Company of New York, Inc., "Individual Plant Examination for Indian Point Unit No. 2 Nuclear Generating Station," August 1992.
[51] Consolidated Edison Company of New York, Inc., "Individual Plant Examination for Indian Point Unit No. 2 Nuclear Generating Station," August 1992.
[52] Nuclear Regulatory Commission, News Release No. 98-158, "NRC Issues Final Design Approval for Westinghouse AP600 Standard Nuclear Reactor," September 4, 1998.

The resolution of Dr. Hopenfeld's safety concerns is long overdue. The petitioners don't want to hear any more excuses or verses of the "maintain safety" song. The petitioners request that the NRC resolve Dr. Hopenfeld's safety concerns before Indian Point Unit 2 is restarted.

PETITION ITEM 3: KI TABLET DISTRIBUTION OR STOCKPILING

Why and how are potassium iodine (KI) tablets linked to the steam generators at Indian Point Unit 2? Perhaps the best answer is provided by the Idaho National Engineering Laboratory:

> To prevent the release of radionuclides, the steam generator tubing must be essentially free of cracks, perforations, and general deterioration.[53]

The presence of more than one thousand plugged tubes in the IP2 steam generators and the mandated requirement to inspect the in-service tubes frequently is *prima facie* evidence that steam generator tubes are **NOT** essentially free of cracks, perforations, and general deterioration. In fact, a steam generator tube rupture (SGTR) is a design bases event for Indian Point Unit 2. Con Ed also concluded:

> Core melt scenarios following a steam generator tube rupture in which isolation is not achieved (.e.g, a safety valves failed to reclose) were determined to be a Type I release with a calculated frequency of 3.73E-7 per year.[54]

A Type I release is defined in the Con Ed report as an event involving the release to the atmosphere of twenty (20) percent or more of the iodine and cesium inventories in the reactor core. The Indian Point 2 reactor core contains 86.3 million curies of Iodine-131.[55] If just a minimal Type I release event occurs (i.e., twenty percent), that means 17.3 million curies of I-131 will be released to the atmosphere. For perspective, the government claims that 10 million curies of radioactive material (I-131 and all other radionuclides) were released to the environment during the Three Mile Island accident.

The radioactivity of the I-131 released during such as event can have serious public health consequences as demonstrated following the Chernobyl accident in 1986. When people breath air containing I-131, the radioactive gas is removed from the air and stored or absorbed by the thyroid gland. The role of KI tablets is to saturate the thyroid with non-radioactive iodine so that any I-131 inhaled is not retained by the thyroid gland. The wisdom of using KI tablets as protection against I-131 was recognized by the NRC less than two years ago. The owner of the Calvert Cliffs nuclear plant in Maryland discovered, after nearly two decades of operation, that the control room ventilation system did not adequately protect operators during accidents:

Table 2.2.2-3 Control Room Operator Does from Postulated Accidents (Rem)[56]		
Accident	Thyroid#	Whole Body
Main Steam Line Break		
Coincident Spike	400	1.5
Pre-existing Spike	390	1.4
SG Tube Rupture		
Coincident Spike	1900	<1
Pre-existing Spike	1500	<1
Locked Rotor		

[53] P. E. MacDonald, V. N. Shah, L. W. Ward, and P. G. Ellison, Idaho National Engineering Laboratory, "Steam Generator Tube Failures," NUREG/CR-6365, April 1996.
[54] Consolidated Edison Company of New York, Inc., "Individual Plant Examination for Indian Point Unit No. 2 Nuclear Generating Station," August 1992.
[55] James S. Baumstark, Vice President - Nuclear Engineering, Consolidated Edison Company of New York, Inc., to Nuclear Regulatory Commission, "NEI Pilot Program for Use of NUREG-1465," October 8, 1999.
[56] Alexander W. Dromerick, Senior Project Manager, Nuclear Regulatory Commission, to Charles H. Cruse, Vice President - Nuclear Energy, Baltimore Gas and Electric Company, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit No. 1 (TAC No. M97855) and Unit No. 2 (TAC No. M97856)," May 23, 1998.

Coincident Spike	900	3.7
Pre-existing Spike	890	3.6

Crediting the compensatory measures committed to by the licensee, self-contained breathing apparatus and KI tablets, reduces the calculated thyroid does below the 30 rem criterion of GDC 19.

Calvert Cliffs operated for many years with a design deficiency that exposed its operators to a potential thyroid gland exposure dose of up to 63 times the maximum federal limit specified in 10 CFR Part 50, General Design Criterion 19. The NRC allowed Calvert Cliffs to continue operating until the ventilation system could be fixed because:

> The staff has again concluded that with such timely and appropriate application of compensatory actions afforded by the SCBAs [self-contained breathing apparatus] and the KI tablets, the control room operators would be protected such that GDC 19 dose guidelines would be met.[57]

Thus, the nuclear industry and the NRC recognize the value of KI tablets as a protective health measure - at least for plant workers. The petitioners are asking the NRC to extend this same protective health measure to the people living near Indian Point Unit 2.

The absolute need for KI tablets for the public is further evidenced in the less-than-stellar performance demonstrated by Con Ed during the two emergencies at Indian Point Unit 2 in the past eight months.

> Several equipment malfunctions were experienced over the duration of the emergency response including: failure of the Emergency Response Data System for the first few hours of the event due to a telephone line failure, and improper operation of the data communication link from the fixed offsite radiation monitor data collection system. [58]

> Performance in the emergency preparedness area continued to exhibit some weakness as evidenced by EP [emergency preparedness] problems that were identified during the August 1999 reactor trip event.[59]

These repetitive emergency response problems suggest strongly that Con Ed will not be able to take timely and effective actions to protect public health in event of a radioactive release from Indian Point Unit 2. With prior distribution of KI tablets to the neighboring residents or stockpiling in the region, members of the public have significantly greater odds of avoiding adverse health consequences in event of an accident.

The petitioners request that Indian Point Unit 2 not be permitted to restart until members of the public are afforded the prudent health measure of KI tablets.

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[57] Alexander W. Dromerick, Senior Project Manager, Nuclear Regulatory Commission, to Charles H. Cruse, Vice President - Nuclear Energy, Baltimore Gas and Electric Company, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit No. 1 (TAC No. M97855) and Unit No. 2 (TAC No. M97856)," May 23, 1998.
[58] Consolidated Edison Company of New York, Inc., Licensee Event Report No. 2000-001-00, "Manual Reactor Trip Following Steam Generator Tube Rupture," March 17, 2000.
[59] A. Randolph Blough, Director - Division of Reactor Projects, Nuclear Regulatory Commission, to A. Alan Blind, Vice President -- Nuclear Power, Consolidated Edison Company of New York, Inc., "Mid_Cycle Plant Performance Review - Indian Point Unit 2," September 30, 1999.

CONCLUSION

In January 1986, the space shuttle *Challenger* exploded shortly after launch, killing seven astronauts. The cause of the explosion was traced to hot gases burning past two O-rings to ignite a fuel tank.

In 1985, seven of the nine shuttles that were launched experienced erosion and/or blowby of the O-rings. The potentially catastrophic consequences of the O-ring problems were well known, but the high frequency of their occurrence led NASA to believe that the problems were tolerable.

The process through which NASA tolerated O-ring problems is directly analogous to the process through which the NRC is tolerating steam generator tube degradation problems:

> [This book] shows how signals of potential danger can be normalized so that action becomes aligned with organizational goals. At a fundamental level, it exposes the incrementalism of most life in organizations and the way that incrementalism can contribute to extraordinary event that happen.
>
> This case directs our attention to the relentless inevitabilities of mistakes in organizations - and the irrevocable harm that can occur when those organizations deal in risky technology. [60]

The intentions of Con Ed and the NRC are probably as honorable as those of NASA before the *Challenger*'s launch. The petitioners hope that the NRC staff will grant all three items requested in our petition so that safety concerns at Indian Point Unit 2 can be fixed <u>before</u> they grow to tragic proportions.

It would seem extremely difficult, if not impossible, for Con Ed and the NRC to tell people after a serious accident at Indian Point Unit 2 involving the steam generators that everything possible was done to protect them when there are undamaged steam generators sitting at the site ready to be used.

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[60] Diane Vaughan, "The Challenger Launch Decision," The University of Chicago Press, 1996